April 29, 2005

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 REPORT 05000254/2005002;
05000265/2005002

Dear Mr. Crane:

On March 31, 2005, 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 April 5, 2005, 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 two issues of very low safety significance (Green). One of these issues was determined to involve a violation of NRC requirements. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective program, the NRC is treating this finding and issue as a Non-Cited Violation 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,

/RA/

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

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

Enclosure: Inspection Report 05000254/2005002; 05000265/2005002
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Quad Cities Nuclear Power Station
Plant Manager - Quad Cities Nuclear Power Station
Regulatory Assurance Manager - Quad Cities Nuclear Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
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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:	05000254/2005002; 05000265/2005002
Licensee:	Exelon Nuclear
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	22712 206th Avenue North Cordova, IL 61242
Dates:	January 1 through March 31, 2005
Inspectors:	K. Stoedter, Senior Resident Inspector M. Kurth, Resident Inspector D. Chyu, Reactor Engineer B. Jose, Reactor Inspector P. Lougheed, Senior Engineering Inspector C. Phillips, Acting Senior Resident Inspector - Dresden T. Ploski, Senior Emergency Preparedness Analyst L. Ramadan, Nuclear Safety Professional G. Wilson, Senior Resident Inspector - Duane Arnold R. Ganser, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000254/2005002, 05000265/2005002; 01/01/2005-03/31/2005; Quad Cities Nuclear Power Station, Units 1 & 2; Maintenance Risk Assessment and Emergent Work, Problem Identification and Resolution, and Other.

This report covers a 3-month period of baseline resident inspection and a regional inspection on emergency preparedness. The inspection was conducted by Region III inspectors and the resident inspectors. Two Green findings and one Non-Cited Violation (NCV) 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. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance due to the licensee's failure to perform operability determinations/evaluations for non-safety related structures, systems, or components discussed in the Updated Final Safety Analysis Report which were discovered to be degraded.

  This finding was more than minor because if left uncorrected, the failure to properly evaluate the continued operability of degraded equipment could result in the licensee inappropriately relying on structures, systems, or components that were unable to perform their safety function during an initiating event. The finding also impacted the cross-cutting area of problem identification and resolution because the licensee has had multiple examples of failures to initiate operability determinations or evaluations which had not been previously identified. No violation of NRC requirements occurred since the completion of operability determinations/evaluations was not required by NRC regulations. (Section 4OA2.2).

- Green. The inspectors identified a NCV of 10 CFR 50.65(a)(4). Specifically, the NRC identified that the licensee non-conservatively evaluated the on-line risk associated with actions taken in response to an emergent residual heat removal service water leak on January 14, 2003.

  The inspectors considered this issue of more than minor significance because, had an adequate risk evaluation occurred, the on-line risk would have changed from Green to Yellow. The inspectors determined that the issue was of very low safety significance, or Green, because although one train of residual heat removal service water was unavailable, the actual safety function of the system could have been performed by the remaining train and the train was not inoperable for greater than the Technical

Specification allowed outage time. Corrective actions for this issue included providing training to operations personnel which focused on crediting manual operator actions in place of automatic actions as part of a risk assessment. (Section 4OA5).

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 approximately 85 percent power during the inspection period with the exception of planned power reductions on February 6 and 27, 2005, to perform control rod adjustments. On March 21, 2005, operations personnel shut down Unit 1 to begin refueling outage Q1R18. Refueling outage activities included replacement of the main power transformer, one switchyard breaker, a reactor recirculation pump motor and the low pressure turbine buckets, maintenance on multiple risk significant systems, and various other activities. Unit 1 remained shut down at the conclusion of the inspection period.

Unit 2 also operated at approximately 85 percent power during the period with the exception of planned power reductions for control rod special maneuvers and scram time testing on January 9, 2005, and turbine valve testing on March 27, 2005.

1. REACTOR SAFETY

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

1R01 Adverse Weather (71111.01)

a. Inspection Scope

On March 30, 2005, Quad Cities Station experienced severe thunderstorms and sustained winds greater than 40 miles per hour. Prior to the severe weather's arrival, the inspectors observed activities in the control room to determine the preparations being taken to address the approaching storm and the potential impact on equipment. The inspectors noted that the operations field supervisor had performed a tour of the outside areas to identify and address potential missiles. In addition, operations personnel in the control room were routinely monitoring weather radar and wind speed information. During discussions with the shift manager and the unit supervisors, the inspectors learned that both units had entered an increased risk condition due to the expected weather. In addition, operations personnel discussed the equipment available to each unit in the event a loss of offsite power occurred during the storm. This was extremely important as the amount of electrical equipment available on Unit 1 was limited due to refueling outage activities. The licensee also discussed the need to stop activities on the refueling floor if sustained winds of greater than 40 miles per hour were observed. This represented the completion of one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of the following risk-significant mitigating systems equipment during times when the equipment was of increased importance due to redundant systems or other equipment being unavailable:

- Unit 1 Reactor Core Isolation Cooling System;
- Safe Shutdown Makeup Pump and the Unit 2 Reactor Core Isolation Cooling System;
- Residual Heat Removal Service Water Loop 1A; and
- Unit 1 A Core Spray System.

The inspectors utilized the valve and breaker checklists listed at the end of this report to verify that the components were properly positioned and that support systems were lined up as needed. The inspectors examined the material condition of the components and observed equipment operating parameters to verify that there were no obvious deficiencies. The inspectors reviewed outstanding work orders and issue reports associated with each system to verify that those documents did not reveal issues that could affect the equipment inspected. The inspectors also used the information in the appropriate sections of the Updated Final Safety Analysis Report to determine the functional requirements of the systems. This review constituted the completion of four inspection samples.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

During the inspection period, the inspectors conducted an in-depth review and walkdown of the reactor protection system. This system was selected due to its high safety significance and risk significance. The inspection consisted of the following activities:

- a review of plant procedures (including selected abnormal and emergency procedures), drawings, the system health report, Technical Specifications, and the Updated Final Safety Analysis Report to determine overall system health, proper system configuration, and the system’s licensing basis;
- a review of outstanding maintenance work requests to determine items in need of repair;

- a review of system predefines to determine if preventive maintenance was completed as recommended;
- a review of predefine deferrals to evaluate the licensee's justification for not conducting recommended preventive maintenance tasks;
- a review of outstanding or completed temporary and permanent modifications to the system; and
- an electrical and/or mechanical walkdown of the system to verify proper alignment, component accessibility, availability, and condition.

The inspectors also reviewed selected issues documented in issue reports to verify that the issues were appropriately addressed. This review constituted the completion of one semi-annual walkdown sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Walkdowns

a. Inspection Scope

The inspectors performed routine walkdowns of accessible portions of the following risk significance fire zones:

- Fire Zone 1.1.1.2 - Unit 1 Reactor Building First Floor;
- Fire Zone 8.2.6.C - Unit ½ Ground Floor;
- Fire Zone 11.1.3 - Unit 1 High Pressure Coolant Injection and Tunnel;
- Fire Zone 11.1.4 - Unit 2 High Pressure Coolant Injection; and
- Various Fire Zones - Station Blackout Diesel Generator Building.

The inspectors verified that transient combustibles were controlled in accordance with the licensee's procedures. During a walkdown of each fire zone, the inspectors observed the physical condition of fire suppression devices and passive fire protection equipment such as fire doors, barriers, and penetration seals. The inspectors observed the condition and placement of fire extinguishers and hoses against the Pre-Fire Plan fire zone maps. The physical condition of accessible passive fire protection features such as fire doors, fire dampers, fire barriers, fire zone penetration seals, and fire retardant structural steel coatings were also inspected to verify proper installation and physical condition. Lastly, the inspectors reviewed the licensee's corrective action program database to ensure that fire protection-related issues were being entered into the program for resolution. This review constituted the completion of five inspection samples.

b. Findings

No findings of significance were identified.

.2 Fire Drill Observation

a. Inspection Scope

On February 8, 2005, the inspectors observed the licensee's fire brigade participate in a quarterly fire drill. The drill scenario consisted of a fire in the Unit 2A reactor feedwater pump auxiliary oil pump skid. Upon hearing the fire alarm, the inspectors observed the fire brigade members don their protective equipment to ensure that the brigade members were appropriately protected from the fire. The inspectors also observed the actions performed by and communications provided by the fire brigade leader to ensure that the leader demonstrated adequate command and control responsibilities, selected an appropriate staging area, performed a proper size up of the fire, selected the proper fire attack strategies, addressed potential adverse impacts on the plant, recognized the need for offsite assistance by local fire departments, and communicated with the control room. Lastly, the inspectors observed the fire brigade members during the fire attack to evaluate the appropriateness of their actions. This inspection represented the completion of one annual fire drill inspection sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On January 5, 2005, the inspectors observed engineering and operations personnel complete performance testing on the 1B residual heat removal heat exchanger. This heat exchanger was chosen for inspection due to its high safety significance and risk significance. During the testing observation the inspectors verified that the acceptance criteria and test results considered differences between test and design basis conditions because testing at the design heat removal rate was not practical. The inspectors performed independent calculations using the licensee's test results to confirm that the results considered possible uncertainties and that the heat exchanger remained capable of performing its safety function.

During the Unit 1 refueling outage, the inspectors performed a visual inspection of the Unit 1 emergency diesel generator heat exchanger internals. Prior to performing the inspection, the inspectors reviewed the licensee's heat exchanger inspection procedures to ensure that all areas of the heat exchanger were addressed. The inspectors verified that the licensee's procedures contained appropriate acceptance criteria. Very little corrosion was identified during the visual inspections. The inspectors also reviewed heat sink-related issue reports generated within the last year to ensure that the issues were being entered into the corrective action program with the appropriate characterization and significance. While inspecting the re-installation of the heat exchanger end bells, the inspectors identified a thread engagement issue on the "B" heat exchanger. The licensee documented this issue in Issue Reports 319103 and 319205. This review constituted the completion of two annual inspection samples.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On February 17, 2005, the inspectors observed two operations crews in the simulator. Each crew's performance was evaluated using a different scenario. The first scenario consisted of a reactor vessel level instrument failure, the loss of the "A" reactor protection system motor generator set, a control rod scram and drift, a fuel failure, a Group I isolation failure, a leak outside containment, and the need to manually blow down the reactor. The second scenario involved an average power range monitor failure, a reactor protection system channel failure, a control rod scram and drift, fuel failure, a turbine building steam leak, and the need to manually blow down the reactor.

The inspectors evaluated each crew's 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 the above areas was compared to licensee management expectations and guidelines as presented in the following documents:

- OP-AA-101-111, "Rules 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 each crew 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 or individuals requiring remediation and appropriately indicated when removal from shift activities was warranted. The inspectors observed the licensee's critique to verify that weaknesses identified during this observation were noted by the evaluators and discussed with the respective crews. This review constituted 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 (10 CFR 50.65) to evaluate maintenance effectiveness for the systems listed below. These systems were selected based on them being designated as risk significant under the Maintenance Rule, being in increased monitoring (Maintenance Rule category a(1) group), or due to an inspector identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Residual Heat Removal Service Water System;
- Automatic Depressurization System; and
- Turbine Building Closed Cooling Water System.

The inspectors review included an examination of specific system issues, an evaluation of maintenance rule performance criteria, maintenance work practices, common cause issues, 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 constituted the completion of three inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the documents listed in the "List of Documents Reviewed" section of this report to determine if the risk associated with the listed activities agreed with the results provided by the licensee's risk assessment tool. The inspectors conducted walkdowns to ensure that redundant mitigating systems credited by the licensee's risk assessment remained available. When compensatory actions were required, the inspectors conducted plant tours to validate that the compensatory actions were implemented. The inspectors discussed emergent work activities with the shift manager and work week manager to ensure that these additional activities did not change the risk assessment results. Lastly, the inspectors performed a word search review of the licensee's corrective action database to ensure that problems related to risk assessments were entered into the licensee's corrective action program. This review represented the completion of ten inspection samples.

- Work Week January 3-8, 2005, including planned maintenance on the control room emergency ventilation system, the switchyard, and the circulating water bays;
- Work Week January 10-16, 2005, including planned maintenance on the "B" fire diesel system and testing on the automatic depressurization system logic;
- Work Week January 17-23, 2005, including planned maintenance on the Unit 1 high pressure coolant injection system and the Unit 1 condensate demineralization system;
- Work Week January 24-29, 2005, including planned maintenance on the 1C reactor feedwater pump, the 1B electrohydraulic control system, the Unit 1 reactor core isolation cooling system, and the Unit 1 high pressure coolant injection system;
- An evaluation of an emergent work condition identified on the 1B residual heat removal service water system on February 10, 2005;
- Work Week February 20-25, 2005, including surveillance testing on the Unit 1 core spray system which made several risk significant systems inoperable and planned maintenance in the switchyard and anticipated transient without scram breakers;
- Work Week March 6-12, 2005, including planned maintenance in the switchyard and surveillance testing on the Unit 1 anticipated transient without scram relays and logic, the Unit 1 125 Volt battery, and the Unit 2 reactor core isolation cooling system;
- An evaluation of a previously identified issue regarding the risk associated with residual heat removal service water maintenance;
- An evaluation of the risk associated with a predicted switchyard low voltage condition; and
- An evaluation of the risk associated with the unexpected loss of Busses 18 and 19.

b. Findings

The inspectors identified one Green finding and one Non-Cited Violation (NCV) during their review. See Section 4OA5 for details.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors assessed the following operability evaluations or issue reports associated with equipment operability issues:

- Issue Report 236954 - Unit 1 Drywell Floor Drain Sump Pump Tripped Thermals;

- Issue Report 209270 - Unit 2 “B” Feedwater Heater Shells May Go Below Minimum Wall Thickness Before Next Refueling Outage;
- Issue Report 223039 - Turbine Control Valve #3 Failed to Fast Close During Testing;
- Issue Report 285762 - Unit 1 Core Recirc Flow Loop B Indication;
- Operability Evaluation 296236 - Unit 2 125 VDC System Ground;
- Issue Report 300636 - Minimum Wall Requirement for RHRSW Line 1-1005B-16 Not Met;
- Issue Report 301135 - Repair/Examination of Line 1-1005A-16" is Required;
- Issue Report 298438 - Potential for Electromatic Relief Valves Not to De-Energize During Loss of Coolant Accident Conditions; and
- Issue Report 309971 - Control Rod Drive Hydraulic Control Unit Accumulators Installed with Split Flanges.

The inspectors reviewed the technical adequacy of the evaluation against the Technical Specifications, Updated Final Safety Analysis Report, and other design information; determined whether compensatory measures, if needed, were taken; and determined whether the evaluations were consistent with the requirements of LS-AA-105, “Operability Determination Process,” Revision 0.

In addition, the inspectors reviewed selected issues that the licensee entered into its corrective actions program to verify that identified problems were being entered into the program with the appropriate characterization and significance. This review represented the completion of nine inspection samples.

b. Findings

One Green finding was identified due to the licensee’s failure to initiate operability determinations/evaluations for equipment discussed in the Updated Final Safety Analysis Report which was determined to be degraded. See Section 4OA2.2 of this report for additional details.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors assessed the operator workaround listed below to determine the potential effects on the functionality of the corresponding mitigating systems. During these inspections, the inspectors reviewed the technical adequacy of the workaround documentation against the Updated Final Safety Analysis Report and other design information to assess whether the workaround conflicted with any design basis information. The inspectors also compared the information in abnormal or emergency operating procedures to the workaround information to ensure that the operators maintained the ability to implement important procedures when needed. This review represented the completion of one inspection sample.

- Operator Workaround 04-013 - Degraded Switchyard Voltage Issues and Transformer Loading Concerns During a Loss of Coolant Accident.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

a. Inspection Scope

During the inspection period, the inspectors reviewed the following permanent plant modifications:

- Engineering Change 352570, "Permanent Repair of Service Water Header 1-3902-30" - O Branch Leak on Header Side of 1-3999-685 Valve," Revision 0; and
- Engineering Change 351170, "Reactor Building Opening and Replacement Siding," Revision 0.

The inspectors reviewed the design adequacy of the modifications by verifying one or more of the following:

- energy requirements were able to be supplied by supporting systems under accident and event conditions;
- replacement components were compatible with physical interfaces;
- replacement component properties met functional requirements under event and accident conditions;
- replacement components were environmentally and seismically qualified;
- sequence changes remained bounded by the accident analyses and loading on support systems was acceptable;
- response times for structures, systems, and components were sufficient to serve accident and event functional requirements assumed by the design analyses;
- control signals were appropriate under accident and event conditions; and
- affected operations procedures were revised and training needs were evaluated in accordance with station administrative procedures.

The inspectors verified that the post modification testing demonstrated system operability by verifying no unintended system interactions occurred, system performance characteristics met the design basis, and post-modification testing results met all acceptance criteria. The inspectors also reviewed issue reports related to permanent plant modifications to ensure that the licensee was entering issues into their corrective action program at an appropriate threshold. These reviews represented the completion of two inspection samples.

b. <u>Findings</u>

Engineering Change 351170 and two temporary modifications listed in Section 1R23 of this report, were developed in support of the steam dryer replacement project. Engineering Change 351170 was initiated to allow the removal and replacement of portions of the reactor building siding. Engineering Changes 351171 and 351277 governed the installation of a new exterior door and a temporary steam dryer enclosure. While reviewing these modifications, the inspectors developed a concern regarding the safety classification of the modifications. Specifically, the licensee had classified each of these modifications as non-safety related even though the reactor building siding, the door, and the enclosure would each serve as part of the secondary containment structure at certain times. In addition, the inadequate classification may have resulted in the licensee installing these modifications without implementing the additional checks and balances required for a safety-related modification.

The inspectors discussed this concern with engineering and regulatory assurance personnel. During these discussions, the inspectors were presented with information which appeared to support the licensee's decision to classify these modifications as non-safety related. The inspectors performed a review of previous NRC documents pertaining to the reactor building siding and identified information which conflicted with the licensee's information. The inspectors provided all of the information to members of the Office of Nuclear Reactor Regulation for additional review and a final determination regarding the appropriate safety classification.

At the conclusion of the inspection period, the licensee had performed evaluations and testing which demonstrated that the door and enclosure could perform the same functions as the existing secondary containment structure. However, the Office of Nuclear Reactor Regulation had not yet determined whether the licensee had appropriately classified the modifications discussed above. As a result, the inspectors considered this item to be unresolved pending a final decision by the Office of Nuclear Reactor Regulation **(URI 05000254/2005002-01; 05000265/2005002-01)**.

1R19 <u>Post Maintenance Testing</u> (71111.19)

a. <u>Inspection Scope</u>

The inspectors reviewed the post maintenance testing activities listed below during the inspection period:

- Corrective maintenance on the Unit 1 high pressure coolant injection inlet drain pot performed using Work Order 765685;
- Emergent maintenance to replace two agastat relays in the Unit 1 high pressure coolant injection system logic performed using Work Orders 7656936 and 774282;
- Corrective maintenance on the Unit 1 C reactor feedwater pump performed using Work Order 435249;

- Installation of steam dryer enclosure and door on refuel floor using Work Order 731537;
- Repair of Unit 1 source range monitor 24 using Work Order 637456; and
- Troubleshooting associated with the unexpected failure of the Transformer 22 to Bus 24 breaker.

For each post maintenance activity selected, the inspectors reviewed the Technical Specifications and Updated Final Safety Analysis Report against the maintenance work package to determine the safety function(s) that may have been affected by the maintenance. Following this review the inspectors verified that the post maintenance test activity adequately tested the safety function(s) affected by the maintenance, that acceptance criteria were consistent with licensing and design basis information, and that the procedure was properly reviewed and approved. When possible the inspectors observed the post maintenance testing activity and verified that the structure, system, or component operated as expected; test equipment used was within its required range and accuracy; jumpers and lifted leads were appropriately controlled; test results were accurate, complete, and valid; test equipment was removed after testing; and any problems identified during testing were appropriately documented. These reviews represented the completion of six inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the licensee’s outage schedule, verified equipment alignments, and observed control room and outage activities. The inspectors verified that the licensee effectively conducted the shutdown; managed elements of risk pertaining to reactivity control during and after the shutdown; and implemented decay heat removal system procedure requirements as applicable.

The inspectors performed the following activities daily:

- attended control room operator and outage management turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- performed periodic walkdowns of the turbine and reactor buildings to observe ongoing work activities; and
- reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance.

Additionally, the inspectors observed the following specific activities, as appropriate:

- shutdown and cooldown to a cold shutdown condition (MODE 4);
- implementation of abnormal operating procedures to address any abnormal occurrences;
- initiation of the shutdown cooling mode of the residual heat removal system;
- control rod withdrawals to criticality and portions of the plant power ascension;
- surveillance tests throughout the duration of the outage;
- troubleshooting efforts for emergent plant equipment issues; and
- reactor vessel disassembly.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing activities and/or reviewed completed surveillance test packages for the tests listed below:

- QCOS 0203-08, "Unit 1 On-line Automatic Blowdown Logic Test";
- QCIS 1300-03, "Reactor Core Isolation Cooling Steam Line High Flow Calibration and Functional Test," and QCOS 1300-06, "Reactor Core Isolation Cooling System Power Operated Valve Test";
- QCOS 1300-18, "Reactor Core Isolation Cooling Drain Pot Level Switch and Drain Valve Operability Test," and QCOS 1300-22, "Reactor Core Isolation Cooling Contaminated Condensate Storage Tank Suction Check Valve Closure Test";
- QCOS 1400-12, "Unit 1 Core Spray Logic Functional Test";
- QCOS 2300-07, "High Pressure Coolant Injection Pump Performance Test";
- QCOS 2300-29, "Unit 1 High Pressure Coolant Injection System Logic Functional Test";
- QCOS 3200-04, "Reactor Feedwater Check Valve 1-220-59A/B and Safe Shutdown Injection Check Valve Closure Test";
- MA-AB-725-112, "Preventive Maintenance Inspection of General Electric 480 Volt Circuit Breakers and Cubicles";
- QCTS 0920-04, "Source Range Monitoring and Intermediate Range Monitoring Overlap Testing";
- QCTS 0750-05, "Snubber Functional Testing"; and
- QCTS 0600-05, "Main Steam Isolation Valve Local Leak Rate Testing."

The inspectors verified that the structures, systems, components, or barriers tested were capable of performing their intended safety function by comparing the surveillance procedure or calibration acceptance criteria and results to design basis information contained in Technical Specifications, the Updated Final Safety Analysis Report, and licensee procedures. The inspectors verified that each test was performed as written, the data was complete and met the requirements of the procedure, and the test equipment range and accuracy were consistent with the application by observing the performance of the activity. Following test completion, the inspectors conducted walkdowns of the associated areas to verify that test equipment had been removed and that the system or component was returned to its normal standby configuration. The inspectors also reviewed actions taken in response to Issue Report 310140 which was generated during the inspections. The reviews listed above represented the completion of eleven inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed documentation for and installation of the following temporary configuration changes:

- Engineering Change 347977 - Steam Dryer Replacement Project Platform Installation and Removal;
- Temporary Configuration Change 353675 - Disable 151N Relays for Bus 13 and 14 Main and Reserve Feed Breakers;
- Engineering Change 351323 - Using the Alternate 125 Volt Battery as a 250 Volt Battery Substitute;
- Engineering Change 351277 - Unit 1 Steam Dryer Temporary Enclosure;
- Engineering Change 351171 - Enclosure Exterior Door, Semi-Permanent Hardware, and Sheet Metal Panels; and
- Engineering Change 350830 - Addition of Instrumentation Feedthrough Modules at X-102B and Rework of Drywell Vent Booster Fan Power Cable.

The inspectors assessed the acceptability of each temporary configuration change by comparing the 10 CFR 50.59 screening and evaluation (if required) and design information against the Updated Final Safety Analysis Report and Technical Specifications. The comparisons were performed to ensure that the new configurations remained consistent with design basis information. The inspectors performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability, and that operation of the modifications did not impact the operability of any interfacing systems. The inspectors also reviewed condition reports initiated during or following the temporary modification installation to ensure that problems encountered during the installation were appropriately resolved. This review represented the completion of six inspection samples.

b. Findings

No findings of significance were identified. However, see Section 1R17 of this report for discussion of an unresolved item related to Engineering Changes 351171 and 351277.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing (71114.02)

a. Inspection Scope

The inspectors discussed with corporate and station-based Emergency Preparedness staffs the operation, maintenance, and periodic testing of the Alert and Notification System in the Quad Cities Nuclear Power Station's plume pathway Emergency Planning Zone to determine whether the Alert and Notification System equipment was adequately maintained and tested in accordance with Emergency Plan commitments and procedures. The inspectors reviewed records of 2003 and 2004 preventive and non-scheduled maintenance activities, as well as July 2004 through December 2004 Alert and Notification System operability test results.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed and discussed with station emergency preparedness staff the procedures that included the primary and alternate methods of initiating an emergency response organization activation to augment the onshift emergency response organization and the provisions for maintaining the station's emergency response organization call-out roster. The inspectors also reviewed reports and a sample of corrective action program records of unannounced off-hours augmentation drills, which were conducted monthly between January 2003 and December 2004, to determine the adequacy of the drills' critiques and associated corrective actions. The inspectors also reviewed the emergency preparedness training records of a random sample of 75 Quad Cities Nuclear Power Station emergency response organization members, who were assigned to key and support positions, to determine whether they were currently trained for their assigned emergency response organization positions.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed a screening review of Revision 16 of the Exelon Standardized Emergency Plan and reviewed the licensee's 50.54(q) evaluation of the changes identified in Revision 16 to determine whether these changes decreased the effectiveness of the licensee's emergency planning for its Illinois nuclear power stations. The inspectors also performed a screening review of the associated Revision 19 of the Quad Cities Annex to the Standardized Emergency Plan and both 50.54(q) evaluations of the changes incorporated in Revision 19 to determine whether changes identified in Revision 19 decreased the effectiveness of the licensee's emergency planning for the Quad Cities Nuclear Power Station. The inspectors reviewed a sample of letters of agreement with offsite support organizations associated with the Quad Cities Nuclear Power Station to determine whether these agreements were current and whether the types of support to be provided were consistent with statements in the Quad Cities Annex to the Standardized Emergency Plan. This review did not constitute an approval of the changes, and as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed a sample of nuclear oversight staff's 2003 and 2004 audits and objective evidence reports on the Quad Cities Nuclear Power Station's emergency preparedness program to verify that these independent assessments met the requirements of 10 CFR 50.54(t). The inspectors also reviewed critique reports and samples of corrective action program records associated with those audits and with two actual emergency events that occurred in 2003 and 2004. The inspectors reviewed critique reports and samples of corrective action program records associated with the 2004 biennial exercise, as well as various emergency preparedness drills conducted in 2003 and 2004, in order to verify that the licensee fulfilled its drill commitments and to evaluate the licensee's efforts to identify, track, and resolve concerns identified during these activities. The inspectors also reviewed samples of implementing procedure revisions that were associated with corrective action records to verify that these procedures were adequately revised.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors conducted several emergency preparedness drill observations during the inspection period. As part of the first observation, the inspectors evaluated an operations crew during a simulator drill on January 10, 2005. The simulator scenario involved a loss of normal feedwater, a loss of coolant accident, the loss of Bus 14-1, and a manual blowdown of the reactor vessel.

On March 4, 2005, the inspectors observed members of the licensee's emergency preparedness organization in both the simulator and the technical support center during a planned emergency preparedness performance indicator drill. The drill scenario consisted of flooding in the 1B residual heat removal room, a reactor water cleanup system leak, an anticipated transient without scram, the failure of the main steam isolation valves to close, a fuel failure, and a simulated release of radioactivity to the environment.

During the drills the inspectors ensured that event classification, notifications, and protective action recommendations were timely, accurate, and correctly communicated by reviewing actual plant data, notification worksheets, and observing actual drill activities. The inspectors also attended the licensee's drill critiques to ensure that any weaknesses or deficiencies noted during the drill were also recognized by the licensee's drill evaluators. These observations represented the completion of three inspection samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Emergency Preparedness

.1 Reactor Safety Strategic Areas

a. Inspection Scope

The inspectors reviewed the licensee's records associated with the three emergency preparedness performance indicators listed below. The inspectors verified that the licensee accurately reported these indicators in accordance with relevant procedures and Nuclear Energy Institute guidance endorsed by the NRC. Specifically, the inspectors reviewed licensee records associated with performance indicator data reported to the NRC for the period July 2004 through December 2004. Reviewed records included: procedural guidance on assessing opportunities for the three performance indicators; assessments of performance indicator opportunities during pre-designated Control Room Simulator training sessions, the 2004 biennial exercise, and "mini-drills"; revisions of the roster of personnel assigned to key emergency response organization positions; and results of periodic Alert and Notification System operability tests. The following performance indicators were reviewed:

Common

- Alert and Notification System;
- Emergency Response Organization Drill Participation; and
- Drill and Exercise Performance.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of the inspectors' observations are included in the respective inspection scopes of each section of this report.

b. Findings

No findings of significance were identified.

.2 Review of Operability Determination/Evaluation and Operational Decision Making Processes

a. Inspection Scope

During the daily review of issue reports, the inspectors identified several concerns with the processing of reports which documented the degradation of equipment discussed in the Updated Final Safety Analysis Report. The inspectors selected the following issue reports for additional review to verify that the licensee had appropriately evaluated whether the degraded equipment would continue to perform its specified function using the operability determination and evaluation process. This review represented the completion of one annual inspection sample.

- Issue Report 236954 - Drywell Floor Drain Sump Pump 1B Degraded;
- Issue Report 209270 - “B” Feedwater Heater Shells May Go Below Minimum Wall by Q2R18;
- Issue Report 223039 - Turbine Control Valve #3 Failed to Fast Close During Testing; and
- Issue Report 285762 - Unit 1 Core Recirculation Flow Loop “B” Indication.

b. Findings

The inspectors determined that the licensee had effectively identified problems with the degraded equipment. However, the inspectors were concerned with the licensee’s evaluation and prioritization of each issue. The inspectors identified a Green finding due to the licensee’s failure to initiate operability determinations/evaluations when required and the failure to evaluate compensatory measures as described in Generic Letter 91-18 and Procedure LS-AA-105.

Description of Concerns

Drywell Floor Drain Sump Pump 1B Degraded

The inspectors reviewed this issue and determined that problems with the 1B drywell floor drain sump pump thermal overloads tripping began on June 30, 2004. During this event, the licensee reset the thermal overloads and restored the pump to an operable status. On July 8, 2004, maintenance personnel initiated Issue Report 234582 when they discovered errors in setting the floor drain sump pump clearances. The issue report initiator hypothesized that the June 30 pump trip could have been due to having inadequate pump clearances. The licensee developed several actions in response to the July 8 issue report. However, none of these actions evaluated the possible connection between the inadequate clearances and the June 30 pump trip. The licensee initiated Issue Report 236954 on July 18, 2004, when a second trip of the

1B drywell floor drain sump pump occurred. The inspectors reviewed these issue reports in the aggregate and were concerned that the licensee had not completed an operability determination/evaluation even though there was information which indicated that the pump was degraded.

The inspectors conducted a control room tour and discovered that an information tag had been placed on the control switch for the 1B drywell floor drain sump pump. In addition, the switch had been placed in the pull to lock position. The inspectors discussed the information tag and the switch position with the control room operators. The operators explained that many personnel suspected that the thermal overloads actuated due to the presence of foreign material in the pump's suction. In late July, 2004 maintenance personnel switched the pump motor's electrical leads in an effort to expel the foreign material by making the pump operate backwards. However, this troubleshooting was unsuccessful. The inspectors questioned the operators regarding whether the sump pump was inoperable or operable but degraded. No clear answers were provided. Instead, operations personnel restated that the pump's control switch was in the pull-to-lock position.

The inspectors reviewed Generic Letter 91-18 and determined that operability determinations and evaluations were to be performed for any structure, system, or component described in the Updated Final Safety Analysis Report. The inspectors reviewed the Updated Final Safety Analysis Report and found that Updated Final Safety Analysis Report Section 9.3 clearly described the drywell floor drain system as having two pumps. The inspectors interviewed several members of the operations department about this issue. In addition, the inspectors conducted an additional review of Issue Report 236954 to determine whether the issue report provided a basis for continued operability of the 1B drywell floor drain sump pump. Within the body of the issue report, the initiator stated that the operation of the 1B drywell floor drain sump pump was questionable. The licensee's procedure and Generic Letter 91-18 clearly stated that the operability of equipment cannot be indeterminate. As a result, operations personnel should have declared the sump pump inoperable. However, operations personnel determined that an operability determination was not necessary since the remaining drywell floor drain sump pump was fully operable and the degraded pump did not result in an entry into the Technical Specifications.

The inspectors also discussed this issue with operations management. Operations management informed the inspectors that an operability determination/evaluation was not required because the Updated Final Safety Analysis Report statement describing the drywell floor drain system as a two pump system was in place to better explain the system features rather than the design and licensing basis of the system. The inspectors performed an additional review of the Updated Final Safety Analysis Report and disagreed with the licensee's position. This disagreement was based upon the fact that the Updated Final Safety Analysis Report clearly described each pump as having its own function. Specifically, one pump was designed to start when a high sump level condition occurred. The other pump was designed to start during a high-high sump level condition.

The inspectors discussed this information with operations personnel. The inspectors were informed that the licensee had changed how they operated the drywell floor drain sump pumps several years ago. Currently, the licensee maintained the drywell floor drain sump pump discharge valves in the closed position. As a result, the sump pumps were unable to start automatically as described in the Updated Final Safety Analysis Report. In addition, it appeared the licensee had substituted a manual action in place of an automatic action without evaluating the potential impacts of the manual actions. The licensee was evaluating this issue at the conclusion of the inspection period. Therefore, the inspectors considered this item to be unresolved pending a review of the licensee's evaluation **(URI 05000254/2005002-02; 05000265/2005002-02)**.

<u>"B" Feedwater Heater Shells May Go Below Minimum Wall by Q2R18</u>

Issue Report 209270 was initiated in March 2004 when licensee personnel discovered that two of the feedwater heater shell sections could degrade below the American Society of Mechanical Engineers, Section VIII, code required minimum wall thickness of 0.112 inches prior to the next refueling outage. The inspectors reviewed the Updated Final Safety Analysis Report and found that the feedwater heaters were discussed in Section 10.4.7. As a result, the inspectors concluded that an operability determination/evaluation needed to be performed prior to the heater shells degrading below minimum wall requirements to ensure that the heaters would continue to perform their intended function.

Procedure LS-AA-105, "Operability Determinations," provided allowances for documenting operability determinations within the body of an issue report. However, the initiator was expected to provide enough detail in the issue report to clearly demonstrate that continued operability was maintained. The inspectors reviewed Issue Report 209270 and found a reference to an engineering evaluation on the heaters. However, the issue report contained very few of the evaluation's details. As a result, the inspectors were unable to determine if the continued operability of the feedwater heaters could be maintained throughout the operating cycle.

During a review of the licensee's operational decision making document (a non-corrective action document) for this issue, the inspectors found that a copy of the engineering evaluation had been attached. The inspectors determined that the engineering evaluation was very detailed and provided a comprehensive discussion on the condition of the feedwater heaters. However, the inspectors were concerned that the licensee had not recognized the need to include a copy of the evaluation as part of the issue report in order to support continued operability of the heaters.

<u>Turbine Control Valve #3 Failed to Fast Close During Testing</u>

This issue report was initiated in May 2004 due to the failure of the turbine control valve #3 fast-acting solenoid to actuate during testing. The licensee entered Technical Specification 3.3.1.1, "Reactor Protection System Instrumentation," since they were unable to determine whether the valve's failure to close was due to a faulty pressure switch (which provides an input into the reactor protection system) or a degraded solenoid. In addition, operations personnel requested that engineering perform an operability evaluation to ensure that the reactor protection system functions provided by the pressure switch remained operable.

During subsequent troubleshooting activities, the licensee determined that the fast-acting solenoid was degraded rather than the pressure switch. Based upon this information, and a determination that the reactor protection system inputs were not impacted, operations personnel canceled the operability evaluation request. The decision to cancel the operability evaluation concerned the inspectors for two reasons. First, the issue report clearly stated that operation of the fast-acting solenoid was credited in the Updated Final Safety Analysis Report for load rejection without bypass events and loss of alternating current/loss of grid events. However, the decision to cancel the operability evaluation was based solely upon the ability to meet Technical Specifications. This demonstrated that operations personnel were not familiar with the requirement to perform operability evaluations on equipment described in the Updated Final Safety Analysis Report. Second, the issue report stated that the core operating limits report required a penalty to be applied to the core thermal limits due to the degraded solenoid. Per the operability evaluation process, the penalty application should have been considered a potential compensatory measure. Since the operability evaluation was not performed, the possibility of potential compensatory measures was not considered. After consultation with an NRC operability specialist, the inspectors subsequently determined that implementing a Core Operating Limits Report penalty was not a compensatory measure.

<u>Unit 1 Core Recirculation Flow Loop “B” Indication</u>

In December 2004 the licensee initiated Issue Report 285762 which documented the need to calibrate one of the Unit 1 core flow indications more frequently. The inspectors reviewed the issue report and found that operations personnel were concerned with the degraded indication because it had the potential to adversely impact the daily jet pump surveillances and could result in unknowingly exceeding Technical Specification limits. Although operations personnel were concerned with the degraded flow indication, they determined that the indication remained operable since all of the Technical Specification requirements were being met. In addition, Operational Decision Making Document 04-049 was developed to address continued plant operation with the degraded indication and the possible repair options.

The inspectors reviewed the operational decision making document and found that this document contained information which had not been included in the issue report. Specifically:

- The degraded indicator had been calibrated four to five times between March and December 2004. The normal calibration frequency was approximately six months;
- The time between calibrations was decreasing while the magnitude of the deviation between indications was increasing at a much quicker rate;

- The indicator would be calibrated on an as-needed basis until it could be repaired; and
- The impact of the degraded indicator on plant operations was that the thermal limit calculations and flow control line determinations were more conservative.

The inspectors determined that portions of the operational decision making document information were very similar to information that would be included in an operability determination or evaluation. The inspectors were also concerned that the decision to calibrate the indication more frequently (in an effort to keep the indication operable) needed to be evaluated as a compensatory measure. The inspectors discussed their concerns with operations and regulatory assurance personnel. Operations personnel explained that the need for an operability determination or evaluation was not considered since the degraded indication continued to meet the Technical Specification requirements. As a result, the compensatory measure was not evaluated.

Analysis of Risk Significance and Enforcement

The inspectors determined that the failure to initiate operability determinations or evaluations when required was a performance deficiency warranting a significance evaluation. The inspectors determined that the issue was more than minor because if left uncorrected, the failure to evaluate degraded equipment using the operability determination and evaluation process could become a more significant issue. The finding also impacted the cross-cutting area of problem identification and resolution because the licensee has had multiple examples of failures to initiate operability determinations or evaluations in accordance with LS-AA-05.

The inspectors completed a significance determination for each piece of degraded equipment using Inspection Manual Chapter 0609, "Significance Determination Process." This finding was determined to be of very low safety significance (Green) because the degraded pieces of equipment did not result in a total loss of safety function of any system. Although the licensee failed to perform operability determinations and evaluations in accordance with their procedure, no violation of NRC requirements occurred since this procedure is not required by any current NRC regulations **(FIN 05000254/2005002-03; 05000265/2005002-03)**. A description of the licensee's corrective actions is included in the following section.

Licensee's Corrective Actions

The licensee initiated Issue Report 311612 to document the inspectors concerns. The licensee's corrective actions consisted of the following:

- Developed briefing materials for operations, engineering, and management personnel which thoroughly explained the purposes of and differences between the operability determination/evaluation process and the operational decision making process;
- Developed briefing materials which emphasized that the operability determination/evaluation process was to be used not only for safety-related

equipment, but also for non-safety related equipment which supported safety-related equipment and non-safety related equipment discussed in the Updated Final Safety Analysis Report;

- Performed an operability determination on the degraded feedwater heater shells; and
- Initiated another issue report to ensure that the other turbine control valve fast acting solenoid functions remained operable.

4OA4 Cross-Cutting Aspects of Findings

A finding discussed in Section 4OA2.2 of this report had, as its primary cause, a problem identification and resolution deficiency, in that, the licensee had identified several examples of degraded, non-safety related equipment which was discussed in the Updated Final Safety Analysis Report. However, the licensee had not recognized the need to perform an operability determination or evaluation for each piece of equipment. In addition, the licensee had completed other non-corrective action program documentation which provided more information regarding corrective actions and operability impacts than what was included in the associated issue reports.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000254/2004009-05: Review of On-line Risk Assessment of Compensatory Actions Taken in Response to a Pinhole Leak.

Introduction: In following up on a previously identified unresolved item associated with the licensee's actions in response to an emergent work condition, the NRC identified a NCV of 10 CFR 50.65(a)(4). Specifically, the NRC identified that the licensee non-conservatively evaluated the on-line risk associated with actions taken in response to an emergent residual heat removal service water leak on January 14, 2003.

Description: On January 14, 2003, the licensee discovered a pinhole leak in the Unit 1, Train "B", residual heat removal service water piping downstream of the residual heat removal heat exchanger. The leak was in an expander just downstream of the normally closed heat exchanger outlet valve, 1-1001-5B. In order to isolate the leak, at 3:29 a.m., the licensee closed normally locked open manual valve 1-1001-201B. Upon closing the valve, the licensee declared the system inoperable, but determined that the system was still available and that on-line risk was still Green. However, in order for that train of residual heat removal service water to perform its safety function the manual valve would need to be reopened. At approximately 10:11 a.m., the licensee hung a work tag on the valve, declared the Train "B" residual heat removal service water system unavailable, and changed the on-line risk to Yellow. The inspectors noted that the on-line risk assessment was contingent upon the status of the residual heat removal service water system. Having one train of residual heat removal service water unavailable by itself increased the site risk above two times normal core damage frequency and changed the on-line risk from Green to Yellow.

Over the 6.7 hours when Train “B” residual heat removal service water was considered inoperable but available, the only notification to the operators that the normally locked open manual valve was closed was an entry in the operator logs. No written guidance was provided to alert operators that this closed manual valve would need to be reopened in order for the Train “B” residual heat removal service water system to perform its safety function. Additionally, no operator was dedicated to ensuring that the valve could be reopened. Also during this time period, a routine shift turnover occurred. The licensee indicated that, to the best of their knowledge, the shift turnover included discussion that the manual valve was closed, and that it would need to be reopened if the system was required; however, this discussion was not documented.

The inspectors reviewed the licensee's work control procedure WC-AA-101. Attachment 7 of this procedure contained examples to guide the operators in making determinations as to whether equipment could be declared “available.” These examples fell into categories including:

- operable equipment;
- inoperable equipment, tagged out of service;
- inoperable equipment due to off-normal alignment during testing with automatic realignment; and
- testing that would require operator action to restore system.

The inspectors noted that none of the examples dealt with emergent conditions where equipment was placed into an abnormal lineup as a compensatory action.

The inspectors determined that the situation most closely resembled either the case of “inoperable equipment, tagged out of service,” or “testing that would require operator action to restore system,” because Train B of the system could not perform its safety function without manual action above and beyond normal system initiation from the control room. For both cases, the licensee’s procedure allowed for equipment to be considered available, if written guidance was provided for restoration.

<u>Analysis</u>: The inspectors reviewed this issue against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports.” The inspectors concluded that the issue was more than minor since the finding involved a change in on-line risk level from Green to Yellow.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, “Significance Determination Process (SDP),” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that the issue was of very low safety significance, or Green, because, although one train of residual heat removal service water was unavailable, the actual safety function of the system could have been performed by the remaining train, the train was not inoperable for greater than the Technical Specification-allowed outage time, and the remaining Phase 1 questions were not applicable.

Enforcement: Title 10 CFR 50.65(a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activity. Contrary to the above, the licensee failed to perform an adequate risk assessment when the Unit 1 Train "B" residual heat removal service water system was rendered inoperable and unavailable on January 14, 2003. The failure to perform an adequate risk assessment resulted in the licensee inappropriately assigning an overall Green risk condition for the plant when actual plant conditions warranted a Yellow risk assessment. Because the failure to adequately assess on-line risk is of very low safety significance and has been entered into the corrective action program as Issue Report 304538, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2005002-04)**. Corrective actions for this issue including providing training to operations personnel which focused on crediting manual actions in place of automatic actions as part of a risk assessment.

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 April 5, 2005. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Emergency Preparedness inspection with Mr. T. Tulon on February 11, 2005.

- Closure of Unresolved Item 05000254/2004009-05 with Mr. T. Scott and Mr. W. Beck on February 22, 2005.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of NRC requirements which met the criteria of Section VI of the NRC Enforcement Manual, Nuclear Regulatory Guide (NUREG)-1600, for being dispositioned as Non-Cited Violations.

Cornerstone: Emergency Preparedness

Title 10 CFR 50.47 (b) (15) requires, in part, that radiological emergency response training is provided to those who may be called on to assist in an emergency. Table B-1 of the licensee's Standardized Emergency Plan required that the minimum on-shift staffing included two radiation protection personnel for in-plant protective actions. In

September 2004 emergency preparedness staff based at another of the licensee's Illinois nuclear stations identified that this emergency plan commitment was met by one on-shift radiation protection technician and one on-shift chemistry technician. However, the licensee also determined that chemistry technicians' training had evolved such that it no longer met all requirements to provide in-plant protection actions.

In early December 2004, the licensee completed an adequate root cause investigation of this concern's possible impact at each of its Illinois nuclear stations. Timely corrective actions included assigning two radiation protection technicians on all back shifts, initiating revision of the standardized emergency response organization training procedure, and initiating an assessment of emergency response organization position qualifications in cases where some emergency response organization training was being performed by other departments. Longer-term actions included provisions for an effectiveness review of measures taken to ensure that two qualified radiation protection technicians were always on-shift. Because no emergencies had occurred that required in-plant protective actions and the licensee's timely corrective actions included staffing a minimum of two radiation protection technicians on-shift, this violation is not more than of very low significance, and is being treated as a NCV.

Cornerstone: Mitigating Systems

Title 10 CFR 50.9 requires, in part, that information required by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

Technical Specification 5.4, "Procedures," required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978.

Regulatory Guide 1.33, Appendix A, Section 8, required, in part, that procedures of a type appropriate to the circumstances should be provided to ensure that tools, gauges, instruments, controls, and other measuring and testing devices are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy.

Quad Cities Procedure MA-AA-716-100, "Maintenance Alterations Process," Revision 1, Section 4.2.2, required, in part, that if applicable, indicate whether an alteration or restoration verification is required by identifying the type of verification required CV, IV, or N/A.

Maintenance Alteration Logs for torus temperature indicators, residual heat removal suction and discharge pressure indicators, residual heat removal service water pump discharge indicators, and secondary containment differential pressure indicators required either concurrent or independent verifications to be performed after alteration and restoration of the instruments.

Contrary to the above, from January 28 to April 16, 2003, two instrument maintenance technicians at Quad Cities failed to perform required concurrent or independent verification while calibrating the torus temperature indicators, residual heat removal suction and discharge pressure indicators, residual heat removal service water pump discharge indicators, and secondary containment differential pressure indicators in accordance with the associated Maintenance Alterations Logs.

Additionally, the two technicians documented on the Maintenance Alteration Logs that the required concurrent or independent verifications had been completed by another technician. This information is material to the NRC because it demonstrated compliance with the Commission's regulations and procedures of the Quad Cities Nuclear Power Station.

The NRC Office of Investigation investigated the matter and concluded that the individual deliberately falsified Maintenance Alteration Logs. Since the incident was determined to be a deliberate violation of NRC requirements, the violation was subject to the traditional enforcement process instead of the NRC's Significance Determination Process. The violation was categorized in accordance with the NRC's Enforcement Policy at Severity Level IV. On February 28, 2005, after considering the circumstances of the case and after consulting with the Director, Office of Enforcement, this violation was treated as a Non-Cited Violation (ADAMS Accession No. ML050600140), consistent with Section VI.A.1.d of the NRC's Enforcement Policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Tulon, Site Vice President
R. Gideon, Plant Manager
R. Armitage, Training Manager
D. Barker, Work Control Manager
W. Beck, Regulatory Assurance Manager
T. Hanley, Maintenance Manager
W. Harris, Emergency Preparedness Manager
D. Hieggelke, Nuclear Oversight Manager
K. Moser, Deputy Engineering Manager
V. Neels, Chemistry/Environ/Radwaste Manager
K. Ohr, Radiation Protection Manager
M. Perito, Operations Manager
A. Scott, Operations

Nuclear Regulatory Commission
M. Ring, Chief, Reactor Projects Branch 1
L. Rossbach, Project Manager, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000254/2005002-01 05000265/2005002-01	URI	Inadequate Classification of Modifications (Section 1R17)
05000254/2005002-02 05000265/2005002-02	URI	Drywell Floor Drain Sump Pump 1B Degraded (Section 4OA2.2)
05000254/2005002-03 05000265/2005002-03	FIN	Failure to Initiate Operability Determinations or Evaluations When Required (Section 4OA2.2)
05000254/2005002-04	NCV	Review of On-Line Risk Assessment of Compensatory Actions Taken in Response to a Pinhole Leak (Section 4OA5)

Closed

05000254/2005002-03 05000265/2005002-03	FIN	Failure to Initiate Operability Determinations or Evaluations When Required (Section 4OA2.2)

05000254/2005002-04	NCV	Review of On-Line Risk Assessment of Compensatory Actions Taken in Response to a Pinhole Leak (Section 4OA5)
05000254/2004009-05	URI	Review of On-Line Risk Assessment of Compensatory Actions Taken in Response to a Pinhole Leak (Section 4OA5)

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 1-1300-03; Unit 1 Reactor Core Isolation Cooling System Valve Checklist; Revision 7
QOM 1-1300-02; Unit 1 Reactor Core Isolation Cooling System Valve Checklist (Reactor Core Isolation Cooling System Room); Revision 5
QCOP 1300-01; Reactor Core Isolation Cooling System Preparation for Standby Operation; Unit 1; Revision 27
QOM 1-1000-05; Unit 1 Residual Heat Removal Service Water Valve Checklist; Revision 16
QOM 1-1000-07; Residual Heat Removal and Residual Heat Removal Service Water System Fuse and Breaker Checklist; Revision 3
QOM 2-1300-02; Unit 2 Reactor Core Isolation Cooling Valve Checklist (RCIC Room); Revision 7
QOM 2-1300-03; Unit 2 Reactor Core Isolation Cooling Valve Checklist (Not in RCIC Room); Revision 8
QOM ½-2900-01; Unit ½ Safe Shutdown Makeup Pump System Checklist; Revision 3
Issue Report 296236; Level III Grounds Detected on Unit 2 125 VDC; dated February 1, 2005
Issue Report 305392; 1A Reactor Protection System EPA 1A-2 Underfrequency Time Outside Condition Report Tolerance; dated February 25, 2005
List of Issue Reports on the Reactor Protection System Generated Since January 2, 2004; dated February 2, 2005
QCOP 0300-24; Installation/Removal of a Temporary Valve for a Leaking 1(2)-0302-17 Valve; Revision 1
Engineering Change 349215; Procedure to Implement Temporary Pressure Boundary Restoration of Hydraulic Control Unit 107 Valve; dated June 4, 2004
List of Open Work Requests on the Reactor Protection System; dated February 14, 2004
QCOP 1400-07; Core Spray Operation with the Torus Unavailable; Revision 6
QCOP 1400-01; Core Spray System Preparation for Standby Operation; Revision 16
QOM 1-1400-09; U1 A Core Spray Valve Checklist; Revision 3

1R05 Fire Protection

Issue Report 289748; Fire Door #192 Requires Assistance to Close; dated January 11, 2005

Issue Report 293853; Discrepancy of Fire Pre-Plan TB 112 Information; dated January 24, 2005
Issue Report 293899; 1-4199-163 Unit 1 Low Pressure Heater Bay Pull Space Deluge Valve Packing Leak; dated January 24, 2005

1R06 Heat Sink Performance

Quad Cities Generic Letter 89-13 Heat Exchanger Tube Plugging Data - Unit 1; dated March 25, 2005
Engineering Evaluation 333328; Provide a Tube Plugging Limit for the Diesel Generator Heat Exchangers; dated October 10, 2001

1R12 Maintenance Effectiveness

Issue Report 111121; Leak Discovered on Piping Weld on 1B RHRSW Low Pressure Pump; dated June 8, 2002
Issue Report 126235; RHR logic Electric Lead Disconnected; dated October 7, 2002
Issue Report 133088; Leak Discovered at Threaded Vent Valve to 2C RHRSW Low Pressure Pump; dated November 25, 2002
Issue Report 189408; 2B RHRSW Pump Leak; dated December 6, 2003
Issue Report 200744; 2B RHRSW High Pressure Pump Discharge Piping Through Wall Leak; dated February 10, 2004
Issue Report 261453; Leak on RHRSW Pipe Fitting Weld on Line 1-1005A-16"-D, dated October 7, 2004
Issue Report 278088; Time Delay Out of Tolerance on 1D RHR Pump Trip Relay; dated December 1, 2004
Issue Report 261415; 1A RHRSW Motor Had Three Successive Starts; dated October 7, 2004
Issue Report 192905; 1D RHRSW d/p below QCOS 1000-04 Acceptance Criteria; dated December 30, 2003
Issue Report 189928; Additional Corrective Action Prudent for CR110756; dated December 10, 2003
Issue Report 186008; RHR Heat Exchanger Head Vent Relief Valve 1-1001-165A Failed Seat Leakage Test; dated November 11, 2003
Issue Report 243624; Degrading Trend in Valve Condition Load on 1-1001-7A; dated August 11, 2004
Issue Report 187704; Extent of Condition for 2B RHRSW Pump Casing Pitting; dated November 19, 2003
Issue Report 184695; NRC NCV Concerning Failure of the Shutdown Cooling Valve; dated November 4, 2003
Issue Report 170060; Unexpected ADS System 1 and 2 Main DC Power Failure Alarm/Reset; dated August 1, 2003
Issue Report 141940; Main Steam Safety Valve Temperature Indication; dated January 30, 2003
Issue Report 151833; Relay 287-105B Failed to Time Out After Adjustment; dated April 1, 2003

Control Room Logs; Various dates
Regulatory Guide 1.160; Monitoring the Effectiveness of Maintenance of Nuclear Plants
NRC Maintenance Rule Guideline Book
ER-AA-310; Implementation of the Maintenance Rule; Revision 3
ER-AA-20; Equipment Reliability Program Description; Revision 2
ER-AA-310-1001; Maintenance Rule Scoping; Revision 1
ER-AA-310-1004; Maintenance Rule Performance Monitoring; Revision 2
ER-AA-310-1005; Maintenance Rule Dispositioning Between (a)(1) and (a)(2); Revision 2
Issue Report 314034; Unit 1 TBCCW System Pipe Hangers Hardware Deficiencies; dated March 17, 2005
Issue Report 292679; Ultrasonic Examination of Service Water Lines Found to be Below Tmin; dated January 20, 2005
Issue Report 267665; Extent of Condition Review from Pipe Support/Hanger Issues; dated October 26, 2004
Issue Report 266734; Unit 1 Reactor Feedwater Pump U-Bolt Pipe Supports Have Loose/Missing Jab Nuts; dated October 21, 2004
Issue Report 266747; Unit 2 Reactor Feedwater Pump U-Bolt Pipe Supports Have Loose/Missing Jab Nuts; dated October 21, 2004
Issue Report 265729; TBCCW Piping Hanger Issues in Cribhouse; dated October 20, 2004
Issue Report 206292; Broken Casing on 2A Reactor Feedwater Pump Seal Cooler During Reassembly; dated March 5, 2004
Issue Report 214046; 2A Reactor Feedwater Pump Speed Changer Bearing Temperature Above Administrative Limit; dated April 7, 2004
Issue Report 262177; 2A TBCCW Pump Bubbler was Found Empty and Filled Twice; dated October 10, 2004
Issue Report 164392; Question on Heat Exchanger Alignment for TBCCW Procedures; dated June 23, 2003
Issue Report 166318; Both 1A and 1B TBCCW Heat Exchangers Have High Differential Pressure; dated July 4, 2003
Issue Report 170412; Concerns Regarding 1-3905 Breaker Remaining Out of Service; dated August 5, 2003
Issue Report 181803; Concerns with Bundling of Work on 2A TBCCW Heat Exchanger; dated October 19, 2003

1R13 Maintenance Risk Assessment and Emergent Work

Daily Production Schedules; dated January 3-8, 10-16, 17-23, 24-29, February 20-25, and March 6-12, 2005
Work Week Risk Assessment for the Weeks of January 3, 10, 17, and 24, February 20, and March 6, 2005
Risk Assessment for Emergent Condition on Predicted Switchyard Low Voltage;
Prompt Investigation Report for IR 317820
QOA 6700-04, “480 V Bus 18 (29) Failure,” Revision 18

1R15 Operability Evaluations

Issue Report 298438; Potential for Electromatic Relief Valves Not to De-Energize During Loss of Coolant Accident Conditions; dated February 7, 2005
Updated Final Safety Analysis Report
Issue Report 300636; Minimum Wall Requirement for RHRSW Line 1-1005B-16" Not Met; dated February 11, 2005
Issue Report 298665; Ultrasonic Examination Results on RHRSW Line 1-1005B-16"; dated February 7, 2005
LS-AA-105; Operability Determinations; Revision 1
Technical Specifications
QGA 100; Reactor Pressure Vessel Control; Revision 7
IR 309971, “CRD [control rod drive] HCU [hydraulic control unit] BWR 6 Accumulators installed with BWR4 Split Flanges.”
Operability Evaluation 3099771, Revision 0
EC 354314, “Seismic Evaluation and the CRD HCU with the BWR 6 Scram Water accumulator in Combination with the BWR 4 Split Flanges

1R17 Permanent Plant Modifications

Engineering Change 351170; Reactor Building Opening and Siding Replacement; Revision 0
Safety Evaluation QC-E-2004-006; 50.59 Evaluation for Modification 351170 and QCTS 0410-02; Revision 0
QCOA 0010-10; Tornado Watch/Warning, Severe Thunderstorm Warning, or Severe Winds; Revision 14
QCOS 1600-34; Monthly Secondary Containment Integrity Surveillance; Revision 8
Safety Evaluation 97-029; 50.59 Evaluation for Cancellation of UFSAR Change No. 97-3; dated February 26, 1997
QDC-7500-—0031; Volume of Secondary Containment; Revision 4
QDC-9400-—0348; Assessment of Control Room Habitability with Increased Standby Gas Treatment Filter Efficiency; Revision 1
QDC-0000-–1020; Impact of Extended Power Uprate on Site Boundary and Control Room Doses for LOCA and Non-LOCA Events; Revision 1
QDC-0020-S-0176; Quad Cities Units 1 and 2 Design Basis Siding Qualification; Revision 1
QDC-0000-S-1410; Tornado Depressurization Analysis for the Design of the Dryer Enclosure as Part of the Steam Dryer Replacement Project; Revision 0
QDC-0020-S-1402; Structural Evaluation Associated with the Design of a Temporary Enclosure on Refueling Floor for the Steam Dryer Replacement Project
Inspection Report 50-254/97028 and 50-265/97028; dated March 6, 1998
EA 97-266; Apparent Violation of 10 CFR 50.9; dated September 17, 1997
EA 97-413; Exercise of Enforcement Discretion and Notice of Violation; dated June 5, 1998
Inspection Report 50-254/96017 and 50-265/97017; dated February 4, 1997
Special Inspection Report 50-254/96019 and 50-265/96019; dated February 4, 1997
Letter ESK-97-152; Letter from Commonwealth Edison to the NRC; dated July 24, 1997

Generic Letter 83-28; Required Actions Based on Generic Implications of Salem Anticipated Transient Without Scram Events; dated July 8, 1983

1R19 Post Maintenance Testing

Issue Report 310140; NRC Identified Issues During Performance of Unit 2 RCIC Surveillances; dated March 8, 2005
Issue Report 292454; Off Light Indication for High Pressure Coolant Injection Emergency Oil Pump Will not Stay Lit; dated January 20, 2005
Issue Report 292358; Unit 1 High Pressure Coolant Injection Turbine Inlet Drain Pot Level Switch Failure; dated January 20, 2005
Issue Report 292726; Time Delay Relay Failed During Logic Test; dated January 21, 2005
Issue Report 292734; Limit Switch Failed Continuity Check During Logic Test; dated January 21, 2005
Issue Report 295613; Failure to Remove Revision F of Inboard Reactor Feed Pump Seal; dated January 28, 2005
Issue Report 295174; Large Steam Leak from 1C Reactor Feed Pump Seal; dated January 28, 2005
Work Order 731537; Remove/Reinstall Metal Siding on Reactor Building Exterior to Allow Steam Dryer Access; dated March 16, 2005
QCTS 0410-02; Secondary Containment Capability Test; Revision 7
Work Order 732489; Pressure Test Enclosure Structure for Steam Dryer Access; dated February 25, 2005
Issue Report 296081; Poorly Designed 1C Reactor Feed Pump Seal Housing Impacts Pump's Return to Service; dated January 28, 2005

1R22 Surveillance Testing

Work Order 561252; HPCI Performance Test; dated January 11, 2005
QCOS 2300-01; HPCI Preparation for Standby Operation; Revision 42
Issue Report 290031; Delay due to Procedure Issue; dated January 12, 2005
Issue Report 290207; Out of Tolerance - 1-287-121B, dated January 12, 2005
Issue Report 292468; Signal Converter Had No Response; dated January 19, 2005
Issue Report 292313; High Pressure Coolant Injection Relay 1-2330-123 Not Replaced During High Pressure Coolant Injection Work; dated January 19, 2005
Schematic Control Diagram 4E-1526; HPCI System Block Diagram 8 Control Switch Development; Revision T
Schematic Diagram 4E-1527, Sheet 1; High Pressure Coolant Injection System Sensors and Auxiliary Relays; Revision T
Schematic Diagram 4E-1527, Sheet 2; High Pressure Coolant Injection System Sensors and Auxiliary Relays; Revision H
Schematic Diagram 4E-1527, Sheet 3; High Pressure Coolant Injection System Sensors and Auxiliary Relays; Revision D
Schematic Diagram 4E-1528, Sheet 1; High Pressure Coolant Injection System Valves and Turbine Auxiliaries; Revision AT

Schematic Diagram 4E-1528, Sheet 2; High Pressure Coolant Injection System Valves and Turbine Auxiliaries; Revision AH
Issue Report 316431; NRC Identified Discrepancy in Documentation; dated March 23, 2005
Work Order 629202; Main Steam Isolation Valve Combined Local Leak Rate Testing; dated March 3, 2005
Work Order 568060; Safety Related Mechanical Snubber Testing; dated March 15, 2005
Action Item 36958-02; Root Cause Report for Main Steam Isolation Valve D Failure; dated November 14, 2000
Issue Report 315544; Main Steam Isolation Valve Wet Test Options; dated March 21, 2005
Issue Report 315590; Main Steam Isolation Valve Failure As-Found Local Leak Rate Test; dated March 22, 2005
Issue Report 315636; Safe Shutdown Makeup Pump Discharge Check Valve Fails QCOS 3200-04; dated March 22, 2005
WO 795647-01, "EM [electrical maintenance] Investigate Bus 18 480 V Feed Breaker Trip"
WO 396202-01, "480 V Main/Bus Tie Breaker Inspection"
NP-7410, Volume 1, "Circuit Breaker Maintenance Volume 1 Low-Voltage Circuit Breakers Part 2 GE AK Models, dated July 1992

1R23 Temporary Modifications

QOP 0020-01; Opening a Penetration in Secondary Containment; Revision 18
Issue Report 297548; 4160 Volt Relay and Metering Current Transformers Single Failure Vulnerability; dated February 3, 2005
Engineering Change 350830; Addition of Instrumentation Feed Through Modules at X-102B and Rework of the Drywell Booster Fan Power Cable; Revision 0
Engineering Change 351277; Temporary Enclosure; Revision 0
Engineering Change 351171; Enclosure Exterior Door, Enclosure Semi-Permanent Hardware, and Sheet Metal Panels; Revision 0
QDC-0000-S-1399; Design of Temporary Enclosure on Refueling Floor for Steam Dryer Replacement Project

1EP2 Alert and Notification System (ANS) Testing

EP-AA-125-1004; Emergency Response Facilities and Equipment Performance Indicator Guidance; Revision 3
Quad Cities Nuclear Power Station Off-Site Siren Test Plan; Revision 3; dated January 2002
Warning System Maintenance and Operational Report for Quad Cities Nuclear Power Station - October 15, 2003, through November 10, 2003
Warning System Maintenance and Operational Report for Quad Cities Nuclear Power Station - October 19, 2004, through November 22, 2004
Exelon Semi-Annual Siren Report For Quad Cities Nuclear Power Station - July 1, 2003, through December 31, 2003

Exelon Semi-Annual Siren Report For Quad Cities Nuclear Power Station - July 1, 2004, through December 31, 2004
NOS Objective Evidence Report 212019-21; EPZ Siren Program; dated April 28, 2004

1EP3 Emergency Response Organization (ERO) Augmentation Testing

EP-AA-112-100; Control Room Operations; Revision 7
EP-AA-112-100-F-01; Shift Emergency Director Checklist; Revision B
EP-AA-112-100-F-06; Midwest ERO Augmentation; Revision C
EP-AA-122-1001; Attachment 2; Conduct of Call-In Augmentation Drills; Revision 3
ERO Off-Hours, Unannounced, Off-Hours Augmentation Call-In Drill Records; January 2003 through December 2004
Internal Memorandum; March 2003 Drive-in Augmentation Drill Results; dated March 13, 2003
Quad Cities Nuclear Power Station ERO Roster; dated February 2005
NOS Objective Evidence Report 230291-23; Duty Teams' Staffing Levels and Readiness; dated July 30, 2004
CR 231594; One On-Call Responder's Estimated Arrival Time Was Two Minutes Beyond Time Limit in June 2004 Augmentation Drill
CR 231625; Five Members of On-Call Team Did Not Call in During June 2004 Augmentation Drill - Positions Filled by Other Qualified ERO Members
CR 267690; Three Members of On-Call Team Did Not Call in During October 2004 Augmentation Drill - Positions Filled by Other Qualified ERO Members
CR 274351; Three Members of On-Call Team Did Not Call in During November 2004 Augmentation Drill - Positions Filled by Other Qualified ERO Members
CR 283216; Five Members of On-Call Team Did Not Call in During December 2004 Augmentation Drill - Positions Filled by Other Qualified ERO Members

1EP4 Emergency Action Level and Emergency Plan Changes

50.54(q) Program Evaluation and Effectiveness Review 04-87 for Revision 16 of the Exelon Standardized Emergency Plan; dated October 27, 2004
50.54(q) Program Evaluation and Effectiveness Review 04-10 for Revision 19 of the Emergency Plan Annex for the Quad Cities Nuclear Power Station; dated May 24, 2004
50.54(q) Program Evaluation and Effectiveness Review 04-84 for Revision 19 of the Emergency Plan Annex for the Quad Cities Nuclear Power Station; dated November 3, 2004
Letters of Agreement With Four Off-Site Support Organizations for Quad Cities Nuclear Power Station

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

EP-AA-122; Drills and Exercises; Revision 4
EP-AA-122-1001; Drill Development, Conduct, and Evaluation; Revision 4
Internal Memorandum; Quad Cities Nuclear Power Station April 2003 Alert Event Critique Report; dated May 16, 2003

Internal Memorandum; Quad Cities Nuclear Power Station June 2004 Unusual Event Critique Report; dated July 26, 2004
Event Summary Report - Quad Cities Nuclear Power Station Unusual Event on June 28,2004; dated February 7, 2005
Internal Memorandum; Health Physics Site Evacuation/Re-Location Drill Results; dated July 7, 2003
Internal Memorandum; Health Physics Core Damage Assessment Drill Results; dated December 10, 2003
Internal Memorandum; June 2004 Health Physics Drill Findings and Observations Report; dated July 9, 2004
Internal Memorandum; September 2003 Assembly/Accountability Drill Results; dated September 2003
Internal Memorandum; Quad Cities Nuclear Power Station 2003 Medical Drill Findings and Observation Report; dated September 17, 2003
Internal Memorandum; Quad Cities Nuclear Power Station 2004 Medical and Health Physics Drill Findings and Observation Report; dated October 5, 2004
Internal Memorandum; Four Mini-Drills Findings and Observations Report; dated May 6, 2003
Internal Memorandum; January 13, 2004 Mini-Drill Findings and Observation Report; dated January 14, 2004
Internal Memorandum; January 20, 2004 Mini-Drill Findings and Observation Report; dated January 21, 2004
Internal Memorandum; January 29, 2004 Mini-Drill Findings and Observation Report; dated January 29, 2004
Internal Memorandum; February 3, 2004 Mini-Drill Findings and Observation Report; dated February 6, 2004
Internal Memorandum; April 23, 2004 Mini-Drill Findings and Observation Report; dated April 27, 2004
Internal Memorandum; April 30, 2004 Mini-Drill Findings and Observation Report; dated May 10, 2004
Internal Memorandum; 2004 Practice Exercise Findings and Observation Report; dated June 2004
Internal Memorandum; 2004 Biennial Exercise Findings and Observation Report; dated July 2004
Internal Memorandum; Three Mini-Drills Findings and Observations Report; dated October 5, 2004
Internal Memorandum; September 2004 Annual Offsite Agency Dinner and Meeting for Quad Cities Nuclear Power Station; dated December 27, 2004
EP-AA-112-400-F-26; Emergency Operations Center Liaison Checklist; Revision A
EP-AA-112-400-F-32; List of Commonly Asked Questions at Emergency Operations Centers; Revision A
EP-AA-112-100-F-01; Shift Emergency Director Checklist; Revision C
EP-AA-114; Notifications; Revision 6
EP-MW-113-100; Assembly, Evacuation, and Accountability; Revision 1
EP-MW-113-100-F-02; Relocation Center Operations Checklist; Revision A
EP Information Newsletter; Lessons Learned from Actual Alert Declaration; dated August 4, 2003

Station Newsletter; Lessons Learned from Third Quarter 2004 EP Drills; dated November 2004
NOS Corporate Comparative Audit Report 2003 Emergency Preparedness, 50.54(t), and Meteorology; dated September 16, 2003
NOS Corporate Comparative Audit Report 2004 - Emergency Preparedness, 50.54(t); dated June 4, 2004
NOS Objective Evidence Report 212019-23; Equipment and Facility Readiness; dated May 10, 2004
Two NOS Objective Evidence Reports Associated with the June 2004 Practice Exercise
Five NOS Objective Evidence Reports Associated with the July 2004 Biennial Exercise
NOS Objective Evidence Report 253897-34; Station Blackout Tabletop; dated November 23, 2004
NOS Objective Evidence Report 278763-12; Followup on Station Blackout Tabletop Action Items; dated January 27, 2005
Root Cause Investigation Report; Emergency Plan Radiation Protection On-Shift Requirement Not Met Due to Lapsed Radiation Protection Qualifications of Chemistry Technicians; dated December 3, 2004
CR 159192; NRC Not Initially Notified Until 48 Minutes After Actual Alert Declaration
CR 159204; ERO Did Not Anticipate or Understand Iowa Officials' Automatic Closure of Public Parks and "State of Emergency" Declaration During Actual Alert Declaration
CR 167095; Improve Site Evacuation Map and Instructions to Evacuees
CR 167097; Reassess Adequacy of Equipment at Offsite Relocation Center
CR 167101; Revise Procedure EP-MW-113-100 to Address Lessons Learned from Site Evacuation Drill
CR 167189; Reassess Supplies for Security Personnel at Relocation Center
CR 216083; Posting of Emergency Information at EPZ Public Use Areas
CR 216199; Unauthorized Heater Equipment Change in Meteorological Instrument Building
CR 257721; Assign Two Radiation Protection Technicians to Every Shift Versus Counting a Chemistry Technician as the Second On-Shift Radiation Protection Technician on Back Shifts
CR 261267; TSC Staff Performance Concerns During July 2004 Exercise
CR 261271; OSC Staff Performance Concerns During July 2004 Exercise
AR 5040EP; Corporate EP Staff Revise ERO Training Procedure to Increase Level of Detail on ERO Training of Radiation Protection Versus Chemistry Technicians
AR 8007EP; Corporate EP Staff Assess Process to Determine Qualification Requirements of ERO Members Whose ERO Training Includes Accredited Training Performed by Other Departments
AR 5040EP; Perform Effectiveness Review in Late 2005 on Actions Implemented for Condition Report 257721

1EP6 Drill Performance

Issue Report 311365; Lessons Learned from March 4 Emergency Preparedness Drill; dated March 11, 2005
Issue Report 311364; Procedure Quality Issues from March 4 Emergency Preparedness Drill; dated March 11, 2005

Issue Report 311361; Facilities/Equipment Issues from March 4 Emergency Preparedness Drill; dated March 11, 2005
Issue Report 311357; Exercise Management/Scenario Issues from March 4 Emergency Preparedness Drill; dated March 11, 2005

4OA1 Performance Indicator Verification

EP-AA-125-1002; ERO Performance; Revision 3
EP-AA-125-1003; ERO Readiness; Revision 4
EP-AA-125-1004; Emergency Response Facilities and Equipment; Revision 3
Quad Cities Nuclear Power Station Off-Site Siren Test Plan; Revision 3; dated January 2002
Siren Daily Siren Reports; July 1, 2004, through December 31, 2004
Monthly Siren Operability Reports; July 2004 through December 2004
LS-AA-2110; Monthly Data Elements for NRC ERO Drill Participation; July 2004 through December 2004; Revision 6
LS-AA-2120; Monthly Data Elements for NRC Drill and Exercise Performance; July 2004 through December 2004; Revision 4
LS-AA-2130; Monthly Data Elements for NRC ANS Reliability; July 2004 through December 2004; Revision 4

4OA3

IR 317820, Bus 18 480 V Feed Breaker Trip; dated March 27, 2005
Prompt Investigation Report for IR 317820; dated March 27, 2005
QOA 6700-04, 480 V Bus 18 (28) Failure, Revision 18

4OA7 Licensee-Identified Violations

Root Cause Report 181083, Improper Verification Practices; dated October 15, 2003

The email document provided below was sent from an NRC regional inspector to the licensee during the inspection period. This document is included as part of NRC Inspection Report 05000254/2005002; 05000265/2005002 as required by NRC Inspection Manual Chapter 0620, “Inspection Documents and Records.”

Email from: NRC regional inspector
To: Exelon Quad Cities Regulatory Assurance and Design Engineering Individuals
Date: January 12, 2005

The following are the few questions/comments that I have on the two packages I reviewed:

EC 351277

1. Why is this package classified as non-safety-related? The Design Considerations Summary (DCS) section 4.1.4.2 states that the steel siding above Reactor Building EL 690'-5" is classified as non-safety-related. What is the source of this information? Isn't the siding part of the secondary containment boundary?

2. Attachment 9, page 2 of 2 (bottom of page) "training impact review" is BLANK. Why is Ops training not affected by the revision of procedures QCOS 1600-34 & QCOA - 0010 - 10?

3. DCS page 4 of 10, last paragraph "Tornado Differential Pressure Drop" states that, the tornado load design parameters include a 3 psi differential pressure drop. Although the items installed under this TCCP will be designed for a 300 mph wind, they will not be designed for the 3 psi maximum pressure drop as defined in UFSAR section 3.3.2. Why is this OK?

4. Page 2 of 5, Evaluation of the Impact of the Steam Dryer Replacement Project on Post Loca Doses, last paragraph of 'reasons for evaluation' states, however, since the available dose margin for control room thyroid dose....................especially for the unit 2 steam dryer replacement. Why is it different for unit 2?

EC351171

1. Page 3 of 4 mentions two special procedure requirements. Are these procedures prepared, reviewed and issued? If so, I would like to see a copy.

2. Page 4 of 4 states that, 'a cross reference will be provided in PASSPORT, which will prevent removal of EC 351171 until EC 351170 is completed, tested, and closed'. Is this activity completed? If so, I would like to see a printed copy of that.

Please note that, in addition to the above comments/questions, I had requested to see several procedures and calculations earlier. I hope to review those procedures and calculations while I am out at the site next week.

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AR	Action Request
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
EP	Emergency Preparedness
EPZ	Emergency Planning Zone
ERO	Emergency Response Organization
IP	Inspection Procedure
NCV	Non-Cited Violation
NOS	Nuclear Oversight
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide
PAM	Post Accident Monitoring Instrumentation
PARS	Public Availability Records
PI	Performance Indicator
RP	Radiation Protection