UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

May 9, 2014

Mr. Michael J. Pacilio
Senior VP, Exelon Generation Co., LLC
President and CNO, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000254/2014002;
05000265/2014002

Dear Mr. Pacilio:

On March 31, 2014, 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 results of this inspection, which were discussed on April 17, 2014, with Mr. S. Darin, and other members of your staff.

Based on the results of this inspection, one NRC-identified finding of very low safety significance was identified during this inspection. The finding did not involve enforcement action because no violation of a regulatory requirement was identified. Additionally, a licensee-identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of any Finding, 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.

Additionally, as we informed you in the most recent NRC integrated inspection report, cross-cutting aspects identified in the last six months of 2013 using the previous terminology were being converted in accordance with the cross-reference in Inspection Manual Chapter 0310. Section 4OA5 of the enclosed report documents the conversion of these cross-cutting aspects which will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the 2014 mid-cycle assessment review. If you disagree with the cross cutting aspect assigned, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Quad Cities Plant.

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

Sincerely,

/RA/

Christine Lipa, Chief
Branch 1
Division of Reactor Projects

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

Enclosure:
IR 05000254/2014002; 05000265/2014002
w/Attachment: Supplemental Information

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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:	05000254/2014002; 05000265/2014002
Licensee:	Exelon Generation Company, LLC
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	Cordova, IL
Dates:	January 1 through March 31, 2014
Inspectors:	R. Murray, Senior Resident Inspector B. Cushman, Resident Inspector J. Bozga, Reactor Inspector B. Jose, Senior Reactor Engineer C. Phillips, Project Engineer
Approved by:	Christine Lipa, Chief Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000254/2014002; 05000265/2014002; 01/01/2014 - 03/31/2014; Quad Cities Nuclear Power Station, Units 1 and 2; Outage Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding was identified by the inspectors. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas" dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

Green. The inspectors identified a finding of very low safety significance (Green) involving the licensee's failure to demonstrate compliance with American National Standards Institute (ANSI) N14.6-1978, Section 3.2.1.1. Specifically, the licensee did not establish the design stress factors based on the fracture toughness characteristics of the socket pins, lock pins, and hook pins for the steam dryer/steam separator lifting device. This issue was entered into the licensee's corrective action program (CAP) as Action Request (AR) 1517114, "Dryer/Separator Strongback Calculation Discrepancies," dated May 23, 2013, and AR 1578475, "Dryer/Separator Strongback Pin Inspection Criteria," dated October 30, 2013.

The inspectors determined the finding to be more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, compliance with ANSI N14.6-1978, Section 3.2.1.1 is to ensure safe load handling of heavy loads over the reactor core, spent fuel, and/or safety-related systems through establishing the design based on the fracture toughness characteristics of the material. The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I -- Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that none of the conditions constituting a loss of control were met as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRS and BWRS," for this finding and no Phase II or Phase III analysis was required. Specifically, the licensee provided information to inspectors that prior nondestructive examinations and inspections of the lifting device found no prior material defects. In addition, the licensee had not experienced any load drop events since placing the steam dryer/steam separator lifting device into service. The lifting device was also load tested successfully in accordance with the applicable requirements of ANSI N14.6. Therefore, the inspectors determined that this finding was of very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to a design calculation from 2005, and thus was not necessarily indicative of current licensee performance.

No violation of regulatory requirements is associated with this finding based on the steam dryer/steam separator lifting device being a non-safety-related structural component. (Section 1R20)

B. Licensee-Identified Violations

Violations of very low safety or security significance or Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. These violations and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1

Unit 1 operated at 100 percent thermal power with the exception of planned power reductions for routine surveillances, main condenser flow reversals, planned equipment repair, and control rod maneuvers from January 1, 2014, through March 31, 2014.

Unit 2

Unit 2 operated at 100 percent thermal power with the exception of planned power reductions for routine surveillances, main condenser flow reversals, planned equipment repair, and control rod maneuvers from January 1, 2014, through February 24, 2014. On February 25, 2014, Unit 2 entered the power coast down period as refueling outage Q2R22 approached. On March 31, 2014, at 4:16 p.m., Unit 2 commenced an unplanned shutdown to comply with Technical Specification Limiting Condition for Operation 3.4.4, "Reactor Coolant System Operational Leakage." During routine inspections of control rod drive hydraulic control unit (HCU) valves, the licensee identified a through body leak on HCU 18-27 valve 2-0305-101, which is the HCU insert isolation valve. Because this valve is unisolable from the reactor coolant system, the leak was considered pressure boundary leakage. Unit 2 entered Mode 3, and forced outage Q2F66, at 11:46 p.m. on March 31, 2014, and was in Mode 3 at the end of the evaluated period. The regulatory aspect of the pressure boundary leakage will be reviewed when the licensee issues the associated Licensee Event Report.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Extreme Cold Conditions

a. Inspection Scope

Since extreme cold conditions were forecast in the vicinity of the facility for January 6, 2014, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On January 6, 2014, the inspectors walked down the reactor building and turbine building ventilation systems, including the applicable heating steam portions and condensate return units. Inspections were conducted of insulation, heat trace circuits, space heater operation, and weatherized enclosures to ensure operability of affected systems. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel for safe plant operation and emergency response would be available. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 turbine building closed cooling water system during inspection of ‘B’ heat exchanger;
- ‘B’ standby gas treatment system with ‘A’ standby gas treatment system out of service for planned maintenance; and
- Unit 2 standby liquid control system.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. 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 and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 3.0, Services Building, Elevation 609’-0”, Cable Spreading Room;
- Fire Zone 8.2.7.C, Unit 1/2 Turbine Building, Elevation 611', Turbine Oil Reservoirs;
- Fire Zone 1.1.1.5, Unit 1 Reactor Building, Elevation 666'-6", Standby Liquid Control 4th Floor West;
- Fire Zone 5.0, Unit 2 Turbine Building, Elevation 595'-0", Safe Shutdown Pump Room ; and
- Fire Zone 9.1, Unit 1 Turbine Building, Elevation 595'-0", Diesel Generator.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event. The inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On February 3, 2014, the inspectors observed a crew of licensed operators in the plant’s simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On March 8, 2014, the inspectors observed the planned Unit 1 down power to isolate and repair the 1A feedwater regulating valve. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications (if applicable).

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- residual heat removal service water (RHRSW)

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Walkdown of Units 1 and 2 high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) in response to elevated risk from maximum emergency generation action grid condition;
- Implementation of Risk Management tools in response to loss of switchyard line 404;
- Work Week 14-12-01 Risk and Unit 1 250 Vdc battery charger out-of-service; and
- Work Week 14-13-02 Risk, including recovery of 1A reactor recirculation pump backup adjustable speed drive programmable logic computer, rebuild of 2B standby liquid control pump, and maintenance on 2D RHRSW system.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, 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 reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed during this inspection are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted four samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Issue Report (IR) 1605272: Contaminated Condensate Storage Tank Level Instrumentation Not Responding as Expected;
- IR 1608269: 2A Drywell Rad Monitor Spiked High Causing 902-55 A1 Alarm;
- IR 1611255: Unit 2 Emergency Diesel Generator Time Delay Relay Failure During Surveillance; and
- WO 1520212: MCC 19-2 Cubicle Inspection of RHRSW Pump 1D Room Cooling Fan.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted four samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- WO 1588069: Bus 24-1 "Degraded Voltage Relay Routine," following under voltage relay replacement;
- WO 1688243: Replace Unit 2 HPCI Room Cooler Tube Bundle;
- WO 1716641: Reactor Building Vent Post-Maintenance Testing, following planned reactor building ventilation maintenance;
- QCOS 1700-07: Unit 2 Reactor Building Ventilation and Fuel Pool Radiation Monitoring Calibration and Functional Test, following relay replacements; and
- WO 1676815: MCC 28/29-5 Auto-Transfer Logic Operability Surveillance.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted five post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 (Closed) Unresolved Item 05000254/2013003-05; 05000265/2013003-05: Steam Dryer/Steam Separator Lifting Device Fracture Toughness Properties

a. Inspection Scope

During the 2013 refueling and other outage activities inspections, the inspectors reviewed the licensee's Control of Heavy Loads Program in conjunction with the NRC's Operating Experience Smart Sample FY2007–03, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP 71111.20," Revision 2. The inspectors identified an unresolved item (URI 05000254/2013003-05; 05000265/2013003-05) involving the fracture toughness material properties of the hook pin, socket pin, and lock pin of the steam dryer/steam separator lifting device.

Prior to the start of the Unit 2 QR22 Outage, which was scheduled to begin on April 7, 2014, the inspectors reviewed information provided by the licensee related to the URI and identified one finding.

This was a partial inspection sample completed under this procedure. This inspection is continued into the next reporting period and the remainder of the inspection and documentation of the completed sample will be included in NRC Integrated Inspection Report 05000254/2014003; 05000265/2014003. This URI is closed.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) involving the licensee's failure to demonstrate compliance with American National Standards Institute (ANSI) N14.6-1978, Section 3.2.1.1. Specifically, the licensee did not establish the design stress factors based on fracture toughness of the socket pins, lock pins, and hook pins for the steam dryer/steam separator lifting device.

Description: The steam dryer/steam separator lifting device was a non-safety-related structure that was used to move the steam dryer and steam separator from inside to outside of the reactor pressure vessel during the refueling outage.

The safety evaluation report for the control of heavy loads Phase 1 at Quad Cities Units 1 and 2, dated June 27, 1983, classified the steam dryer/steam separator lifting device as a special lifting device and provided documentation how compliance with ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials," was met. American National Standards Institute N14.6-1978, Section 3.2.1.1 states in part, "…When materials that have yield strengths above 80 percent of their ultimate strength are used, each case requires special consideration, and the foregoing stress design factors do not apply. Design shall be on the basis of the material's fracture toughness, and the designer shall establish the criteria…"

The inspectors reviewed Calculation No. 487-002Ca, "Steam Dryer/Steam Separator Lifting Device Calculation," Revision 1, that was used to demonstrate compliance with ANSI N14.6. The inspectors identified in Calculation No. 487-002Ca that the socket pin, hook pin, and lock pin of the steam dryer/steam separator lifting device had yield strengths greater than 80 percent of their ultimate strengths. Also, the inspectors identified that Calculation No.487-002Ca did not contain an evaluation of these structural elements based on their fracture toughness properties and did not contain a brittle fracture analysis.

Subsequent to the 2013 NRC inspection, the licensee performed a brittle fracture analysis of the socket pin, hook pin, and lock pin of the steam dryer/steam separator lifting device in order to address the inspectors' concern. The analysis concluded that several new changes were needed to be implemented in order to ensure that the design of the socket pins, lock pins and hook pins was based on the fracture toughness characteristics of the pin material and that crack propagation through the socket pin, lock pin, and hook pin of the steam dryer/steam separator lifting device will not occur.

- Visual examination of the eight individual socket pins in accordance with the applicable American Society of Mechanical Engineers (ASME) Sections III and V requirements as delineated in ANSI N14.6.
- Magnetic Particle/Liquid Penetrant examination of the four individual lock pins in accordance with the applicable ASME Section III and V requirements as delineated in ANSI N14.6.
- Acceptance criteria for maximum allowable flaw size (depth and length) for the socket pin, lock pin, and hook pin.

The brittle fracture analysis was reviewed by regional specialists, and no performance deficiencies were identified with the analysis.

This issue and associated changes were entered into the licensee's CAP as the licensee initiated IR 1517114, "Dryer/Separator Strongback Calculation Discrepancies," dated May 23, 2013, and AR 1578475, "Dryer/Separator Strongback Pin Inspection Criteria," dated October 30, 2013.

<u>Analysis</u>: The inspectors determined the licensee's failure to comply with ANSI N14.6-1978, Section 3.2.1.1 for the socket pins, lock pins, and hook pins of the steam dryer/steam separator lifting device was a performance deficiency. In accordance with Inspection Manual Chapter (IMC) 0612, "Issue Screening", Appendix B, the inspectors determined the performance deficiency was more than minor, and a finding, because the performance deficiency was associated with the Initiating Events Cornerstone attribute of design control and adversely affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, compliance with ANSI N14.6-1978, Section 3.2.1.1 is to ensure safe load handling of heavy loads over the reactor core, over spent fuel, and/or over safety-related systems through establishing the design based on the fracture toughness characteristics of the pin

material. The lack of acceptance criteria for maximum allowable flaw size in applicable procedures would increase the likelihood of a load drop and would decrease the load handling reliability of the lifting device. Specifically, an identified indication on the socket pins, lock pins, or hook pins might not be properly evaluated and/or properly dispositioned before being returned to service. In addition, the lack of visual examinations of the eight individual socket pins and the lack of magnetic particle/liquid penetrant examinations of the four individual lock pins would increase the likelihood of a load drop and would decrease the load handling reliability of the lifting device because an existing indication may not be identified on either the socket pins or lock pins.

The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I -- Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that none of the conditions constituting a loss of control were met as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRS and BWRS," for this finding and no Phase II or Phase III analysis was required. Specifically, the licensee provided information to inspectors that prior nondestructive examinations and inspections of the lifting device found no prior material defects. In addition, the licensee had not experienced any load drop events since placing the steam dryer/steam separator lifting device into service. The lifting device was also load tested successfully in accordance with the applicable requirements of ANSI N14.6. Therefore, the inspectors determined that this finding was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to a design calculation from 2005, and thus was not necessarily indicative of current licensee performance.

Enforcement: No violation of regulatory requirements is associated with this finding based on the steam dryer/steam separator lifting device being a non-safety related structural component **(FIN 05000254/2014002-01; 05000265/2014002-01, "Steam Dryer/Steam Separator Lifting Device Fracture Toughness Properties")**.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- QCOS 1000-04: Residual Heat Removal Service Water Pump Operability Test (IST);
- QCOS 1600-07: Reactor Coolant Leakage in the Drywell (RCS);
- QCOS 2300-27: HPCI Pump Comprehensive/ Performance Test (IST);
- QCOS 1400-01: Core Spray System Flow Rate Test (Routine);

- QCOS 6600-42: Unit 2 Emergency Diesel Generator Load Test (IST); and
- QCOS 1400-10: Core Spray Operability Verification (Routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted two routine surveillance testing samples, three inservice testing samples, and one reactor coolant system leak detection inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. <u>Findings</u>

No findings were identified.

.4 OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI) for Quad Cities Units 1 and 2 for the period from the first quarter 2013 through the fourth quarter 2013. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of January 2013 through December 2013 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted two unplanned scrams per 7000 critical hours samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for Quad Cities Units 1 and 2 for the period from the first quarter 2013 through the fourth quarter 2013. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of January 2013 through December 2013 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted two unplanned scrams with complications samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for Quad Cities Units 1 and 2 for the period from the first quarter 2013 through the fourth quarter 2013. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports and NRC integrated inspection reports for the period of January 2013 through December 2013 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted two unplanned transients per 7000 critical hours samples as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

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

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents in relation to ASME Class III service water leaks to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of September 2013 through March 2014, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semi-annual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds (OWAs) on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP, and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

This review constituted one operator workaround annual inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.5 Selected Issue Follow-Up Inspection: Engineering Change 393738, "Structural Evaluation of U1 RB EL 623 Ceiling Concrete", Revision 1 and Engineering Change 394030, "Structural Evaluation of U2 RB Concrete Due to Leak from Skimmer Surge Tank Duct", Revision 0

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting water intrusion into the spent fuel pool structure. The licensee performed engineering evaluations to assess the spent fuel pool structural integrity for Units 1 and 2. The evaluations determined the spent fuel structure demonstrated compliance with the design basis requirements. The inspectors reviewed the engineering evaluations for potential impacts on design basis requirements.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.6 Selected Issue Follow-Up Inspection; Issue Report 1604150: Chimney SPING [System Particulate Iodine and Noble Gases] Erratic Indication

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting erratic indication of the chimney SPING (the offgas chimney stack radiation monitor). The licensee declared the mid and high range channels of the main chimney noble gas monitors inoperable on January 6, 2014, and attributed the equipment issue to extreme cold weather (freezing flow lines). The licensee was able to restore the system to an operable status on January 10, 2014. The inspectors reviewed the licensee's emergency plan and procedures, and evaluated the effect of the loss of the SPING monitors.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Retraction of Event Notification 49727: 2A Drywell Monitor Inoperable

a. Inspection Scope

The inspectors reviewed the plant's event notification retraction, dated February 7, 2014, for Event Notification 49727 (initial notification made on January 15, 2014, after the Unit 2A drywell radiation monitor was declared inoperable), to ensure the licensee met the requirements of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Discussed) Unresolved Item 05000254/2012010-01; 05000265/2012010-01: Concern with Meeting One Time Visual Inspections ASME Section XI Requirements

During the Phase I inspection of Commitment Items (NRC IR 05000265/2012008, ML12129A226), through direct observation, review of the work orders used to perform these examinations and interviews with the appropriate plant personnel, the inspectors identified the licensee was not performing visual examinations in accordance with the

ASME Code. The commitment to perform visual examinations in accordance with the ASME Section XI code applies to the following programs:

- B.1.23 One Time Inspection – Ventilation Systems, Compressed Gas Systems, Standby Liquid Control Chemistry Program;
- B.1.24 Selective Leaching;
- B.2.8 Periodic Inspection of Plant Heating Steam; and
- B.2.9 Periodic Inspection of Components Subject to Moist Air Environments.

The Commitment Item specifically states, "The inspection will be performed in accordance with ASME Code requirements. Certified non-destructive examination (NDE) examiners will conduct a VT-3 visual inspection (VT-1 for the Selective Leaching Program, these inspections will consist of visual inspection consistent with ASME Section XI VT-1 visual inspection requirements.)."

For visual examinations performed in accordance with the code, ASME Section XI IWA 2210 requires visual examination be performed in accordance with ASME, Section V, Article 9. This section requires a procedure be used when performing visual examinations that include instructions on how the visual inspection is to be performed as well as illumination and instruments to be used. The ASME Section XI has specific requirements associated with the different levels of visual examinations (VT-1 or VT-3) including distance, illumination, and character card resolution requirements. As discussed in NRC Inspection Report 05000254/2012010; 05000265/2012010, the inspectors identified the licensee did not perform the VT-1 or VT-3 examinations in accordance with written procedures which documented the requirements for lighting, distance, and other key parameters as well as acceptance criteria.

The licensee disagreed with the inspectors, indicating the intent of the Commitment was to have a qualified person performing the inspection, not to perform the examination in accordance with the requirements of the code. The licensee also indicated ASME requirements were verbally communicated to the examiners during their pre-job brief, therefore ensuring compliance with the code. In a letter dated May 18, 2012, in (ML12173A423) the licensee requested a Commitment Item change with the purpose of clarifying the original commitment. The inspectors opened an unresolved item to solicit additional support from staff in the Division of License Renewal in the Office of Nuclear Reactor Regulation to further understand the commitment.

The staff concluded the original license renewal Safety Evaluation Report NUREG 1796 (ML042050507) related to the adequacy of the above programs was based on statements in the License Renewal Application and Request for Additional Information (RAI) responses that VT-1 and VT-3 examinations would be performed for inspections of Systems, Structures, and Components (SSCs) being managed by the above programs. By citing VT-1 and VT-3 visual examination techniques, the Aging Management Program included inspection requirements essential to an effective visual examination regardless of whether the SSC is ASME Class 1, 2, or 3.

However, the staff acknowledged that SSCs that are not within the scope of ASME Code Section XI can be effectively age-managed without using ASME Code Section XI inspections as long as critical aspects of inspections (e.g., purpose of the inspection, acceptance criteria, exam distance, illumination, examination coverage) are controlled by formal instructions. The staff subsequently reviewed the licensee's letter and asked several questions in a Request for Additional Information (ML12291A831). In a letter, "Quad Cities Nuclear Power Station, Units 1 and 2: Changes to Commitment for License Renewal (ML13304A524)," the staff concluded that the additional programmatic controls described by the licensee in their RAI responses for **future** visual examinations conducted outside the formal jurisdiction of the ASME Code Section XI requirements would be sufficient to demonstrate the effects of aging will be adequately managed. The staff further concluded the program as described in their correspondences was consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21 (a)(3).

With respect to the licensee's inspections conducted prior to the period of extended operation and prior to the implementation of additional programmatic controls posed by the licensee in their response to the RAI, the staff concluded the inspectors' concerns with the quality of those inspections remained valid. Because these steps were not in place and with the limited level of detail provided in the work order instructions and the pre-job brief observed by the inspectors, the staff and inspectors could not conclude that aging of in-scope SSCs would be adequately managed. Specifically:

- Critical inspection parameters were not consistently documented in work instructions or pre-job brief forms. Specifically, during some of the inspections observed, the inspectors identified inconsistent compliance with appropriate lighting levels, angle of observation, and distance to the SSC; and
- Based on the observation of regional inspection personnel, the pre-job briefs did not consistently cover maximum direct examination distance, illumination, or use of white-light meter and/or use of test card.

The staff further concluded the licensee did not comply with the inspection details contained in its UFSAR for the Selective Leaching of Materials, Periodic Inspection of Plant Heating Steam, and Periodic Inspection of Components Subject to Moist Air Environments Programs. The licensee did not fulfill the intent of its regulatory commitments associated with VT-1 and VT-3 inspections for the One-Time Inspection, Selective Leaching of Materials, Periodic Inspection of Plant Heating Steam, and Periodic Inspection of Components Subject to Moist Air Environments Programs. The staff concluded the knowledge-based controls (verses procedure controls) to conduct the past inspections did not appear to be sufficient to ensure that potential aging effects could be consistently identified.

This unresolved item will remain open pending review of the licensee's actions in response to the information provided above, including conducting additional inspections or providing justification on the adequacy of previously conducted inspections.

.2 Institute of Nuclear Power Operations Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the Institute of Nuclear Power Operations plant assessment conducted in February 2013. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings were identified.

.3 Cross-Cutting Aspects Cross-Reference

The table below provides a cross-reference from the third and fourth quarter 2013 findings and associated cross-cutting aspects to the new cross-cutting aspects resulting from the common language initiative. These aspects and any others identified since January 2014, will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the 2014 mid-cycle assessment review.

Finding	Old Cross-Cutting Aspect	New Cross-Cutting Aspect
05000265/2013004-1	H.3.(b)	H.5

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 17, 2014, the inspectors presented the inspection results to the Mr. S. Darin, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- On March 6, 2014, the inspectors presented the results of the IP 71111.20, outage activities inspection to Mr. K. O'Shea, Plant Manager, and other members of the licensee's staff. The licensee personnel acknowledged the inspection results presented. The inspectors asked the licensee whether any materials examined during the inspection are considered proprietary. It was agreed that all paper copies of these proprietary documents would be shredded, and all electronic files of these proprietary documents would be deleted.

- An interim exit meeting for the discussion on URI 05000254/2012010-01; 05000265/2012010-01 was conducted on April 15, 2014, via telephone to discuss the subject update with Mr. W. Beck, Regulatory Assurance Manager, and other members of the licensee's staff.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) or Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

On February 5, 2014, Mechanical Maintenance Department was performing a bi-annual Fire Protection Walkdown, and identified that the fire protection coating was missing on an I-beam that was a recently installed pipe support in the turbine building. This pipe support was installed by a licensee contractor as part of an engineering change to install above ground piping for safety-related service water. The licensee's investigation discovered that the work instructions did not provide steps to replace the coating or to initiate the fire impairment in accordance with the licensee's work planning process. Technical Specification 5.4.1.c. states that written procedures shall be established, implemented, and maintained covering the Fire Protection Program. The licensee established procedure QCAP 1500-01, Revision 32; "Administrative Requirements for Fire Protection," as their implementing procedure for the Fire Protection Program. Step D.7.a.(1)(e) states, in part, that fire barriers (structural steel fire coating) "protecting safety related or safe shutdown areas **SHALL** be intact when in Mode 1, 2, & 3." Additionally, Step D.7.a.(3) states, "**IF** fire barrier inoperable, **THEN** compensatory and reporting requirements **SHALL** be followed per steps D.7.c and D.7.d." Contrary to the above, the licensee failed to have an intact fire barrier while in Mode 1 without implementing any compensatory measures or reporting requirements. This was a violation of Technical Specification 5.4.1.c. This finding is more than minor because it is associated with the Reactor Safety Mitigation Systems Cornerstone attribute of protection against external factors (fire) and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was determined to be of very low safety significance (Green) because the reactor would be able to reach and maintain a safe shutdown condition. The licensee documented this issue in their corrective action program in IR 1617549, "Fire Protection Coating Missing on Steel I-Beam."

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Darin, Site Vice President
W. Beck, Regulatory Assurance Manager
D. Collins, Radiation Protection Manager
M. DeVault, Training Director
H. Dodd, Site Maintenance Director
D. Kimler, Operations Director
D. Luebbe, Work Control Manager
K. Ohr, Site Engineering Director
T. Petersen, Regulatory Assurance Lead
S. Piepenbrink, Security Manager
T. Wojick, NOS Manager
J. Wooldridge, Chemistry Manager

Nuclear Regulatory Commission

A. Boland, Director, Division of Reactor Projects
C. Lipa, Chief, Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000254/2014002-01; 05000265/2014002-01	FIN	Steam Dryer/Steam Separator Lifting Device Failure to Meet American National Standards Institute (ANSI) N14.6 (Section 1R20.2)

Closed

05000254/2014002-01; 05000265/2014002-01	FIN	Steam Dryer/Steam Separator Lifting Device Failure to Meet American National Standards Institute (ANSI) N14.6 (Section 1R20.2)
05000254/2013003-05; 05000265/2013003-05	URI	Steam Dryer/Steam Separator Lifting Device Fracture Toughness Properties (Section 1R20.1)

Discussed

05000254/2012010-01; 05000265/2012010-01	URI	Concern with Meeting One Time Visual Inspections ASME Section XI Requirements (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- QCOP 0010-02: Required Cold Weather Routines; Revision 42
- IR 1605182: Grid Capacity Lessons Learned
- IR 1605060: NRC: Tele-Tower Scaffold Presents a Seismic Risk to ½ EDG
- IR 1604238: U1 East TB Heating Coils Have Leak
- IR 1605159: U2 Reactor Building Heating Pump Trap Leak By

1R04 Equipment Alignment

- QCOP 3800-02: Unit 1 TBCCW System Operation; Revision TIC 32021
- QCOP 7500-01: Standby Gas Treatment System (SBGTS) Standby Operation and Start-Up; Revision 19
- M-44: Diagram of Standby Gas Treatment; Revision AP
- EC 393548, “Review Seismic Impact of the Unit 1 & 2 SBLC [Standby Liquid Control] Pump Stuffing Box Covers Installed Without Studs and Hold-Down Nuts In-Place,” Revision 0
- IR 1486115, “Install Studs & Hold Down Nuts – 1A SBLC Pump Stuffing Box”
- IR 1621133, “Operator Timed Response and SBLC Injection,”
- M-40, “Diagram of Standby Liquid Control Piping,” Revision AY

1R05 Fire Protection

- Pre-Fire Plan: FZ 3.0; SB 609’-0” Elevation Cable Spreading Room
- Pre-Fire Plan: FZ 8.2.7.C; Unit 1/2 TB 611' Elevation Turbine Oil Reservoirs
- Quad Cities Generating Station Pre-Fire Plan, Section FZ 1.1.1.5; Unit 1 RB 666’-6” Elevation Stand-By Liquid Control 4th Floor West, February 2013
- Quad Cities Generating Station Pre-Fire Plan, Section FZ 5.0; Unit 2 TB 595’-0” Elevation Safe Shutdown Pump Room, October 2013
- Quad Cities Generating Station Pre-Fire Plan, Section FZ 9.1; Unit 1 TB 595’-0” Elevation Diesel Generator, March 2012
- Quad Cities Fire Hazard Analysis, Section Fire Zone 1.1.1.5; Revision 20
- Quad Cities Fire Hazard Analysis, Section Fire Zone 5.0; Revision 20
- Quad Cities Fire Hazard Analysis, Section Fire Zone 9.1; Revision 20
- IR 1632299: Fire Extinguisher Inspection Tag Has Incorrect Date

1R12 Maintenance Effectiveness

- ER-AA-310-1004; Maintenance Rule – Performance Monitoring; Revision 11
- ER-AA-310-1005; Maintenance Rule – Dispositioning Between (a)(1) and (a)(2); Revision 6
- ER-AA-310; Implementation of the Maintenance Rule; Revision 8
- IR 1638379; 2D RHRSW Room Cooler Header UT Readings Below Min Wall

1R13 Maintenance Risk Assessments and Emergent Work Control

- OP-AA-108-107-1001: Station Response to Grid Capacity Condition; Revision 4
- WC-AA-101: Online Work Control; Revision 20
- IR 1604689: 1A Core Spray Subdoor is Degraded
- System Planning Operating Guide (SPOG) 1-3-C; Station 4, Quad Cities Unit 1 and 2 Special Protection System; Revision 9
- WO 1708992: Recover 1A ASD PLC 'A'

1R15 Operability Determinations and Functionality Assessments

- IR 1605272: 'B' CCST Level Not Responding As Expected
- IR 1608269: 2A DW Rad Monitor Spiked High Causing 902-55 A1 Alarm
- QCOS 6600-17: Unit 2 Diesel Generator Fails to Start; Revision 2
- IR 1611255: U2 EDG TD2 Relay Failure During Surveillance
- IR 1638763: U1 EDG Start Failure Relay Replacement - EOC for U2 SF Relay
- IR 1638768: U0 EDG Start Failure Relay Replacement - EOC for U2 SF Relay
- IR 1648259: NRC SRI Questions on Start Failure Relay Qualified Life
- WO 1520212: MCC 19-2 CUB F4 1-5745D RHRSWP 1D ROOM COOL FAN C
- EC 334115: Evaluate the Effect of the Installation of a Dehumidifier in the RHRSW Vaults
- Calculation QDC-5700-H-0695: RHR and RHRSW Pump Room Cooling Following Failure of Lock and Dam 14
- Calculation VT-16: ECCS and RHRSW Room Cooling

1R19 Post Maintenance Testing

- WO 1588069: Bus 24-1 Degraded Voltage Relay Routine
- QCOS 6500-10: Functional Test of Unit 2 Second Level Under Voltage; Revision 30
- WO 1688243: Replace Unit 2 HPCI Room Cooler Tube Bundle
- QCOS 5750-09: ECCS Room and DGCWP Cubicle Cooler Monthly Surveillance; Revision 35
- IR 1617216: Unable to Backflush FI 2-3491-49
- WO 1716641: Reactor Building Vent PMTs
- QCIS 1700-07: Reactor Building Ventilation and Fuel Pool Radiation Monitoring Calibration and Functional Test; Revision 21
- WO 1676815: MCC 28/29-5 Auto-Transfer Logic Operability Surveillance
- QCOS 6700-02: MCC 28/29-5 Auto-Transfer Logic Operability Surveillance; Revision 16
- QCOS 5750-10: Reactor Building Ventilation Isolation Dampers Pneumatic Accumulator System Pressure Decay and Fail Safe Test; Revision 20
- QCOS 5750-12: Power Operated Automatic SCIV(s) Isolation Time Test; Revision 3

1R20 Refueling and Other Outage Activities

- Crane and Heavy Lift Inspection (OpESS FY2007–03)
- ANSI N14.6-1978: Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials; 1978
- IR 1517114: Dryer/Separator Strongback Calculation Discrepancies; dated May 23, 2013
- IR 1578475: Dryer/Separator Strongback Pin Inspection Criteria; dated October 30, 2013
- Analysis No. 0000-0166-2254: Fracture Toughness Assessment for Steam Dryer-Separator Strongback 124D1216G001; Revision 1
- Procedure No. QCMPM 5800-31: Lifting Rig Pin Surveillance and Non-Destructive Examination; Revision 11

1R22 Surveillance Testing

- WO 1685846: RHR Service Water Pump ‘B’ Flow (IST)
- QCOS 1000-04 TIC 3182: RHR Service Water Pump Operability Test; Revision 55
- QCOS 2300-27: HPCI Pump Comprehensive/Performance Test; Revision 33
- IR 1620225: Procedure Deficiency for QCOS 2300-27
- QCOS 1600-07: Reactor Coolant Leakage in the Drywell
- QCOS 1400-01: Core Spray System Flow Rate Test; Revision 44
- QCOS 6600-42: U2 Emergency Diesel Generator Load Test; Revision 43
- QCOS 1400-10: Core Spray Operability Verification; Revision 23

4OA1 Performance Indicator Verification

- Nuclear Energy Institute (NEI) Document 99-02: Regulatory Assessment Performance Indicator Guideline, Revision 7
- Licensee 2013 Performance Indicator Submittals

4OA2 Problem Identification and Resolution

- EP-AA-1006: Radiological Emergency Plan Annex for Quad Cities Station; Revision 35
- IR 1604150: Chimney SPING Erratic Indication
- EP-AA-121: Emergency Response Facilities and Equipment Readiness; Revision 12
- EP-AA-110-200: Dose Assessment; Revision 6
- NUREG/CR-7111: “A Summary of Aging Effects and Their Management in Reactor Spent Fuel Pools, Refueling Cavities, Tori, and Safety-Related Concrete Structures”, dated January 2012
- American Concrete Institute (ACI) 349.3R-02: “Evaluation of Existing Nuclear Safety-Related Concrete Structures”, 2010
- Engineering Change (EC) 393738: “Structural Evaluation of U1 RB EL 623 Ceiling Concrete”, Revision 1
- Engineering Change (EC) 394030: “Structural Evaluation of U2 RB Concrete Due to Leak from Skimmer Surge Tank Duct”, Revision 0
- Condition Report 1479735: “Leakage from Unit 2 Reactor Building 2nd Floor Ceiling”, dated February 25, 2013
- Condition Report 1417450: “Reactor Building Concrete Spall”, dated September 24, 2012
- Condition Report 1404147: “Fuk: Seismic, Concrete Spall RB 623”, dated August 23, 2012
- ER-AA-5400-1001: Raw Water Corrosion Program Guide; Revision 6
- Summary of BOP-SW PRA-Based Safety Consequences at Quad Cities Station

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ANSI	American National Standards Institute
AR	Action Request
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
HPCI	High Pressure Coolant Injection
HCU	Hydraulic Control Unit
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OWA	Operator Workaround
PARS	Publicly Available Records System
PI	Performance Indicator
RAI	Request for Additional Information
RCIC	Reactor Core Isolation Cooling
RHRSW	Residual Heat Removal Service Water
SPING	System Particulate Iodine and Noble Gases
SSC	Systems, Structures, and Components
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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

Sincerely,

/RA/

Christine Lipa, Chief
Branch 1
Division of Reactor Projects

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

Enclosure:
IR 05000254/2014002; 05000265/2014002
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