UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, ILLINOIS 60532-4352

October 31, 2006

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

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORTS 05000254/2006006;
05000265/2006006

Dear Mr. Crane:

On September 30, 2006, 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 3, 2006, 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 four issues of very low safety significance (Green). All of these issues involved violations of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the corrective program, the NRC is treating these findings and issues 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 Regulation 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,

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

Docket Nos. 50-254; 50-265; 72-053
License Nos. DPR-29; DPR-30

Enclosure:
Inspection Report 05000254/2006006; 05000265/2006006
w/Attachments:
1. Supplemental Information
2. Confirmatory Measurements Comparison Criteria
3. Tritium Sample Results

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
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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|---|---|
| Docket Nos.: | 50-254, 50-265 |
| License Nos.: | DPR-29, DPR-30 |
| Report Nos.: | 05000254/2006006; 05000265/2006006 |
| Licensee: | Exelon Nuclear |
| Facility: | Quad Cities Nuclear Power Station, Units 1 and 2 |
| Location: | Cordova, Illinois |
| Dates: | July 1, 2006, through September 30, 2006 |
| Inspectors: | K. Stoedter, Senior Resident Inspector<br>M. Kurth, Resident Inspector<br>J. McGhee, Reactor Engineer<br>R. Orlikowski, Senior Resident Inspector - Duane Arnold<br>R. Ganser, Illinois Emergency Management Agency<br>S. Orth, Health Physics Program Manager<br>W. Snell, Senior Health Physicist<br>E. Bonano, Health Physicist |
| Approved by: | M. Ring, Chief<br>Projects Branch 1<br>Division of Reactor Projects |

Enclosure

## SUMMARY OF FINDINGS

IR 05000254/2006006, 05000265/2006006; 07/01/2006 - 09/30/2006; Quad Cities Nuclear Power Station, Units 1 & 2; Event Followup and Other Activities.

The report covered a 3-month period of inspection by resident and regional inspectors. Four Green findings, with associated 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-revealed finding was identified due to the discovery of two separate Unit 1 feedwater leaks in July 2006. The leaks were caused by poor maintenance work practices and resulted in two unplanned power reductions. The failure to implement and maintain procedures governing power operations contributed to the leak creation and resulted in a Non-Cited Violation of Technical Specification 5.4.1. Immediate corrective actions included securing the feedwater pumps, performing leak repairs, revising the appropriate procedures, and conducting a review of maintenance work practices.

  This finding was more than minor because, if left uncorrected, the poor maintenance work practices and procedural compliance issues could become a more significant safety issue and result in other equipment degradation. This finding was of very low safety significance because the feedwater leaks did not contribute to both the likelihood of an initiating event and that mitigating systems equipment would not be available. This finding affected the work practices component of the human performance cross-cutting area. Specifically, the licensee failed to ensure that supervisory and management oversight of work activities was appropriate to support nuclear safety. (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, due to the licensee's failure to correct a condition adverse to quality. Specifically, the licensee had not implemented timely actions to correct the repeated inability of the main steam safety valves to actuate within Technical Specification values. Immediate corrective actions for this issue included developing a schedule for submitting a required Technical Specification change, determining if the Target Rock valve was suitable for its current application, and reviewing the factors that contributed to the licensee's lack of timeliness.

This issue was more than minor because it affected the mitigating systems objective of ensuring the reliability of systems that respond to initiating events. This finding was of low safety significance because the valve performance did not cause the reactor vessel overpressure limits to be exceeded, did not adversely impact automatic depressurization system operation, and did not significantly impact the licensee's response to an Appendix R event. This finding was attributable to the corrective action program component of the problem identification and resolution cross cutting area. Specifically, the licensee failed to take actions to address this adverse trend in a timely manner commensurate with its significance and complexity. (Section 4OA3.3)

- Green. A self-revealing finding and a Non-Cited Violation of Technical Specification 5.4.1 were identified when operations performed activities which resulted in the unexpected start of the Unit 1 emergency diesel generator. The unexpected actuation was caused by the failure to follow procedures. Immediate corrective actions included discussing this issue with operations personnel, reinforcing procedural adherence and equipment status requirements, and formalizing the use of the "Procedures in Progress" book.

  This issue was more than minor because if left uncorrected, it could result in future risk-significant configuration control issues. This finding was of very low safety significance because it did not result in an actual loss of safety function. This finding impacted the work practices component of the human performance cross cutting area. Specifically, the licensee failed to maintain compliance with OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel." (Section 4OA3.4)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50.65 due to the licensee's failure to demonstrate that the performance and condition of the turbine building internal flooding protection check valves was being effectively controlled through the performance of appropriate preventive maintenance. Immediate corrective actions included assessing the current check valve condition and implementing actions to correct the common mode failure.

  This finding was more than minor because it was left uncorrected and resulted in significant check valve degradation. This finding was of very low risk significance because the valve failures did not result in a total loss of the residual heat removal service water system. In addition, the failures did not result in an actual loss of safety function for risk significant maintenance rule equipment. (Section 4OA5.1)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

## REPORT DETAILS

### Summary of Plant Status

Unit 1 operated at maximum achievable power levels during the inspection period with the following exceptions:

- On July 2 operations personnel performed a planned load drop to approximately 70 percent power to allow recovery of a previously inoperable control rod and to perform traveling screen maintenance.
- On July 3 operations personnel reduced reactor power to 85 percent (unplanned load drop) due to discovering a leak on the 1B reactor feedwater pump discharge drain line (see Section 4OA3.1 for details).
- On July 19 operations personnel conducted an additional unplanned load drop to 85 percent to repair a 1A reactor feedwater pump flow venturi instrument sensing line leak (see Section 4OA3.1 for details).
- Between July 31 and August 3, several power reductions were required due to extremely hot Mississippi River temperatures and reduced river flow.
- On September 20 operations personnel conducted an emergency power reduction to 80 percent due to the loss of the high pressure feedwater heaters. Unit 1 operated at reduced power levels for approximately 20 hours.
- Additional planned power reductions were conducted to allow for routine turbine testing, control rod sequence exchanges, and main condenser flow reversals.

Unit 2 also operated at maximum achievable power levels with the following exceptions:

- Between July 31 and August 3, several power reductions were required due to extremely hot Mississippi River temperatures and reduced river flow.
- On August 8 operations personnel conducted an unplanned load drop to 85 percent due to increased vibrations on the 2B reactor feedwater pump inboard motor bearing. Unit 2 stayed at reduced power levels until August 13.
- Additional planned power reductions were conducted to allow for routine turbine testing, control rod sequence exchanges, and main condenser flow reversals.

## 1. REACTOR SAFETY

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

### 1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During July and August 2006, the licensee experienced multiple periods of extreme hot weather. Outside air temperatures exceeded 100$^0$F and Mississippi River temperatures reached greater than 91$^0$F. The abnormally high temperatures resulted in alarms being received for multiple pieces of equipment including the main power transformers, the reactor recirculation motor generator sets, and the feedwater pumps. Increased temperatures were also experienced in both drywells. The inspectors conducted frequent control room tours to remain cognizant of control room alarms, equipment status, and unexpected changes in conditions. The inspectors reviewed the licensee's hot weather, annunciator response, and abnormal operating procedures to ensure that the hot weather inspections were being performed as required and that the actions to be taken if a temperature setpoint was exceeded were clearly stated. The inspectors also reviewed the status of mitigating systems equipment to ensure that the equipment was available to perform its safety function. Lastly, the inspectors reviewed the licensee's risk information to verify that the licensee had considered the impact that the hot weather could have on initiating event frequency.

This item represented the completion of one inspection sample.

b. Findings

No findings of significance were identified.

### 1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown of the systems listed below to verify the operability of redundant or diverse trains and components when equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system and potentially increase risk. The inspectors reviewed operating procedures, walked down systems components, and verified that selected breakers, valves, and support equipment were in the correct position. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems by reviewing the resolution of previously initiated issue reports.

- Unit 1/2A Diesel Fire Pump;
- 2B Core Spray;
- Unit 1/2A Standby Gas Treatment; and
- Unit 2 Reactor Building Closed Cooling Water.

These actions represented the completion of four partial walkdowns.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Protection - Tours

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; that fire detection and suppression equipment was available for use and access was not obstructed; that passive fire barriers were maintained in good material condition; that procedures were maintained and adequate to support fire fighting activities; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan.

- Fire Zone 1.1.1.4 - Unit 1 Reactor Building, 647 Feet, Third Floor;
- Fire Zone 5.0 - Unit 2 Turbine Building, Safe Shutdown Pump Room, 595 Feet Elevation;
- Fire Zone 7.1 - Unit 1 Turbine Building, 250 V Battery Room, 628 Feet Elevation;
- Fire Zone 7.2 - Unit 2 Turbine Building, 250 V Battery Room, 628 Feet Elevation;
- Fire Zone 8.2.1.B - Unit 2 Turbine Building, Condensate Pump Room, 547 Feet Elevation;
- Fire Zone 8.2.6.A - Unit 1 Turbine Building, 595 Feet, 4kV Switchgear and U-1 Trackway;
- Fire Zone 8.2.6.E - Unit 2 Turbine Building, 4KV Switchgear & Trackway Area, 595 Feet Elevation;
- Fire Zone 8.2.7.C - Unit ½ Closed Cooling Water Area, 612 Feet Elevation; and
- Fire Zone 9.3 - Unit ½ Reactor Building, 595 Feet, ½ Diesel Generator.

These inspections represented the completion of nine quarterly samples.

b. Findings

No findings of significance were identified. Minor issues identified during the inspection were documented in Issue Report 527851.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On July 10, 2006, the inspectors observed operations crews respond to two separate simulator scenarios. The first scenario consisted of a local power range monitor failure, a large steam line break, and flooding of the reactor pressure vessel. The second scenario consisted of a reactor recirculation pump seal leak, a small break loss of coolant accident, a high drywell temperature condition, a loss of reactor water level instrumentation, and flooding the reactor pressure vessel.

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.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in the following documents:

- OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel";
- OP-AA-103-102, "Watchstanding Practices";
- OP-AA-103-104, "Reactivity Management Controls"; and
- OP-AA-104-101, "Communications."

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 these observations were noted by the evaluators and discussed with the respective crews.

Performance of this inspection represented the completion of two inspection samples.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

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 system listed below. This system was selected based on it being designated as risk significant under the Maintenance Rule, being in increased monitoring, or due to an inspector identified issue or problem that potentially impacted system work practices, reliability, or common cause failures.

- Reactor Recirculation (Function Z0202)

The inspectors review included an examination of specific system 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 review represented the completion of one inspection sample.

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 work week activities listed below to verify that the appropriate risk assessments were performed prior to removing equipment from service. 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 catergories in accordance with procedures.

- Work Week 27 including emergent maintenance on the 1B feedwater pump and planned maintenance on the 2D residual heat removal pump;
- Work Week 29 including planned maintenance on the Unit 1 high pressure coolant injection system, the 2A core spray system, and the Bus 13 to Bus 23 cross-tie breaker and emergent maintenance on the 1A reactor feedwater pump and the Unit 2 traversing incore probe system;
- Work Week 31 including emergent high temperature conditions resulting in Unit 1 and Unit 2 intermittent power reductions;
- Work Week 33 including planned maintenance on the 1/2A fire pump, the Unit 2 high pressure coolant injection system, and the Unit 2 24/48 Volt battery chargers and emergent work on the 2B reactor feedwater pump;
- Work Week 36 including planned maintenance on the 1B instrument air compressor, the Unit 2 reactor core isolation cooling system, the 2B core spray system, and the Unit 2 emergency diesel generator; and
- Work Week 37 including planned maintenance on the 1/2A standby gas treatment system, the Unit 1/2 emergency diesel generator, the 1B instrument air compressor, and the Unit 2 station blackout diesel generator.

This review represented the completion of six inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated the technical adequacy of the evaluations listed below to ensure that Technical Specification (TS) 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 (UFSAR) to verify that the system or component remained available to perform its intended function. 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 505156 - Breaker Access Panels Are Removed and Issue Report 510523 - Breaker Installed During Q2R18 Incorrectly;
- Issue Report 507290; Unit 2 Diesel Generator Cooling Water Pump Flow Indicates Lower Than Expected;
- Issue Report 513685 - Electrohydraulic Control Fluid Reservoir Level Decrease;
- Issue Report 524997 - High Pressure Coolant Injection Gland Seal Hotwell Pump Cycled Repeatedly;
- Operability Evaluation 523803-02 - Target Rock Safety Relief Valve Accumulator Check Valve; and
- Issue Report 527348 - Reactor Core Isolation Cooling Minimum Flow Valve Exceeded Inservice Testing Acceptable Range.

These reviews constituted the completion of six inspection 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 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).

- QCOS 7500-05 - Standby Gas Treatment System Operability following the replacement of the 1/2B standby gas treatment fan;
- QCOS 1400-01 - Core Spray Pump Flow Rate Test and QCOS 1400-08 - Core Spray Valve Timing Test following planned maintenance;

- QOP 3200-03 - Starting the Second and Third Reactor Feedwater Pump following repairs to the 2B feedwater pump motor bearing;
- QCOS 1000-04 - Residual Heat Removal Service Water (RHRSW) Pump Operability Test following maintenance on the 1D RHRSW pump discharge elbow;
- QCOS 0010-07 - Equipment External Leak Check following the replacement of residual heat removal valve 2-1053-F; and
- QCOS 6620-01 - Station Blackout Diesel Generator Quarterly Load Test following the completion of Work Order 851373 - Unit 2 SBO Diesel Governor Control Erratic.

This inspection represented the completion of six samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed the surveillance tests and/or reviewed test data for the tests listed below, to assess whether the structures, systems, and components met the requirements specified in the TSs, the UFSAR, and Section XI of the American Society of Mechanical Engineers Code. The inspectors also determined whether the testing effectively demonstrated that the structures, systems, and components were operationally ready and capable of performing their intended safety functions.

- QCOS 1000-04 - Residual Heat Removal Service Water Pump Operability Test;
- QCOS 1000-06 - B Loop Residual Heat Removal Low Pressure Coolant Injection Mode Flow Rate Test;
- QCOS 1600-07 - Reactor Coolant Leakage in the Drywell;
- QCOS 1600-31 - Suppression Pool Temperature Monitoring;
- QCOS 2300-05 - Unit 1 High Pressure Coolant Injection Pump Flow Test;
- QCOS 2300-05 - Unit 2 High Pressure Coolant Injection Pump Flow Test;
- QCOS 6620-01 - Unit 2 Station Blackout Diesel Generator Quarterly Load Test; and
- QCOS 6600-41 - Unit 1 Emergency Diesel Generator Load Test.

These inspections represented the completion of four in-service tests, one reactor coolant system leakage detection surveillance, and three routine tests.

b. Findings

No findings of significance were identified.

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 requirements specified in the UFSAR and the TSs. The comparison was performed to verify that the modification did not affect operability or availability of the affected system. The inspectors also walked down the modification to ensure that it was installed in accordance with the design documents. Lastly, the inspectors observed the licensee perform post-installation testing to ensure that the modification accomplished its intended purpose.

- Temporary Modification 361329 - Install Supplemental Cooling Units for the 1A and 1B Reactor Recirculation Motor Generator Sets

This inspection represented the completion of one sample.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

**Cornerstone: Public Radiation Safety**

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program (71122.03)

.1 Reviews of Radiological Environmental Monitoring Reports, Data and Quality Control

a. Inspection Scope

The NRC performed a number of confirmatory measurements of water samples to evaluate the licensee's proficiency in collecting and in analyzing water samples for tritium and other radioactive isotopes. The samples were collected independently by the inspectors and by licensee personnel and sent to the NRC's contract laboratory for the analysis of tritium. The NRC and licensee obtained these samples from surface water and groundwater sampling points identified in the licensee's Radiological Environmental Monitoring Program and from onsite and offsite groundwater monitoring wells. In particular, samples were obtained as part of the licensee's environmental study of tritium, potential groundwater contamination, and residual onsite contamination from historical leaks and spills (ADAMS ML062760010). While tritium was the primary radionuclide of concern, selected samples were also analyzed for gamma emitting radionuclides and for strontium. The inspectors performed these reviews to assess the licensee's analytical detection capabilities for radio-analysis of environmental samples and its ability to accurately quantify radionuclides to an acceptable level of sensitivity. The criteria used to compare the sample results is provided in Attachment 2, and the results of the comparisons between the NRC and licensee results is provided in Attachment 3.

The inspectors considered the following activities in evaluating the cause of any comparisons that did not result in an agreement:

- re-analysis by licensee or NRC's contract laboratory;
- review of licensee's interlaboratory cross check program results; and
- review of data for any apparent statistical biases.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

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," the inspectors reviewed all items entered into the corrective action program during the inspection period. No new findings or adverse trends were identified.

4OA3 Event Followup (71153)

.1 Review of Unit 1 Feedwater Pump Operational Activities

a. Inspection Scope

The inspectors interviewed operations and engineering personnel, reviewed procedures and control room instrumentation information, attended multiple licensee meetings, and assessed the information contained in the licensee's corrective action documents to determine the sequence of events which led to two separate degradations of the Unit 1 feedwater system.

This inspection represented the completion of two event response samples.

b. Findings

Introduction: A self-revealed Green finding was identified due to the discovery of two separate Unit 1 feedwater leaks. The leaks resulted in two unplanned power reductions during the period. The licensee determined that poor maintenance work practices caused the leaks to develop. However, the failure to appropriately implement and maintain procedures governing power operations also contributed to the creation of the leaks. The failure to properly implement and maintain the power operations procedure constituted a Non-Cited Violation of TS 5.4.1.

Description: On July 2, 2006, operations personnel reduced Unit 1 reactor power to approximately 70 percent to allow the recovery of a control rod and planned maintenance on the traveling screen system. One day prior to the power reduction, the shift manager recognized that total feedwater flow would drop to less than 9.8 million pounds per hour (Mlb/hr) for approximately 2.5 hours. Procedural guidance provided in

QCGP 3-1, "Reactor Power Operations," directed operations personnel to shut down one of the feedwater pumps when total feedwater flow was less than, or equal to, 9.8 Mlb/hr. This direction was provided to increase plant efficiency and to prevent the feedwater system from experiencing increased vibrations due to the operation of three feedwater pumps at reduced flow conditions. However, the shift manager was also concerned that shutting down and re-starting the feedwater pump in a relatively short time frame could subject the pump seals to thermal stresses and cause a potential seal failure.

The shift manager discussed his concerns with other operations and maintenance personnel. Based upon the feedback provided, the shift manager decided that all three feedwater pumps would be kept in operation during the power reduction. The operations field supervisor was assigned to perform increased monitoring of the feedwater pumps during the reduced feedwater flow conditions. This decision was discussed with the remaining members of the operations crew prior to implementation.

Operations personnel began the power reduction at midnight on July 2. At 1:05 a.m. the control room operators determined that total feedwater flow was less than 9.8 Mlb/hr. The field supervisor implemented his increased monitoring of the feedwater pumps as feedwater flow continued to decrease to 8.35 Mlb/hr due to the power reduction. The increased monitoring continued until 3:35 a.m. when total feedwater flow was restored to greater than 9.8 Mlb/hr.

At 2:00 a.m. on July 3, operations personnel identified that the 1B feedwater pump upstream discharge drain line was leaking. Licensee personnel developed a leakage monitoring plan the same day. Within hours of implementing the plan, the licensee identified a significant increase in the leakage rate. Based upon the increased leakage, and the potential for a drain line failure, operations personnel shut down the 1B feedwater pump to repair the drain line. After cutting and capping the drain line, the 1B feedwater pump was returned to service on July 4.

<u>1A Feedwater Upstream Flow Sensing Line Leak</u>

On July 10, 2006, operations personnel identified a small leak (about 10 drops per minute) coming from the 1A feedwater upstream flow sensing line root isolation valve. This valve was located approximately 20 feet in the air and was not easily accessible. The valve's inaccessibility drove operations to report that the leak appeared to be a packing leak.

One week later, engineering and radiation protection personnel accessed the root isolation valve locally. Through direct visual inspection of the valve, the engineer determined that the leak had originated from a welded connection immediately downstream of the valve rather than from the packing. The licensee immediately implemented a leakage monitoring plan to assess the leak's severity. Over a 12 hour period, the leakage rate increased from 600 milliliters/hour to 1000 milliliters every 8 minutes. Operations personnel immediately shut down the 1A feedwater pump upon being informed of the sharp increase in leakage rates. Maintenance personnel corrected the condition by performing a weld repair.

Apparent and Contributing Causes

The licensee conducted an extensive apparent cause evaluation of both feedwater leaks. This evaluation included a visual inspection of the 1A feedwater flow sensing line weld area and a metallurgical analysis of the 1B feedwater pump discharge header upstream isolation piping. The licensee determined that poor maintenance work practices had caused both leaks. Specifically, a lack of fusion during welding in 1998 had caused the 1A feedwater flow sensing line leak. The 1B leak was caused by excessive grinding of the piping during a May 2006 valve replacement.

In addition to the poor maintenance work practices, the inspectors and the licensee independently identified that procedure adherence weaknesses had contributed to the development of both leaks. Quad Cities Station was designed with the capability to reverse flow in the main condensers when condenser back pressure reached elevated levels. QCOP 4400-09, "Circulating Water System Flow Reversal," directed operations personnel to reduce condenser back pressure to specified levels prior to beginning flow reversal activities. During extreme summer conditions, operations personnel were required to reduce condenser back pressure by lowering reactor power levels. In certain cases, the reactor power reductions resulted in total feedwater flow decreasing to less than 9.8 Mlb/hr.

The inspectors reviewed the control room operating logs and the circulating water flow reversal data sheets for both units for the period of June 1 through August 3, 2006. The inspectors determined that operations personnel had reduced reactor power, and decreased total feedwater flow to less than 9.8 Mlb/hr, approximately seven times during the period. However, no actions had been taken to shut down a feedwater pump as directed by QCGP 3-1. In addition, the licensee failed to initiate a procedure change to QCGP 3-1 to allow the third feedwater pump to remain in operation during flow reversal activities. The inspectors discussed historical flow reversal actions with operations and engineering personnel and learned that the operations department routinely left the third feedwater pump in service during flow reversal activities. The inspectors and the licensee concluded that the increased system vibrations caused by the failure to secure a feedwater pump during reduced flow conditions contributed to the overall degradation of the 1A and 1B feedwater piping.

Analysis: The inspectors determined that the poor maintenance work practices, and the failure to follow or change QCGP 3-1, were more than minor because if left uncorrected, these practices could become a more significant safety issue. The inspectors performed a Phase 1 Significance Determination Process evaluation in accordance with Inspection Manual Chapter 0609. The inspectors determined that this finding was of very low safety significance (Green) because the feedwater leaks did not contribute to both the likelihood of an initiating event and the likelihood that mitigating equipment would not be available. This finding also affected the work practices component of the human performance cross-cutting area. Specifically, the licensee failed to ensure that supervisory and management oversight of work activities was appropriate to support nuclear safety.

Enforcement: Technical Specification 5.4.1 required that written procedures be established, implemented, and maintained for those activities recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Regulatory

Guide 1.33, Appendix A, Section 2.g required that written procedures be established, implemented, and maintained for power operations. QCGP 3-1, "Reactor Power Operations," was the licensee's procedure used to demonstrate compliance with Section 2.g of RG 1.33 and TS 5.4.1. Step F.2.d stated, "When total feedwater flow is less than, or equal to, 9.8 Mlb/hr, then enter two reactor feedwater pump operation per QCOP 3200-05." Contrary to the above, prior to August 3, 2006, operations personnel operated the plant with total feedwater flow less than 9.8 Mlb/hr multiple times and failed to enter two reactor feedwater pump operation per QCOP 3200-05. The historic failure to secure the feedwater pumps contributed to the creation of two leaks and resulted in two unplanned power reductions. However, because this issue is of very low safety significance and has been entered into the corrective action program as Issue Reports 505944, 506893, 513344, and 529584, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2006006-01)**. Corrective actions for this issue included repairing the feedwater piping, revising QCGP 3-1 and other associated procedures, providing additional training on procedure adherence to operations personnel, and developing actions to address the maintenance work practices issues.

.2 <u>(Closed) Licensee Event Report 05000265/05-001</u>: Core Spray Discharge Pressure Switch Out of Calibration due to Apparent Manufacturing Defect.

On May 25, 2005, the licensee conducted routine surveillance testing and identified that pressure switch 2-1462-B was inoperable. This pressure switch was used to satisfy the portion of the automatic depressurization system initiation logic which required at least one low pressure emergency core cooling system pump to be in operation prior to depressurizing the reactor vessel. The licensee determined that the recently-installed switch had become inoperable due to an internal manufacturing defect which was unable to be detected prior to installation. Corrective actions included replacing the pressure switch, informing the switch vendor of the manufacturing problem, and instituting a new switch receipt inspection to ensure that no additional switches were received with a similar defect. This issue was more than minor because it impacted the mitigating systems attribute and objective of ensuring the availability and reliability of equipment to respond to initiating events. The issue was of very low safety significance (Green) because the pressure switches on the remaining low pressure emergency core cooling system pumps would have successfully satisfied the automatic depressurization system initiation logic. This licensee-identified finding involved a violation of TS 3.3.5.1, "ECCS Instrumentation." The enforcement aspects of the violation are discussed in Section 4OA7 of this report.

.3 <u>(Closed) Licensee Event Report 05000265/06-001</u>: Two Main Steam Safety Valves and Two Main Steam Safety/Relief Valves Outside of TS-Allowed Tolerance.

<u>Introduction</u>: A Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, were identified due to the failure to correct a condition adverse to quality. Specifically, the licensee had not implemented actions to correct the repeated inability of the main steam safety valves to actuate within plus or minus 1 percent of the setpoint as required by TS 3.4.3 and TS Surveillance Requirement (TSSR) 3.4.3.1.

<u>Description</u>: The main steam systems for Quad Cities Units 1 and 2 were designed with nine main steam safety valves. Under certain conditions, operations personnel were

directed to use these valves to ensure that the reactor pressure vessel was not subjected to an extreme high pressure condition. One of the main steam safety valves, the Target Rock valve, was also utilized as part of the automatic depressurization system.

Technical Specification 3.4.3 required all nine main steam safety valves to be operable when the respective reactor was operated in Modes 1, 2, or 3. To maintain main steam safety valve operability, the licensee utilized a two-phase approach. First, each valve was tested prior to installation to verify that it met the TS requirements. Second, the licensee tested each valve after removing it from the plant to demonstrate that it remained capable of actuating within the TS values.

In early 2004, the inspectors identified a concern regarding the licensee's historical inability to comply with the TS requirements listed above. The inspectors documented this concern as part of Inspection Report 2004002 which was issued on April 19, 2004.

On June 18, 2004, the licensee submitted Licensee Event Report (LER) 50-265/04-004 to document the failure of the Unit 2 Target Rock valve to meet TS 3.4.3 and TSSR 3.4.3.1. Within the LER, the licensee provided information regarding previous examples of the Target Rock valve failing to meet TSs. However, no information was provided regarding previous examples of other main steam safety valves failing to meet TSs. The inspectors brought this oversight to the licensee's attention. The licensee agreed to revise the LER based upon the inspectors' feedback.

On July 23, 2004, the licensee initiated Issue Report 238434 to document the concerns raised in Inspection Report 2004002. The licensee found that during initial plant licensing the TSs and the American Society of Mechanical Engineers (ASME) Code required the main steam safety valves to actuate within plus or minus 1 percent of the setpoint. During the late 1980's or early 1990's, the ASME Code was revised to allow actuation of the main steam safety valves to occur within plus or minus 3 percent of the setpoint. However, the licensee decided against pursuing a TS change which would have made TSSR 3.4.3.1 consistent with the ASME Code. Corrective actions for Issue Report 238434 included requesting an amendment to TS 3.4.3 and TSSR 3.4.3.1 by July 29, 2005.

The inspectors closed the concern identified in Inspection Report 2004002 by documenting a Green Non-Cited Violation of TS 3.4.3 in Inspection Report 2004010. The inspectors reviewed historical main steam safety valve test results and found that at least two or more of the nine valves failed to meet TSSR 3.4.3.1 during each of the last six refueling outages. Corrective actions included installing main steam safety valves which were certified to actuate within the TS values, submitting a TS amendment to change the valve operating tolerances to plus or minus 3 percent, and revising LER 50-265/04-004 to include the previous failures.

On June 6, 2005, the licensee submitted LER 50-254/05-003 which documented that three Unit 1 main steam safety valves tested outside of the TS requirements. As in previous LERs, the corrective actions included replacing the safety valves with certified valves and pursuing the previously discussed TS amendment.

On July 25, 2005, the licensee submitted a revision to LER 50-265/04-004 which included information regarding the previous main steam safety valve failures. In addition, the licensee stated that they were pursuing a revision to TS 3.4.3 to incorporate the ASME Code allowable values. Four days later, the licensee closed the corrective action assignment tracking the TS amendment to a regulatory commitment assignment. The inspectors reviewed LS-AA-125, "Corrective Action Program," and determined that closing the corrective action assignment to a regulatory commitment was not allowed by procedure. Specifically, Section 4.8.1 of LS-AA-125 stated that corrective action assignments could only be closed to another corrective action assignment or after the defined action was completed. The inspectors were also aware of a previously performed Monte Carlo analysis which demonstrated that the actual main steam safety valve performance would not have resulted in exceeding the design basis reactor pressure vessel overpressure limit. The inspectors concluded that the inappropriate closure of the corrective action discussed above, and the satisfactory results of the Monte Carlo analysis, resulted in the licensee believing that the restoration of the licensing basis through the pursuit of a TS amendment or other actions was not required to be completed in a timely manner.

On April 10, 2006, the licensee determined that two main steam safety valves and the Target Rock valve (all installed in April 2005) failed to meet TSSR 3.4.3.1. In addition, a Target Rock valve installed in March 2004 and removed in April 2005 also failed to actuate within the TS allowed values. This issue was communicated to the NRC via LER 50-265/06-001 on June 9, 2006. The inspectors reviewed LER 50-265/06-001 and identified the following:

- Both Target Rock valves actuated at greater than 3 percent above the setpoint;
- Corrective actions consisted of pursuing changes to TS 3.4.3 and TSSR 3.4.3.1; and
- The proposed TS change would not have addressed the most recent Target Rock valve performance.

On June 20, 2006, the inspectors began discussing their concerns regarding LER 50-265/06-001 with licensee management. The inspectors were particularly concerned with the licensee's lack of timeliness regarding the previously proposed TS amendment and what other actions were going to be implemented to ensure that the Target Rock valves were suitable for their current application. Significant interactions with licensee management were required to ensure that the licensee appreciated the importance of maintaining their licensing basis. These discussions continued until September 26, 2006, when the licensee initiated three issue reports to document the inspectors' concerns.

The inspectors reviewed multiple documents to determine if the licensee's failure to correct the repeated inability to meet TS 3.4.3 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 upon the examples provided in the Topical Report, the inspectors concluded that the repeated inability to meet TS 3.4.3 was an adverse trend and a nonconformance with specified requirements which was required to be promptly identified and corrected.

Analysis: This issue was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone. In addition, the failure affected the mitigating systems objective of ensuring the operability and reliability of systems that respond to initiating events to prevent undesirable consequences. The inspectors conducted a Phase 1 Significance Determination Process screening and determined that a Phase 2 evaluation was needed as this finding impacted both the mitigating systems and the barrier integrity cornerstones.

The inspectors used the Risk-Informed Inspection Notebook for Quad Cities Nuclear Power Station, Units 1 and 2, Revision 2, dated September 30, 2005, to complete the Phase 2 evaluation. The actual dates of the valves failing to meet TSs could not be determined since these valves cannot be tested during reactor operation. Because of this, the inspectors assumed that the finding occurred half-way through the operating cycle (or approximately 1 year). This resulted in a fault exposure time of greater than 30 days. For each Significance Determination Process worksheet completed, the inspectors assumed that all other mitigating systems equipment was available. The inspectors allowed credit for recovery since the valves would have actuated once the pressure applied exceeded the setpoint. Using these assumptions, the inspectors evaluated one core damage sequence on the anticipated transients without scram worksheet. The result of this sequence was eight points. Based upon the counting rule, the overall increase in risk and safety was determined to very low (Green).

The inspectors also requested that a regional senior reactor analyst conduct a Phase 3 evaluation since actuation of the Target Rock valve was credited as part of the licensee's Appendix R analysis. The analyst determined that the increase in the Target Rock valve's lift pressure would not cause a significant change in the time to reach top of active fuel or in the torus water temperature response during an Appendix R event. The results of the Phase 3 evaluation determined that this was a finding of very low safety significance. The causes of this finding were determined to be attributable to the corrective action program component of the problem identification and resolution cross cutting area. Specifically, the licensee failed to take actions to address this adverse trend in a timely manner commensurate with its significance and complexity.

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, as of September 30, 2006, the licensee had failed to establish measures to ensure that the repeated inability to meet TS 3.4.3 was promptly corrected. However, because this violation was of very low safety significance, and because the issue was entered into the corrective action program as Issue Reports 529720, 535957, and 535965, the issue is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2006006-02; 05000265/2006006-02)**. Corrective actions for this issue included establishing a schedule for submitting the TS amendment, initiating a review to determine if the Target Rock valves were suitable for the application, and reviewing management decision-making with regards to historical main steam safety valve issues. The licensee planned to develop additional corrective actions following the completion of the management decision-making review.

.4 (Closed) Licensee Event Report 05000254/06-003: Unexpected Start of the Division II Emergency Diesel Generator Due to Failure to Open Test Switch.

Introduction: A Green finding and a Non-Cited Violation of TS 5.4.1 was self-revealed when operations performed activities which resulted in the unexpected start of the Unit 1 emergency diesel generator. The unexpected equipment actuation was caused by operations personnel failing to follow procedures related to the authorities and responsibilities for safe operation.

Description: On May 14, 2006, an unexpected start of the Unit 1 emergency diesel generator occurred when the control switch was placed in the auto condition. At the time of the event, Unit 1 was in an abnormal electrical lineup to allow testing of a newly installed reserve auxiliary transformer. The abnormal lineup resulted in de-energizing Bus 14-1 (one of two Unit 1 safety-related 4kV busses). Power was restored to Bus 14-1 using the cross tie from Bus 24-1.

The inspectors reviewed the sequence of events and determined that Bus 14-1 was initially de-energized on May 13 during the implementation of QCOP 6500-08, "4kV Bus Cross-Tie Operation." QCOP 6500-08 provided operations personnel with two methods which could be used to prevent the emergency diesel generator from auto-starting. The first method called for the manipulation of two specific test switches. The second method allowed the operators to place the emergency diesel generator control switch in the stop position. Operations personnel elected to utilize the control switch method since the switch was already in the stop position as part of a separate clearance order activity. The operators also placed an equipment status tag on the control switch to ensure that the switch would remain in the stop position after the clearance order activity was completed. Operations then placed a copy of QCOP 6500-08 in the "Procedures in Progress" book.

Ten minutes later operations personnel used QCOP 6500-27, "De-energizing 4kV Bus 14 for Maintenance and Re-energizing," to de-energize Bus 14. QCOP 6500-27 also provided two separate methods to prevent the emergency diesel generator from auto-starting. Much like QCOP 6500-08, QCOP 6500-27 defeated the auto-start function by opening a third test switch or by placing the emergency diesel generator control switch in the stop position. Operations personnel chose to defeat the auto-start function through the use of the control switch. A copy of QCOP 6500-27 was placed in the "Procedures in Progress" book. However, an equipment status tag was not placed on the control switch.

During the evening of May 13, operations personnel were informed that the clearance order activities discussed earlier were nearing completion. Operations personnel believed that the Unit 1 emergency diesel generator could be considered operable and available to Unit 2 if the two test switches associated with QCOP 6500-08 were opened and the Unit 1 emergency diesel generator control switch was returned to the auto position. None of the three on-shift senior reactor operators recognized that the third test switch associated with QCOP 6500-27 was also required to be opened. In addition, the senior reactor operators failed to recognize that returning the control switch to the auto position with the associated bus de-energized was not allowed by procedures in use at the time. As a result, the night shift operations crew opened the two test switches discussed in QCOP 6500-08 and removed the equipment status tag.

As part of shift turnover activities on May 14, the off-going night shift operations crew informed the day shift operations crew that "all" of the test switches associated with preventing an auto-start of the Unit 1 emergency diesel generator had been opened. Due to the imprecise communications, neither crew was able to identify that the proper number of test switches had not been opened. As a result, the Unit 1 emergency diesel generator immediately auto-started when the control switch was placed in the auto position later in the shift.

Analysis: The licensee conducted a root cause investigation of this event. The root cause investigation results showed that numerous data points, such as operations logs, the "Procedures in Progress" book, and a verbal turnover were available to ensure that the operations crews understood the status of the electrical busses and the Unit 1 emergency diesel generator. However, the inspectors determined that the imprecise communications during shift turnover activities, an inadequate review of operations logs, an inadequate review of the "Procedures in Progress" book, the inconsistent use of equipment status tags, and the failure to recognize the need for a procedure change led to the inadvertent auto-start of the Unit 1 emergency diesel generator.

The failure of operations personnel to adequately utilize human performance tools to maintain their awareness of equipment status was more than minor because if left uncorrected, it could result in future risk-significant configuration control issues. This finding was of very low safety significance (Green) because it did not result in an actual loss of safety function. This finding impacted the work practices component of the human performance cross cutting area. Specifically, the licensee failed to maintain compliance with OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel," and QCOP 6600-11, "Unit 1 Emergency Diesel Generator Outage Report."

Enforcement: Technical Specification 5.4.1 required that written procedures be established, implemented, and maintained for those activities recommended in RG 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Section 1.b required that written procedures be established, implemented, and maintained governing the authorities and responsibilities for safe operation and shutdown. OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel," was the procedure the licensee established to delineate the authorities and responsibilities for safe shutdown and operation. Step 3.4 of OP-AA-101-111 stated that all on-shift licensed personnel and operating supervisors were to be aware of and responsible for the status of the plant at all times. Steps 4.1.12, and 4.2.5 of OP-AA-101-111 required the shift manager and unit supervisor to ensure that operational activities were performed in accordance with approved procedures. Contrary to the above, Steps 3.4, 4.1.12, and 4.2.5 were not implemented in a manner which assured that all on-shift licensed personnel were aware of the status of the plant and that restoration of the Unit 1 emergency diesel generator was performed in accordance with approved procedures. This resulted in an inadvertent actuation of safety-related equipment. However, because this issue is of very low safety significance and has been entered into the corrective action program as Issue Report 489921, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2006006-03; 05000265/2006006-03)**. Corrective actions for this issue included reinforcing expectations regarding communications, providing additional training on procedural adherence, formalizing the "Procedures in Progress" process, and requiring that the "Procedures in Progress" book be reviewed as part of shift turnover activities.

.5 Unit 1 Loss of High Pressure Feedwater Heaters

a. Inspection Scope

On September 20, 2006, the inspectors observed licensed operations personnel respond to a loss of the Unit 1 high pressure feedwater heaters. The inspectors observed the control room operators, interviewed operations personnel, reviewed the control room logs, procedures, sequence of events recorder, and strip chart recorders to verify that the operators responded as directed by procedures. The inspectors also used the information discussed above, and observed licensee personnel in the outage control center, to assess the actions taken to determine the cause of the heater loss.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000254/2006005-02; 05000265/2006005-02: Evaluate Potential that Internal Flooding Protection Function Should Have Been Classified as a(1).

Introduction: The inspectors identified a Green finding and a Non-Cited Violation of 10 CFR Part 50.65 due to the licensee's failure to demonstrate that the performance and condition of the turbine building internal flooding protection check valves was being effectively controlled through the performance of appropriate preventive maintenance. As a result, the check valves were subjected to a common mode failure mechanism which impacted the ability of the valves to perform their intended function.

Description: In June 2006, the inspectors reviewed Issue Report 482166 which documented that one of the turbine building internal flooding protection check valves failed to seat during testing. The inspectors determined that this failure, and the failure of several other internal flooding check valves, was caused by pieces of the residual heat removal service water (RHRSW) vault sump pump's liner becoming trapped within the check valve.

The inspectors reviewed the licensee's maintenance rule database information to determine how the licensee had characterized the sump pump and check valve failures that had occurred over the past 24 months. On August 10, 2004, the licensee identified that check valve 1-3999-515B failed its surveillance test. The failure had not been assessed against the licensee's condition-based monitoring criteria due to inadequacies in the criteria. In February 2005, the licensee identified that the 1D RHRSW sump pump was pumping significantly less that the pump's design flow rate. The licensee failed to address this issue through the conduct of appropriate maintenance until April 2006 when one of the associated internal flooding check valves failed due to plastic from the sump pump's liner becoming lodged inside the valve. On February 10, 2006, the licensee discovered that both of the internal flooding protection check valves for the 1A RHRSW sump had failed due to the lodging of plastic in the check valve internals. At the time of the failure, the licensee concluded that the sump pump liner had degraded due to an overheating condition caused by a faulty sump pump float switch. The licensee completed a maintenance rule evaluation of this issue on March 22, 2006, and concluded that the check valve failure was a functional failure. However, the failure was deemed to be non-maintenance preventable. In addition, the licensee failed to evaluate whether the internal flooding protection function was being effectively controlled.

The inspectors discussed the conclusions of the March 22, 2006 maintenance rule evaluation with the maintenance rule coordinator and the system engineer. The inspectors were concerned that the licensee had differentiated between the sump pump failure (which was determined to be maintenance preventable) and the check valve failure (which was classified as non-maintenance preventable). The inspectors determined that the licensee's conclusion had resulted in masking the fact that the internal flooding protection function was not being effectively controlled through the performance of preventive maintenance.

Analysis: The inspectors consulted the guidance provided in Appendix D to NRC Inspection Procedure 71111.12 to assess the significance of this finding. The inspectors determined that this finding was more than minor because the condition of the internal flooding check valves had degraded to a point that two individual failures and a functional failure had occurred. In addition, the licensee failed to take action to address the trend in internal flooding check valve performance.

The inspectors reviewed the information provided in Inspection Report 2006005, the results of the licensee's 24 month retroactive review of the internal flooding protection function, and the guidance provided in Appendix D to NRC Inspection Procedure 71111.12. Based upon the information listed above, the inspectors determined that a Category III violation of 10 CFR 50.65 occurred and contributed to the licensee's failure to identify the degraded condition of the internal flooding protection function in a more timely manner.

Appendix D to 71111.12 required the inspectors to assess the significance of Category III maintenance rule violations utilizing the most significant equipment or functional failure. The inspectors performed a Phase 1 significance determination evaluation using the February 10, 2006, internal flooding protection functional failure. The inspectors concluded that this finding was of very low risk significance (Green) since the functional failure would not have resulted in a total loss of RHRSW and was not an actual loss of safety function for equipment considered to be risk significant under the maintenance rule.

Enforcement: Title 10 CFR Part 50.65(a)(1) requires, in part, that holders of an operating license shall monitor the performance or condition of structures, systems, or components (SSCs) within the scope of the rule, as defined by 10 CFR Part 50.65(b), against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

Title 10 CFR Part 50.65(a)(2) states, in part, that monitoring as specified in 10 CFR Part 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function.

Contrary to the above, between August 2004 and June 2006, the licensee failed to demonstrate that the performance or condition of the Unit 1 internal flooding protection check valves had been effectively controlled through the performance of appropriate preventive maintenance. In addition, the licensee had not monitored the performance or condition of the Unit 1 internal flooding protection check valves against licensee established goals. As a result, the licensee had not performed the required goal setting or monitoring. Because this violation is of very low safety significance and has been entered into the corrective action program as Issue Report 500774, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2006006-04)**. Corrective actions for this issue included briefing the remaining engineering staff on the maintenance rule aspects of this issue, generating additional preventive maintenance tasks, and replacing the RHRSW sump pumps with a model that was not prone to shedding plastic.

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 3, 2006. 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 exit meetings were conducted for:

- Public Radiation Safety with Ms. V. Neels on October 12, 2006

4OA7 Licensee-Identified Violations

The following violation of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a Non-Cited Violation.

**Cornerstone: Mitigating Systems**

- Technical Specification 3.3.5.1 required that the emergency core cooling system instrumentation be operable for each function listed in Table 3.3.5.1-1. Table 3.3.5.1-1, Function 4d, required that the core spray pump discharge pressure signals which provided an input to the automatic depressurization system initiation logic be operable. Contrary to the above, on May 25, 2005, the licensee conducted surveillance testing which determined that a pressure switch that provided one of the core spray pump discharge pressure signals to the automatic depressurization system had become inoperable during the previous quarter. This was identified in the licensee's corrective action program as Issue Report 338831. This finding was of very low safety significance because the remaining core spray and residual heat removal discharge pressure signals remained operable and would have satisfied the automatic depressurization system initiation logic.

Attachments:
1. Supplemental Information
2. Confirmatory Measurements Comparison Criteria
3. Tritium Sample Results

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee personnel

T. Tulon, Site Vice President
R. Gideon, Plant Manager
R. Armitage, Training Manager
D. Barker, Work Control Manager
W. Beck, Regulatory Assurance Manager
D. Craddick, Maintenance Manager
D. Moore, Nuclear Oversight Manager
K. Moser, Deputy Engineering 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. Banerjee, NRR Project Manager

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000254/2006006-01 | NCV | Poor Maintenance Work Practices and Procedural Compliance Issues Results in two Feedwater Leaks and two Unplanned Power Reductions |
| 05000254/2006006-02; 05000265/2006006-02 | NCV | Failure to Correct Main Steam Safety Valve TS Issues |
| 05000254/2006006-03; 05000265/2006006-03 | NCV | Unexpected Start of Unit 1 Emergency Diesel Generator |
| 05000254/2006006-04 | NCV | Failure to Effectively Control Condition of Internal Flooding Protection Check Valves Through Performance of Appropriate Preventive Maintenance |

Closed

| | | |
|---|---|---|
| 05000254/2006006-01 | NCV | Poor Maintenance Work Practices and Procedural Compliance Issues Results in two Feedwater Leaks and two Unplanned Power Reductions |

| | | |
|---|---|---|
| 05000254/2006006-02; 05000265/2006006-02 | NCV | Failure to Correct Main Steam Safety Valve TS Issues |
| 05000254/2006006-03; 05000265/2006006-03 | NCV | Unexpected Start of Unit 1 Emergency Diesel Generator |
| 05000254/2006006-04 | NCV | Failure to Effectively Control Condition of Internal Flooding Protection Check Valves Through Performance of Appropriate Preventive Maintenance |
| 05000265/05-001 | LER | Core Spray Discharge Pressure Switch Out of Calibration due to Apparent Manufacturing Defect |
| 05000265/06-001 | LER | Two Main Steam Safety Valves and Two Main Steam Safety/Relief Valves Outside of TS-Allowed Tolerance |
| 05000254/06-003 | LER | Unexpected Start of the Division II Emergency Diesel Generator Due to Failure to Open Test Switch |
| 05000254/2006005-02; 05000265/2006005-02 | URI | Evaluate Potential that Internal Flooding Protection Function Should Have Been Classified as a(1) |

<u>Discussed</u>

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
QOM ½-4100-01; U1/2 Fire Protection Valve Checklist; Revision 10
QOM ½-4100-02; U1/2 Fire Protection Valve Checklist; Revision 12
QCOP 4100-01; Firewater System Lineup for Standby Operation; Revision 4
QOM 2-1400-10; 2B Core Spray Valve Check List; Revision 3
QOM 1//2-7500-01; U1/2 Standby Gas Treatment Valve Check List; Revision 6
QCOP 7500-01; Standby Gas Treatment System Standby Operation and Startup; Revision 18

1R05 Fire Protection
Quad Cities Generating Station Pre-Fire Plans
Quad Cities Generating Station Fire Hazards Analysis
Issue Report 527851; Fire pre-plan Discrepancies; dated September 6, 2006

1R12 Maintenance Effectiveness
Maintenance Rule Performance Criteria for Function Z0202; dated July 31, 2006
Maintenance Rule Expert Panel Scoping Determination for Function Z0202; dated July 31, 2006
Maintenance Rule Evaluation History for Function Z0202 for the Period January - July 2006; dated July 31, 2006
Issue Report 253853; Initiated Manual Lockup of 2B Recirc MG Set; dated September 17, 2004
Issue Report 259905; Failed K4 Relay on 2B Scoop Tube Positioner; dated October 4, 2004
Issue Report 325404; Scoop Tube Lock-up; dated April 16, 2005
Issue Report 246387; 1B Recirc Scoop Tube Locked Up When Cover Removed/Replaced; dated August 22, 2004
Issue Report 339490; 1B Recirc MG Set Scoop Tube Locked Up; dated May 29, 2005
Issue Report 339503; 1B MG Set Scoop Tube Lock Up; dated May 30, 2005
Issue Report 439924; When 1B Reactor Feedwater Pump was Started 1B Reactor Recirc Scoop Tube Locked Up; dated January 8, 2006

1R13 Maintenance Risk and Emergent Work Evaluation
Work Week Safety Profiles
Daily Production Schedules
Completed Risk Factor Charts for specified periods

1R15 Operability Evaluations
Engineering Change 361646; Seismic Evaluation of HFD Breaker Mounting in Direct Current Distribution Panels and Broken Panel Cover Bolts; dated July 20, 2006
Engineering Change 361853; Seismic Assessment of DC Distribution Panels with Missing Breaker Covers for Past Functionality; no date provided
Issue Report 524997; HPCI Gland Seal Hotwell Pump Cycled Repeatedly; dated August 29, 2006
Issue Report 174387; Impact of HPCI Rm Cooler Fan on App K LOCA Analysis; dated August 29, 2003

Issue Report 507290; Unit 2 Diesel Generator Cooling Water Pump Flow Indicates Lower Than Expected; dated July 7, 2006
QCOS 6600-06; Diesel Generator Cooling Water Pump Flow Rate Test; Revision 30
Issue Report 527348; MO 2-1301-60, RCIC Min Flow Exceeded IST Acceptable Range; dated September 5, 2006
ER-AA-321, Revision 6; Administrative Requirements for Inservice Testing

1R19 Post Maintenance Testing
Engineering Change 349792; Evaluate Minimum Terminal Voltage for Replacement Standby Gas Treatment Fan Motors; Revision 0
Engineering Change 344853; Standby Gas Treatment Replacement Fan Motors; Revision 0

1R22 Surveillance Testing
Issue Report 525311; 1C RHRSW Pump IST Adverse Trend; dated August 30, 2006
Issue Report 525317; 1D RHR Pump IST Adverse Trend; dated August 30, 2006
Issue Report 525291; 1C RHR Pump IST Adverse Trend; dated August 30, 2006
Issue Report 525304; 1D RHR Pump IST Adverse Trend; dated August 30, 2006
Issue Report 518891; U2 DWFDS Integrator Run-on; dated August 10, 2006
Issue Report 301545; CM-U Breaker for Drywell Floor Drain Sump Pump; dated February15, 2005
Issue Report 303985; LIS 1-2002-100B Pegged High After Back Blowing Sensing Line; dated February 22, 2005
Issue Report 317459; U1 DWFDS Valve Labeled Incorrectly; dated March 26, 2005
Issue Report 345181; DWFDS High Flow Alarm; dated June 17, 2005
Issue Report 346903; U1 DW Floor Drain Sump High Flow Alarm; dated June 23, 2005
Issue Report 349213; Loss of Math Functions in DWFDS and DWEDS Recorders; dated June 30, 2005
Issue Report 379639; OOT, LI 2-2002-191 Trend Code B4; dated September 29, 2006
Issue Report 337665; U1 EDG Governor Too Responsive During Load Test; dated May 23, 2005
Issue Report 357205; U1 EDG Load Oscillation During Monthly Load Test; dated July 27, 2005
Work Order 815169; U1 EDG Governor Too Responsive During Load Test; dated July 12, 2005
Work Order 851373; Unit 2 SBO Diesel Governor Control Erratic; dated July 24, 2006
Vendor Technical Manual C0115 Vol. 1, EMD Engine Maintenance Manual Section 12, Governor
Vendor Technical Manual C0115 Vol. 1, 645E4 EMD Engine Maintenance Manual Changes Section 11, dated August 1967
Vendor Technical Manual C0115 Vol. 1, Woodward Governor Co. Manual 03040D, UG Dial Governor

1R23 Temporary Modifications
Night Orders; dated June 19, 2006
Issue Report 498279; 1A MG Set Motor Winding High Temperature; dated June 8, 2006
Temporary Interim Change 1518 for QCOP 5750-16, "Reactor Recirculation Motor Generator Set Ventilation System; dated July 13, 2006

4OA3 Event Followup
Issue Report 338831; Inconsistent Operation of ADS Permissive from "B" Core Spray; dated May 26, 2005

QCIS 1400-01; Core Spray Pump Discharge Pressure Calibration and Functional Test; Revision 11
Training Lesson Plans for Mercoid Pressure Switches; Revision 2
VETIP Manual C0055; Mercoid Pressure Switches; Introduction to Pressure and Temperature Swtiches
Work Order 757659; Core Spray Pump Discharge Pressure Cal and Functional Test; dated February 17, 2005
Work Order 754564; Core Spray Hi/Lo Pressure Switch Out of Tolerance; dated February 17, 2005
OP-AA-108-101; Control of Equipment and System Status; Revision 3
Shift Manager Turnover Checklist; dated May 13-14, 2006
Unit Supervisor Turnover Checklist; dated May 13-14, 2006
Unit 1 NSP Turnover Checklist; dated May 13-14, 2006
OP-AA-112-101; Shift Turnover and Relief; Revision 2
OP-QC-108-1002; Procedures in Progress Book Guidance; Revision 0
Engineering Change 362004; Evaluate Unit 1 Feedwater Discharge Small Bore Pipe Vibrations; no date provided
Power Labs Report QDC-10910; Failure Analysis of the 1B Reactor Feed Pump Discharge Header Upstream Drain Line; dated July 21, 2006
Apparent Cause Report 505944; 1B Reactor Feedwater Pump Discharge Drain Header Leak; dated September 14, 2006
Apparent Cause Report 506893; Operations Continued Operation of 3 Reactor Feed Pumps Below 9.8 Mlb/hr Feed Flow; dated September 14, 2006
Apparent Cause Report 513344; Unscheduled Load Drop for 1A Reactor Feed Pump Sensing Line Leak; dated September 14, 2006
QCOA 3500-01; Feedwater Temperature Reduction with Main Turbine Online; Revision 24
Operations Logs; dated September 20, 2006
Sequence of Events Recorder Printout; dated September 20, 2006
QCOA 0400-02; Core Instabilities; Revision 17
Issue Report 534298; Power Reduction due to 1D2 Heater Level Control Valve Closure; dated September 21, 2006
Issue Report 533936; Unit 1 Unexpected High Pressure Heaters Trip; dated September 20, 2006

4OA5 Other Activities
Apparent Cause Report 500774; Maintenance Rule Performance Criterion not Clear; dated August 3, 2006

4OA7 Licensee Identified Violations
Issue Report 338831; Inconsistent Operation of ADS Permissive from "B" Core Spray; dated May 26, 2005
QCIS 1400-01; Core Spray Pump Discharge Pressure Calibration and Functional Test; Revision 11
Training Lesson Plans for Mercoid Pressure Switches; Revision 2
VETIP Manual C0055; Mercoid Pressure Switches; Introduction to Pressure and Temperature Swtiches
Work Order 757659; Core Spray Pump Discharge Pressure Cal and Functional Test; dated February 17, 2005
Work Order 754564; Core Spray Hi/Lo Pressure Switch Out of Tolerance; dated February 17, 2005

## LIST OF ACRONYMS USED

| | |
|---|---|
| ASME | American Society of Mechanical Engineers |
| CFR | Code of Federal Regulations |
| IMC | Inspection Manual Chapter |
| LER | Licensee Event Report |
| Mlb/hr | Million Pounds per Hour |
| NCV | Non-Cited Violation |
| NRC | Nuclear Regulatory Commission |
| RG | Regulatory Guide |
| RHRSW | Residual Heat Removal Service Water |
| SDP | Significance Determination Process |
| SSC | Structure, System or Component |
| TS | Technical Specification |
| TSSR | Technical Specification Surveillance Requirement |
| UFSAR | Updated Final Safety Analysis Report |

Attachment 2

Confirmatory Measurements Comparison Criteria

The NRC applied the comparison criteria contained in NRC Inspection Procedure (IP) 84750, "Radioactive Waste Treatment, and Effluent and Environmental Monitoring," dated March 15, 1994, to determine if the licensee's measurement results were in statistical agreement with the NRC measurement results. For the purposes of this comparison, the NRC result is divided by its associated uncertainty to obtain the resolution. (Note: For purposes of this process, the uncertainty is defined as the relative standard deviation, one sigma, of the NRC's contract laboratory's analysis.) The licensee's result is then divided by the corresponding NRC result to obtain the ratio (licensee result/NRC). The licensee's measurement is in agreement if the value of the ratio fall within the limits shown in the following table for the corresponding resolution.

| **Resolution** | **Acceptance Range (Licensee Result/NRC Result)** |
|---|---|
| <4 | Technical Judgement[1] |
| 4-7 | 0.5-2.0 |
| 8-15 | 0.6-1.66 |
| 16-50 | 0.75-1.33 |
| 51-200 | 0.80-1.25 |
| >200 | 0.85-1.18 |

For analyses that are below the minimum detectable concentration (either for the licensee or NRC's contract laboratory), the measurements are determined to be in agreement if both are below the minimum detectable concentration or if one has an uncertainty that is within the minimum detectable concentration.

[1]The inspectors used technical judgement in reviewing results having a relative 1 sigma uncertainty greater than 25 percent (i.e., resolution less than 4). In these cases, the values were typically very close to the laboratory's detection capabilities, and greater variability was expected. Consequently, these sample comparisons were made based on the inspectors' qualitative review of the analytical results.

Attachment 3

## Tritium Sample Results
## Quad Cities Generating Station

| # | Collection Date | NRC | | | | Licensee | | | Ratio: Licensee to NRC | Result |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Sample ID | Tritium pCi/L ± uncertainty | | MDC | Sample ID | Tritium pCi/L ± uncertainty | | | |
| 1 | 04/21/2006 | Q-06-1-01 | < MDC | | 180 | *Q-36 | < MDC | | n/a | Agreement |
| 2 | 04/21/2006 | Q-06-1-02 | < MDC | | 180 | *Q-33 | < MDC | | n/a | Agreement |
| 3 | 04/21/2006 | Q-06-1-03 | < MDC | | 180 | *Q-35 | < MDC | | n/a | Agreement |
| 4 | 04/21/2006 | Q-06-1-04 | < MDC | | 180 | *Q-34 | < MDC | | n/a | Agreement |
| 5 | 05/31/2006 | Q-06-2-01 | < MDC | | 180 | MW-QC-107I | < MDC | | n/a | Agreement |
| 6 | 05/31/2006 | Q-06-2-02 | < MDC | | 180 | MW-QC-106I | < MDC | | n/a | Agreement |
| 7 | 05/31/2006 | Q-06-2-03 | < MDC | | 180 | FTW | < MDC | | n/a | Agreement |
| 8 | 05/31/2006 | Q-06-2-04 | 520 | 120 | 180 | MW-QC-001 | 550 | 143 | 1.06 | Agreement |
| 9 | 05/31/2006 | Q-06-2-05 | 440 | 120 | 180 | MW-QC-002 | 497 | 140 | 1.13 | Agreement |
| 10 | 05/31/2006 | Q-06-2-06 | < MDC | | 180 | MW-QC-FHW | < MDC | | n/a | Agreement |
| 11 | 05/31/2006 | Q-06-2-07 | < MDC | | 180 | MW-QC-107I | < MDC | | n/a | Agreement |
| 12 | 05/31/2006 | Q-06-2-08 | 400 | 120 | 180 | MW-QC-LFW | 371 | 134 | 0.93 | Agreement |
| 13 | 05/31/2006 | Q-06-2-09 | 1090 | 140 | 180 | MW-QC-108S | 1460 | 217 | 1.34 | Non-agreement |
| 14 | 05/31/2006 | Q-06-2-10 | 230 | 120 | 190 | MW-QC-106S | < MDC | | n/a | Agreement |
| 15 | 05/31/2006 | Q-06-2-11 | < MDC | | 190 | MW-QC-106I | < MDC | | n/a | Agreement |
| 16 | 05/31/2006 | Q-06-2-12 | 32780 | 620 | 180 | MW-QC-102I | 32600 | 977 | 0.99 | Agreement |
| 17 | 05/31/2006 | Q-06-2-13 | 10250 | 310 | 190 | MW-QC-102S | 9410 | 655 | 0.92 | Agreement |
| 18 | 06/01/2006 | Q-06-2-14 | < MDC | | 190 | MW-QC-103I | < MDC | | n/a | Agreement |
| 19 | 06/01/2006 | Q-06-2-15 | < MDC | | 190 | MW-QC-DCS | < MDC | | n/a | Agreement |
| 20 | 06/01/2006 | Q-06-2-16 | 670 | 130 | 180 | MW-QC-BFW | 740 | 152 | 1.10 | Agreement |
| 21 | 06/01/2006 | Q-06-2-17 | < MDC | | 190 | MW-QC-STP | < MDC | | n/a | Agreement |
| 22 | 06/01/2006 | Q-06-2-18 | < MDC | | 190 | MW-QC-Well#1 | < MDC | | n/a | Agreement |

## Tritium Sample Results
## Quad Cities Generating Station

| # | Collection Date | NRC | | | | Licensee | | | Ratio: Licensee to NRC | Result |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Sample ID | Tritium pCi/L ± uncertainty | | MDC | Sample ID | Tritium pCi/L ± uncertainty | | | |
| 23 | 06/01/2006 | Q-06-2-19 | < MDC | | 190 | MW-QC-MW-1 | < MDC | | n/a | Agreement |
| 24 | 06/01/2006 | Q-06-2-20 | < MDC | | 190 | MW-QC-Well#5 | < MDC | | n/a | Agreement |
| 25 | 06/01/2006 | Q-06-2-21 | < MDC | | 190 | MW-QC-2 | 250 | 126 | n/a | Agreement |
| 26 | 06/01/2006 | Q-06-2-22 | < MDC | | 180 | MW-QC-105I | < MDC | | n/a | Agreement |
| 27 | 06/01/2006 | Q-06-2-23 | < MDC | | 190 | MW-QC-104S | 262 | 130 | n/a | Agreement |
| 28 | 06/01/2006 | Q-06-2-24 | < MDC | | 190 | MW-QC-101I | < MDC | | n/a | Agreement |
| 29 | 06/01/2006 | Q-06-2-25 | < MDC | | 190 | MW-QC-101S | < MDC | | n/a | Agreement |
| 30 | 07/14/2006 | Q-06-3-01 | < MDC | | 180 | *Q-36 | < MDC | | n/a | Agreement |
| 31 | 07/14/2006 | Q-06-3-02 | < MDC | | 180 | *Q-33 | + | | n/a | n/a |
| 32 | 07/14/2006 | Q-06-3-03 | < MDC | | 180 | *Q-35 | < MDC | | n/a | Agreement |
| 33 | 07/14/2006 | Q-06-3-04 | < MDC | | 180 | *Q-34 | + | | n/a | n/a |

MDC - Minimum Detectable Concentration
* REMP Sample Locations
+ Licensee quarterly composites of weekly grab samples that had not been analyzed.

NRC sample uncertainties are based on two sigma counting statistics.

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Sincerely,

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

Docket Nos. 50-254; 50-265; 72-053
License Nos. DPR-29; DPR-30

Enclosure:
Inspection Report 05000254/2006006; 05000265/2006006
w/Attachments:
1. Supplemental Information
2. Confirmatory Measurements Comparison Criteria
3. Tritium Sample Results

cc w/encl:
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Plant Manager - Quad Cities Nuclear Power Station
Regulatory Assurance Manager - Quad Cities Nuclear Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
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