May 1, 2000

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: QUAD CITIES INSPECTION REPORT 50-254/200003(DRP); 50-265/200003(DRP) INCLUDING SUPPLEMENTAL INSPECTION

Dear Mr. Kingsley:

On April 4, 2000, the NRC completed an inspection at your Quad Cities Units 1 and 2 reactor facilities. The results were discussed with Mr. Dimmette and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of 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. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on resident inspection activities, and a supplemental inspection to review causes and corrective actions for the Unit 2 reactor core isolation cooling system exceeding a Performance Indicator threshold for system unavailability.

Based on the results of this inspection, NRC identified two issues which were categorized as being of very low risk significance and a third issue for which the risk was not yet categorized. The two very low risk issues involved a failure of the Unit 1 high pressure coolant injection system and a problem where Technical Specification requirements for control room ventilation flows were not met. These issues have been entered into your corrective action program. The control room ventilation issue involved a non-cited violation of regulatory requirements. The unresolved item involved the battery systems and has not yet been evaluated for risk. All three issues are listed in the summary of findings and are discussed in the report.

If you contest the violation or the severity level of the 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-001, with a copy to the Regional Administrator, Region III, Resident Inspector and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, namely >http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,
Original signed by
Mark A. Ring, Chief

Mark A. Ring, Chief
Reactor Projects Branch 1

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

Enclosure: Inspection Report 50-254/200003(DRP);
50-265/200003(DRP)

cc w/encl: D. Helwig, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President, Nuclear Operations
H. Stanley, Vice President, Nuclear Operations
R. Krich, Vice President, Regulatory Services
DCD - Licensing
J. Dimmette, Jr., Site Vice President
G. Barnes, Quad Cities Station Manager
C. Peterson, Regulatory Affairs Manager
M. Aguilar, Assistant Attorney General
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
W. Leech, Manager of Nuclear
MidAmerican Energy Company

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/200003(DRP); 50-265/200003(DRP)
Licensee:	Commonwealth Edison Company (ComEd)
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	March 1 through April 4, 2000
Inspectors:	C. Miller, Senior Resident Inspector K. Walton, Resident Inspector L. Collins, Resident Inspector R. Ganser, Illinois Department of Nuclear Safety
Approved by:	Mark Ring, Chief Reactor Projects Branch 1 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Quad Cities Nuclear Power Station, Units 1 & 2
NRC Inspection Report 50-254/200003(DRP); 50-265/200003(DRP)

The report covers a 6-week period of resident inspection.

Mitigating Systems

- GREEN. During logic testing on March 21, 2000, the Unit 1 high pressure coolant injection auxiliary oil pump failed to properly operate. This condition rendered the system inoperable for automatic initiation from approximately April 30, 1999, until March 21, 2000. Inspectors determined that sufficient information was not available in the operability evaluation to conclude that the system was operable following the March 21 maintenance. After reviewing supplemental information gathered and provided by the licensee upon inspector request, the inspectors determined that adequate justification was available to consider the system operable.

  Inspectors used Phase 2 of the Significance Determination Process for internal events, and found the risk of high pressure coolant injection being inoperable for automatic starts for approximately 1 year was very low. However the risk due to external events, specifically fires, was still being evaluated at the end of the period (Section 1R15).

- TBD. On March 16, 2000, the inspectors determined that operator actions were required during a design basis accident to maintain operability of the 125 Vdc and 250 Vdc battery systems. These operator actions were not referenced by emergency operating or annunciator response procedures and were not described in the Updated Final Safety Analysis Report. At the end of the inspection period, the inspectors had not yet determined whether these manual actions were part of the design and licensing basis of the battery system.

  The risk significance of this issue has not yet been determined pending further resolution of outstanding issues on the design and licensing basis of the battery systems. This issue is an unresolved item (Section 1R16).

Event Follow-up

- GREEN. On September 10, 1999, the licensee identified that the control room emergency ventilation system was inoperable. Flow rates were out-of-specification due to repositioning of a ventilation damper 9 days previously. The action statement for Technical Specification 3.8.D allowed the system to be inoperable for 7 days. Failing to comply with the allowed outage time requirements was considered a non-cited violation of Technical Specification 3.8.D.

  This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review (Section 40A3).

Report Details

1. REACTOR SAFETY

Plant Status

Both units were operated at or near full power during the period, with short duration power changes for surveillance testing and/or control rod pattern adjustment.

1R03 Emergent Work

a. Inspection Scope (71111-03)

The inspectors reviewed the licensee's control of maintenance activities and the risk evaluation for the unplanned system inoperability when the Unit 2 high pressure coolant injection 2-2301-6 valve showed indications of breaker degradation.

b. Observations and Findings

There were no observations or findings associated with this inspection activity.

1R04 Equipment Alignment

a. Inspection Scope (71111-04)

The inspectors performed a semi-annual equipment alignment walkdown of the Unit 1 and Unit 2 Station Blackout Diesel Generators. The inspectors reviewed outstanding work requests, station procedures, and NRC Information Notice 97-21, "Availability of Alternate AC Power Source Designed For Station Blackout Event." Specifically, the following procedures were reviewed to verify the standby readiness of the station blackout diesel generators:

QCOA 6100-03	"Loss of Offsite Power,"
QCOA 6100-04	"Station Blackout,"
QCOA 6900-18	"Loss of SBO 125 VDC Stationary Battery Charger 6(7)A,"
QCOP 6629-01	"SBO DG 1(2) Starting Air System Operation,"
QCOP 6620-02	"SBO DG 1(2) Jacket Water System Operation,"
QCOP 6620-03	"SBO DG 1(2) Fuel Oil System Operation,"
QCOP 6620-04	"SBO DG 1(2) Lube Oil System Operation," and
QCOP 6620-05	"SBO DG 1(2) Preparation For Standby Readiness."

b. Observations and Findings

The inspectors noted that a caution statement in the loss of offsite power procedure guided operators to start the station blackout diesel generators within 1 hour after a loss of offsite power event. The intent of the caution statement was to provide power to the station blackout diesel generator battery chargers in order to maintain the availability of the alternate AC power source. However, actual procedure steps did not require these actions to be taken. Engineering calculations on the station blackout diesel generator battery capacity showed that the batteries would likely retain sufficient capacity to start the station blackout diesel generators for many hours, much longer than the 1 hour period described by the caution in the procedure. The caution statement could have led operators to take actions that were not necessary in a complicated transient. Since the battery capacity was robust, this procedure accuracy problem did not render the procedure inadequate or affect the station blackout event. The inspectors provided this information to the licensee for use during the next update of the loss of offsite power procedure.

1R05 Fire Protection

a. Inspection Scope (71111-05)

The inspectors attended a briefing regarding areas in the plant considered to have high risk significance and how these areas were addressed in the Individual Plant Examination for External Events. The inspectors then performed a plant walkdown to look for transient combustibles, as well as detection and suppression system problems in the auxiliary electric room, cable spreading room and feed pump rooms. On a separate occasion, the inspectors performed a walkdown of the diesel-driven fire pumps and carbon dioxide fire suppression system for the emergency diesel generators

b. Observations and Findings

There were no observations or findings associated with this inspection activity.

1R09 Inservice Testing

a. Inspection Scope (71111-09)

The inspectors reviewed Problem Identification Form Q2000-00609 and other problems found with motor operated valve operators to determine licensee corrective actions for these failures.

b. Observations and Findings

There were no observations or findings associated with this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope (71111-11)

The inspectors observed simulator training of Operating Crew F on March 8, 2000. Scenarios observed included a toxic gas analyzer failure, loss of feedwater heaters, and main steam leak inside the drywell. Inspectors observed communications, procedure adherence, and implementation of emergency operating procedures. In addition, event classification and reporting actions were observed.

b. Observations and Findings

No observations or findings were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111-12)

The inspectors reviewed how the licensee dispositioned the following problem identification forms under the maintenance rule:

Q1999-00404	"Unit 1 Reactor Core Isolation Cooling Overspeed Test Aborted,"
Q1999-00427	"Governor Control Problems During Overspeed Testing,"
Q1999-02780	"Unit 2 Reactor Core Isolation Cooling Quarterly Surveillance,"
Q1999-02800	"Potential Water Hammer in Unit 2 Reactor Core Isolation Cooling Piping,"
Q2000-00269	"2-1001-47 not prepared for operation,"
Q2000-00413	"Rainbow Fire Pump Fuel Oil Leak,"
Q2000-00276	"Valve 2-0220-45 failed to close on loss of power,"
Q1999-03787	"2C RHRSW pump room cooler leak,"
Q2000-00127	"Unit 1 APRM 5 Flow bias nonconservative,"
Q2000-00247	"U1A Rod Worth Minimizer Failure,"
Q2000-00297	"ADS Valves Taken OOS in Mode 3," and
Q2000-00443	"2B RHR Room Cooler."

b. Observations and Findings

There were no findings identified during this inspection activity.

1R13 Maintenance Work Prioritization and Control

a. Inspection Scope (71111-13)

The inspectors reviewed the risk evaluation for maintenance work executed the week of March 20.

b. Observations and Findings

There were no observations or findings during this inspection.

1R15 Operability Evaluations

a. Inspection Scope (71111-15)

The inspectors reviewed operability evaluations associated with the following problem identification forms:

Q1999-02341	"Valves Calculated Structural Limits Potentially Exceeded,"
Q1999-02771	"Standby Gas Treatment Train Start Time Out-of-Tolerance,"

Q2000-01025	“Body/Bonnet Leak on Air Operated Valve 2-2001-15,” and
Q2000-01214	“Unit 1 High Pressure Coolant Injection Test Failure During Testing.”

b. Observations and Findings

During logic testing on March 21, 2000, the Unit 1 high pressure coolant injection auxiliary oil pump failed to properly operate. This condition rendered the system inoperable for automatic initiation from approximately April 30, 1999, until March 21, 2000. Inspectors used Phase 2 of the Significance Determination Process, and determined the risk from internal events of high pressure coolant injection being inoperable for automatic starts for approximately 1 year was very low (GREEN). Risk due to external events, specifically fires, was still being assessed at the end of the period. Inspectors reviewed the operability evaluation following the repair of the high pressure coolant injection oil system, and determined that sufficient information was not available to conclude that the system was operable. After reviewing supplemental information gathered and provided by the licensee upon inspector request, the inspectors determined that adequate justification was available to consider the system operable.

In part of the logic testing, the auxiliary oil pump was found to cycle on and off continuously for a period of about 10 minutes until an operator took manual control of the pump. This condition rendered the high pressure coolant injection system inoperable because hydraulic oil pressure was not sufficient to operate the turbine valves. Technicians lowered the setpoint of oil pressure Regulating Valve 3, which regulated system oil pressure, and monitored system parameters to determine the cause of the problem. Licensee engineers determined that a setpoint conflict between pressure Regulating Valve 3 and Pressure Switch 4, which automatically started and stopped the auxiliary oil pump, caused the pump to cycle on and off. This prevented sufficient buildup of oil pressure to operate the turbine valves. Interactions of the high pressure coolant injection emergency oil pump caused an increased system pressure, which also affected the pressure at Pressure Switch 4. After the setpoint change, the licensee concluded that subsequent testing showed that the auxiliary oil pump remained running long enough to render normal high pressure coolant injection system hydraulic parameters. Engineering Operational Problem Response 00-01-2300-002 dated March 22, 2000, written to discuss the technical aspects of the problem, indicated the high pressure coolant injection system was operable after the setpoint change and subsequent testing. Operators declared the high pressure coolant injection system operable based on the engineering operational problem response.

Inspectors found the following problems with the engineering operational problem response:

- The engineering operational problem response indicated the pressure regulating valve setpoint had drifted, but no information was available to tell whether or not the drift trend was stopped.
- Data for the tolerance band for setpoints of Pressure Switch 4 and Pressure Regulating Valve 3 and potential overlap of the setpoints was not discussed.
- Acceptable setpoint settings for the system when the emergency oil pump was running were not provided.

- When high pressure coolant injection was started, operators saw some indications of Pressure Switch 4 shutting off the auxiliary oil pump. This interaction occurred once or twice during the high pressure coolant injection run, but did not prevent system startup. However, the reason for this interaction, and the potential effects on system operability were not discussed in the engineering operational problem response. Engineers planned the next test of the high pressure coolant injection auxiliary oil pump for 3 months after the problem was identified, even though there was some question as to the drift characteristics of the pressure regulating valve and to potential adverse interactions of Pressure Switch 4.

- Inspectors estimated, based on direct observation, that the number of times the auxiliary oil pump cycled on and off was between 50 and 100. Communication between operators conducting the test and engineers appeared poor in that the system engineer was unaware of the high number of cycles which occurred. The engineering operational problem response indicated the auxiliary oil pump cycled only a few times. In addition, station management was not aware of the cycling which occurred during the failed test. As a result, potential effects on the 250 Volt battery during accident conditions and potential adverse affects on the motor had not been evaluated.

Inspectors reviewed corrective actions from a similar problem with a high pressure coolant injection auxiliary oil pump cycling in 1996. Licensee Event Report 50-265/00005 documented the Unit 2 problem and the corrective actions. Better trending was listed as a proposed corrective action. The inspectors concluded that better trend data taking was performed, but the trend was not analyzed sufficiently to detect and prevent this problem. Engineers did not recognize that the increase in Pressure Regulating Valve 3 setpoint in April 1999 might have jeopardized operability of the system. Inspectors noted that without a determination of acceptable setpoints for the system controls, including the acceptable setpoints with the emergency oil pump in operation, analysis of the trend would have been difficult.

The extent of condition was not known at the end of the period. Inspectors inquired into potential similar conditions for Unit 2 and for the Dresden station. Engineers were finishing a root cause report at the end of the period, and the station was in the process of writing a licensee event report describing the problem. A plan for installing a modification to seal in the auxiliary oil pump once it started was proposed (Design Change Packages 9900080 and 9900079), but dates for the modification installation were not available at the end of the period. Inspectors planned further review, including assessment of corrective actions, during closure of the pending licensee event report.

1R16 Operator Work-Arounds

.1 Safety-Related Batteries

a. Inspection Scope (71111-16)

The inspectors reviewed operator actions required by Quad Cities Operating Abnormal Procedure 6900-05, “Loss of Safety-Related 250 VDC Battery Chargers Concurrent With a Design Basis Accident.”

b. Observations and Findings

On about March 16 the inspectors identified that the abnormal operating procedure required operators to trip nonsafety-related equipment between 30 minutes and 2 hours after a loss of offsite power in order to ensure that the 250 Vdc battery system could perform its design function. The design function of the 250 Vdc battery system as described in the Technical Specification bases and the updated final safety analysis report, was to start and carry the normal direct current loads required for safe shutdown on one unit, and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours without power from the chargers.

The inspectors determined that these operator actions were not described in the Updated Final Safety Analysis Report and that the procedure requiring the actions was not referenced by the emergency operating procedures. Based on discussions with licensee staff, the inspectors determined that this procedure was not referenced by any other procedure and that there was limited operator knowledge of these required operator actions. This and other abnormal operating procedures were reviewed as part of operator required reading once per year. However, other training, such as simulator exercises that would require operators to recognize the need to perform the load stripping, was not conducted. As a result, the inspectors concluded that the likelihood of these operator actions being successful was very low.

The licensee confirmed that the operator actions to strip nonsafety-related 250 Vdc loads were required in order to match the assumptions in the battery sizing calculation. The licensee revised annunciator response procedures for the loss of the 250 Vdc battery chargers and the loss of offsite power procedure to incorporate a reference to Quad Cities Operating Abnormal Procedure 6900-05 which would direct the load stripping and preserve battery operability. The licensee also determined that the same condition existed for the 125 Vdc system and made similar procedure changes.

At the end of the inspection period the inspectors identified several unresolved issues associated with the manual actions required to maintain battery operability. These issues included:

- clarifying if operator actions to strip loads were incorporated as part of the original design and/or licensing basis of the plant and reviewed by the NRC,
- determining the extent of the load stripping actions required for the 125 Vdc battery system,
- understanding if the updated safety analysis report was properly maintained to accurately describe important, time-critical, operator actions required for systems to perform their required functions,
- determining if procedures were adequate to ensure the operability of the battery systems prior to the recent procedure changes,
- determining the risk significance of the design/procedure deficiencies as assessed by the Significance Determination Process.

The inspectors considered these issues to be an **Unresolved Item (50-254/200003-01; 50-265/200003-01)** pending further review by the licensee and the inspectors.

.2 General Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of operator work-arounds on the plant, and the potential adverse consequences on performing emergency or abnormal procedures. The inspectors also reviewed open and recently closed operator work-arounds and operator challenge lists and nuclear station operator turnover checklists. The inspectors also examined the quarterly operator burden review performed by the licensee.

b. Observations and Findings

The inspectors determined that no new operator work-arounds were added since the previous inspection that would significantly degrade the ability of operators to perform important safety functions. Electrical disconnection of the shutdown cooling suction valves and manual actions to be taken for control room ventilation were the primary work-arounds that may affect operators while performing emergency or abnormal procedures. These were previously evaluated as acceptable manual actions, and corrective actions to eliminate the work-arounds were scheduled.

1R19 Post Maintenance Testing

a. Inspection Scope (71111-19)

The inspectors observed portions of the Unit 2 Station Blackout Diesel Generator post maintenance test on February 24, 2000, and the Unit 1 emergency diesel generator post maintenance test (Quad Cities Mechanical Maintenance Surveillance 6600-05) on March 1, 2000.

b. Observations and Findings

There were no observations or findings documented during this inspection.

1R22 Surveillance Testing

a. Inspection Scope (71111-22)

The inspectors observed performance of the following Quad Cities Operating Surveillance (QCOS) tests:

QCOS 1000-01	“High Drywell Pressure Scram Functional and Calibration Test,”
QCOS 2300-29	“HPCI System Logic Functional Test,” and
QCOS 1300-23	“RCIC system Logic Functional Test.”

b. Observations and Findings

There were no findings identified during this inspection.

Emergency Preparedness (EP)

1EP1 Drill, Exercise, and Actual Events

a. Inspection Scope (71114-01)

The inspectors observed simulator training of Operating Crew F on March 8, 2000. Three scenarios were observed including a toxic gas analyzer failure, loss of feedwater heaters, and main steam leak inside the drywell (Section 1R11). Inspectors observed communications, procedure adherence and implementation of emergency operating procedures. In addition, event classification and reporting actions were observed.

b. Observations and Findings

No observations or findings were identified. One opportunity to declare an Alert was performed successfully and one opportunity to report an Alert was performed successfully.

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up

a. Inspection Scope (71153)

The inspectors reviewed licensee event reports and other items using Inspection Procedure 71153.

b. Observations and Findings

(Closed) Violation 50-254/98004-04; 50-265/98004-04: Missed Technical Specification Surveillance. The licensee failed to test source range instruments as required by Technical Specifications due to inadequate implementation of the Technical Specification Upgrade Program. There was no risk significance to this event since this condition was identified with Unit 2 shut down and all control rods inserted. Subsequent testing of the source range monitors was successfully completed. The licensee implemented corrective actions as a result of this event. This item is closed.

(Closed) Licensee Event Report 50-254/99003-00 and 50-254/99003-01: High Pressure Coolant Injection System Inoperable due to Manual Closure of Steam Supply Valve. On two separate occasions, the licensee identified that the outboard (containment isolation) steam supply valve to the Unit 1 high pressure coolant injection turbine failed to close upon demand. The licensee then closed the inboard isolation valve to comply with Technical Specifications. The licensee identified inadequate instructions in the maintenance procedure for the electrical breaker as the root cause of this event. The inspectors determined this was a non-cited violation of the maintenance rule as discussed in Section 1R12 of Inspection Report 50-254/99020(DRP); 50-265/99020(DRP). The risk significance of this issue was minimal since the inboard containment isolation valve was operable and was able to fulfill the containment isolation function. This issue is closed.

(Closed) Licensee Event Report 50-265/99003-00; Reactor Core Isolation Cooling Overspeed Trip due to a Failed Governor Control Power Resistor. This event was documented in Inspection Report 50-254/99018(DRP); 50-265/99018(DRP) and in Section 40A4 of this report. The risk significance for this event was documented in

Inspection Report 50-254/99018(DRP); 50-265/99018(DRP) as being GREEN. This item is closed.

(Closed) Licensee Event Report 50-254/99004-00 and 50-254/99004-01: Control Room Emergency Ventilation System Inoperable due to Airflow Rate in Excess of Technical Specifications. On September 10, 1999, the licensee identified that the system flow rates were out of specification high due to repositioning of a ventilation damper 9 days previously. Since the flow rate did not meet Technical Specification surveillance requirements, the licensee declared the control room emergency ventilation system inoperable. The reportability aspects of this event were documented in Inspection Report 50-254/99020(DRP); 50-265/99020(DRP). The action statement for Technical Specification 3.8.D allowed the system to be inoperable for 7 days. Failing to comply with the allowed outage time requirements was considered a violation of Technical Specification 3.8.D. This violation is being treated as a **Non-Cited Violation (50-254/200003-02; 50-265/200003-02)** consistent with the Interim Enforcement Policy for pilot plants. This violation is in the licensee's corrective action system as Problem Identification Form Q1999-02987. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This item is closed.

(Closed) Unresolved Item 50-265/99020-05: Fault Exposure Hours not Included for Unit 2 Reactor Core Isolation Cooling Failure. The licensee revised the reporting of the safety system unavailability - heat removal systems, to included 987.3 fault exposure hours in the November 1999 submittal to the NRC. This item is closed.

(Closed) Licensee Event Report 50-254/00001-00: Inadvertent Start of the Shared Emergency Diesel Generator. The shared emergency diesel generator inadvertently started during logic testing due to a personnel error. The diesel appropriately started when the relays were bumped by personnel. There was no risk significance to this event. The licensee implemented corrective actions for this event. This item is closed.

4OA4 Other

1. Supplemental Inspection - White Performance Indicator - Safety System Unavailability - Heat Removal System

a. Inspection Scope (95001)

The inspectors utilized Inspection Procedure 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," to review the licensee's root cause assessment and corrective actions associated with a failure of the Unit 2 reactor core isolation cooling system to operate in August 1999. The reactor core isolation cooling system was a risk significant system for boiling water reactors. This system was monitored by the Safety System Unavailability Performance Indicator for heat removal systems. The failure of this system to operate when commanded resulted in the Unit 2 heat removal performance indicator turning WHITE (Regulatory Response Band).

The purpose of this inspection was to provide assurance that the root cause was understood by the licensee; that the extent of condition was identified; and that the licensee's corrective actions were sufficient to address the root cause to prevent recurrence of a similar failure.

2. Evaluation of Inspection Requirements

2.1 Problem Identification

a. **Determine that the evaluation identified who and under what conditions the issue was identified.**

The inspectors determined that the licensee's root cause evaluation appropriately documented that the licensee identified the failure of the Unit 2 reactor core isolation cooling pump to start during testing.

On August 25, 1999, the licensee completed maintenance on the Unit 2 reactor core isolation cooling pump, and returned the pump to service. The operators started the pump for a quarterly surveillance test and noted that the pump tripped on an overspeed condition. Subsequent troubleshooting identified that a failure of a resistor in the turbine governor control circuit resulted in a failure of the turbine governor valve to control discharge flow.

b. **Determine that the evaluation documented how long the issue existed, and prior opportunities for identification.**

The last successful operation of the Unit 2 reactor core isolation cooling system was on June 2, 1999. On August 24, 1999, the licensee removed a set of fuses to the turbine governor control circuit to allow replacement of two power supplies. During the subsequent surveillance test, after the return to service, operators detected the system did not operate properly. The licensee believed, and documented in the root cause evaluation, that the reinstallation of the power supply fuses to the turbine governor control circuit may have produced a current surge which resulted in the failure of the resistor.

The licensee successfully checked the operation of the resistor every refuel outage. The resistor functioned properly during turbine governor ramp generator and signal converter testing in April 1997.

Operation of this circuit was not annunciated. A failure of this resistor would produce a zero reading on both local and control room indication of turbine speed. However, since the reactor core isolation cooling system is a standby system, turbine speed typically indicated zero.

The inspectors concluded that the licensee's evaluation documented prior opportunities for identification of the problem. However, it was not known with certainty when the actual failure of the resistor occurred.

c. **Determine that the evaluation documented the plant specific risk consequences and compliance concerns associated with the issue.**

The inspectors determined that the licensee's root cause evaluation adequately addressed compliance issues, but the evaluation did not address the risk significance of removing the system from service. However, the licensee evaluated risk to the facility for systems being unavailable and documented this on Work Week Safety Profile Sheets. The licensee removed Unit 2 reactor core isolation cooling system from service for planned maintenance during the week of August 23, 1999. The licensee determined that the increased risk of removing this system from service was 2.8 times nominal risk. Other risk significant equipment on Unit 2 was available during this time. The licensee deferred testing of the Unit 2 emergency diesel generator until the limiting condition for operation for the reactor core isolation cooling system was exited. The inspectors analyzed the loss of the reactor core isolation cooling system using the Significance Determination Process. The issue was documented in Inspection Report 50-254/99018(DRP); 50-265/99018(DRP) as a GREEN finding with low risk significance.

The licensee's evaluation documented that the licensee remained in a limiting condition for operation for troubleshooting the Unit 2 reactor core isolation cooling system after the failure was identified. Similarly, the licensee's evaluation documented that a 4 hour emergency notification call to the NRC was completed. The evaluation also included that this condition was documented in the licensee's corrective action program as Problem Identification Form Q1999-02780.

The licensee accumulated about 24 hours of planned safety system unavailability for the maintenance activities and accumulated an additional 29.5 hours of unplanned safety system unavailability for troubleshooting/repair activities. The inspectors identified that the licensee evaluated this component failure as being a maintenance rule functional failure and a maintenance preventable functional failure. Previously, the reactor core isolation cooling system was classified as an a(1) system under the maintenance rule, since reliability and availability goals were not met.

The inspectors determined that the licensee included both the planned and unplanned unavailable hours into the safety system unavailability performance indicator as reported to the NRC. However, the licensee did not initially report fault exposure hours since the licensee believed there was reasonable assurance that the resistor failed during the reinstallation of power supply fuses to the turbine governor circuit.

2.2 Root Cause and Extent of Condition Evaluation

a. **Determine that the problem was evaluated using a systematic method to identify root causes and contributing causes.**

The inspectors determined that the licensee used a failure mode/cause tree analysis as described in Procedure NSP-WC-3010, "Troubleshooting." The licensee generated a list of each potential cause of failure. Each item was tested to confirm expected results until the licensee identified a failure of the resistor to the turbine governor ramp generator and signal converter circuit. For follow up actions, the licensee sent the failed resistor to a laboratory for a failure analysis and documented the findings in the root cause evaluation. The resistors failure was attributed to a combination of aging degradation and a power surge.

b. **Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem.**

ComEd corporate office provided guidance for root cause evaluations in its Root Cause Investigation and Report Handbook. The root cause evaluation for this event was structured in accordance with the handbook guidance. The inspectors verified that the level of detail was commensurate with the significance of the problem.

c. **Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.**

The inspectors confirmed that the licensee's evaluation followed the licensee's Root Cause Investigation and Report Handbook for review of previous events and operating experiences. There were no site-specific events or industry events relevant to this component failure. However, the licensee included a corrective action to review Information Notice 90-51, regarding resistor failures on emergency diesel generators.

d. **Determine that the root cause evaluation included consideration of potential common causes and extent of condition of the problem.**

The inspectors determined that the root cause evaluation did not explicitly document potential common causes or extent of condition. The licensee's Root Cause Investigation and Report Handbook required evaluation of similar components on the other unit and similar situations to determine the extent of the unacceptable condition. The handbook did not define extent of condition. However, in the corrective actions section of the report, the licensee addressed replacement of the same resistor in the Unit 1 reactor core isolation cooling system and assigned a corrective action to review periodic replacement of power supply resistors in other safety systems.

2.3 Corrective Actions

a. **Determine that appropriate corrective actions were specified for each root and/or contributing cause or that there was an evaluation that no actions were necessary.**

The inspectors determined that the corrective actions appeared to adequately address the root cause. The licensee's root cause report identified the reason that the Unit 2 reactor core isolation cooling pump failed to operate on August 25, 1999, was a failed resistor in the turbine governor circuit. The licensee attributed the resistor failure to a combination of aging and a power surge during the return to service. The licensee's corrective actions to prevent recurrence were to replace both Unit 1 and Unit 2 reactor core isolation cooling system turbine governor circuit resistors. Also, the licensee developed a preventive maintenance task to replace the resistors at a 10-year frequency.

b. **Determine that the corrective actions have been prioritized with consideration of the risk significance and regulatory compliance.**

The inspectors determined that the corrective actions were appropriately prioritized. The licensee replaced the Unit 2 reactor core isolation cooling turbine governor circuit resistor after the failure was identified. Replacement of the Unit 1 resistor was completed in October 1999. The licensee also issued a predefine activity to inspect these resistors at a 10-year frequency. Other corrective actions had dates assigned that were appropriate to the risk.

c. **Determine that a schedule has been established for implementing and completing the corrective actions.**

The inspectors verified that the licensee's corrective action program adequately assigned individuals, actions, and completion dates for corrective actions not completed.

d. **Determine that quantitative or qualitative measures of success have been developed for determining the effectiveness of the corrective actions to prevent recurrence.**

The inspectors verified that the licensee scheduled effectiveness reviews for the corrective actions implemented for this issue. These reviews were scheduled to be completed in the year 2001 and as such, were not completed for this inspection.

4OA4 Management Meetings

The inspectors presented the inspection results to Mr. Dimmette and other members of licensee management at the conclusion of the inspection on April 4, 2000. The licensee acknowledged the findings presented. This exit meeting also served as the performance meeting for the White performance indicator discussed in the supplemental inspection portion of this report (4OA4). No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

J. Dimmette	Site Vice President
G. Barnes	Station Manager
G. Boerschig	Engineering Director
M. McDowell	Operations Manager
M. Perito	Maintenance Manager
R. Chrzanowski	Nuclear Oversight Manager

<u>NRC</u>

M. Ring	Branch Chief, Division of Reactor Projects

<u>Illinois Department of Nuclear Safety</u>

Bob Ganser	Resident Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-254/200003-01; 50-265/200003-01	URI	Safety-Related Batteries
50-254/200003-02; 50-265/200003-02	NCV	Control Room Emergency Ventilation System Inoperable due to Airflow in Excess of Technical Specifications

Closed

50-254/200003-02; 50-265/200003-02	NCV	Control Room Emergency Ventilation System Inoperable due to Airflow in Excess of Technical Specifications
50-254/98004-04; 50-265/98004-04	VIO	Missed Technical Specification Surveillance
50-254/99003-00	LER	High Pressure Coolant Injection System Inoperable due to Manual Closure of Steam Supply Valve
50-254/99003-01	LER	High Pressure Coolant Injection System Inoperable due to Manual Closure of Steam Supply Valve
50-265/99003-00	LER	Reactor Core Isolation Cooling Overspeed Trip due to a Failed Governor Control Power Resistor
50-254/99004-00	LER	Control Room Emergency Ventilation System Inoperable due to a Airflow Rate in Excess of Technical Specifications
50-254/99004-01	LER	Control Room Emergency Ventilation System Inoperable due to a Airflow Rate in Excess of Technical Specifications
50-265/99020-05	URI	Fault Exposure Hours not Included for Unit 2 Reactor Core Isolation Cooling Failure
50-254/00001-00	LER	Inadvertent Start of the Shared Emergency Diesel Generator

LIST OF BASELINE INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure		Report
Number	Title	Section
71111-03	Emergent Work	1R03
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-09	Inservice Testing	1R09
71111-11	Licensed Operator Requalification	1R11
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Work Prioritization & Control	1R13
71111-14	Nonroutine Evolutions	1R14
71111-15	Operability Evaluations	1R15
71111-16	Operator Workarounds	1R16
71111-19	Post Maintenance Testing	1R19
71111-22	Surveillance Testing	1R22
71114-01	Drill, Exercise, and Actual Events	1EP1
71153	Event Follow-up	4OA3
(none)	Other	4OA4
(none)	Management Meetings	4OA5

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
APRM	Average Power Range Meter
CFR	Code of Federal Regulations
DG	Diesel Generator
DRP	Division of Reactor Projects
EOPR	Engineering Operational Problem Response
IDNS	Illinois Department of Nuclear Safety
IFI	Inspection Follow-up Item
LER	Licensee Event Report
OOS	Out-of-Service
PRV	Pressure Regulating Valve
QCOA	Quad Cities Operating Abnormal Procedures
QCOP	Quad Cities Operating Procedures
QCOS	Quad Cities Operating Surveillances
PERR	Public Electronic Reading Room
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SBO	Station Black Out
URI	Unresolved Item
VDC	Volts - Direct Current
VIO	Violation