June 14, 2000

EA-00-128

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 50-254/2000005(DRP); 50-265/2000005(DRP)

Dear Mr. Kingsley:

On May 16, 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.

This report discusses an issue of low to moderate safety significance. As described in Section 1R13 of this report, the Unit 2 safe shutdown makeup pump injection valve failed to operate on January 19, 2000, due to the failure of the yoke bushing. This issue was assessed using the applicable Significance Determination Process as a potentially safety significant finding that was preliminarily determined to be WHITE (i.e., an issue with some increased importance to safety, which may require additional NRC inspection). The issue has a low to moderate safety significance because the safe shutdown makeup pump may not have been available for injection to Unit 2 during some fire scenarios for which it was considered a mitigating system.

Although we believe that we have sufficient information to make our final significance determination for the issue, we are providing an opportunity for you to send us additional information including your position on the significance of the issue and the bases for your position. Also, please inform us if you would like to schedule a regulatory conference to discuss your evaluation and any differences with the NRC evaluation. No enforcement is presently being issued for this inspection finding. Please contact Mark Ring at 630-829-9703 within 10 days of the date of this letter to inform the NRC of your intentions. If we have not heard from you by telephone or in writing regarding a conference within 14 days, we will continue with our significance determination decision, and you will be advised by separate correspondence of the results of our deliberations on this matter.

Based on the results of this inspection, the NRC determined that one violation of NRC requirements occurred. Unqualified parts were installed in the Unit 1 average power range monitor system. Operators then authorized the return of this system with unqualified parts for use as an input to the reactor protection system. This issue has been entered into your corrective action program. This issue is also listed in the summary of findings and discussed in this report. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the Enforcement Policy. If you contest this NCV, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Quad Cities facility.

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

Sincerely,

/RA/
/s/Marc L. Dapas for

Geoffrey E. Grant, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-254/2000005(DRP);
50-265/2000005(DRP)

See Attached Distribution

DOCUMENT NAME: G:\QUAD\insp rpt 200005.wpd
To receive a copy of this document, indicate in the box:"C" = Copy without enclosure "**E**"= Copy with enclosure"**N**"= No copy

OFFICE	RIII	N	RIII	E	RIII	E	RIII	N
NAME	Lerch/trn		Ring		Clayton		Grant	
DATE	06/12/00		06/12/00		06/12/00		06/13/00	

OFFICIAL RECORD COPY

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

ADAMS Distribution:
AJM
WES
SNB (Project Mgr.)
R. Borchardt, OE
D. Dambly, OGC
V. Ordaz, NRR
T. Frye, NRR
A. Madison, NRR
S. Stein, NRR
R. Mathew, NRR
J. Dyer, RIII
J. Caldwell, RIII
B. Clayton, RIII
K. Lambert, RIII,
SRI Quad Cities
DRP
DRS
RIII PRR
RIII_IRTS
JRK1
BAH3

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/2000005(DRP); 50-265/2000005(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:	April 5 through May 16, 2000
Inspectors:	C. Miller, Senior Resident Inspector K. Walton, Resident Inspector L. Collins, Resident Inspector
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 50-254/2000005(DRP); 50-265/2000005(DRP)

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

Mitigating Systems

- TBD. The Unit 2 safe shutdown makeup pump injection valve failed to operate on January 19, 2000, due to the failure of the yoke bushing. This was determined to be an apparent risk significant issue for fire initiating events during a preliminary Phase 3 Significance Determination Process review (Section 1R13).

- GREEN. The licensee discovered that maintenance technicians had installed unqualified parts in the Unit 1 average power range monitoring system on January 12, 2000. Subsequent to the installation, operators placed the average power range monitor into service, as an input to the reactor protection system. This was considered a non-cited violation (NCV) of 10 CFR 50, Appendix B requirements (Section 1R15).

  The risk significance of this issue was determined to be very low (GREEN) because other average power range monitors could have provided redundant reactor protection system signals, if needed.

- NO COLOR. 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 for approximately one year. The risk from internal events for this condition was determined to be very low (GREEN) in inspection report 50-254/2000003; 50-265/2000003. The effect on risk due to external events, specifically fires, was determined to be potentially significant during a preliminary Significance Determination Process review. However, this issue was also the substantial contributor to a YELLOW high pressure coolant injection unavailability performance indicator, which represents performance that minimally reduces safety margin and requires NRC oversight. Therefore, the NRC considers the inspection finding and the performance indicator to be a single issue and agency action will be determined based on application of the Action Matrix for the YELLOW performance indicator (Section 1R22).

Report Details

1. REACTOR SAFETY

Plant Status (71150)

Operators maintained Unit 1 at or near full power operations during the period, except for minor power decreases for testing and control rod positioning. Operators maintained Unit 2 at or near full power operation until April 18, 2000, when Unit 2 operators detected a decreasing condenser vacuum which required reducing reactor power from 100 percent to about 50 percent power. The licensee identified a leakage path for air to the condenser through a loop seal which was not filled. Operators refilled the loop seal, and full power was achieved the following day. On May 5, 2000, an instrument technician error during a main steam high flow surveillance test resulted in the main steam isolation valves closing. This produced a Unit 2 reactor trip. Safety systems responded appropriately to the event. Operators returned Unit 2 to power operations on May 6 and to full power on May 7. Operators maintained Unit 2 at or near full power operations for the remainder of the period.

1R04 Equipment Alignment (71111-04)

b. Inspection Scope

The inspectors reviewed the following system alignments related to the Mitigating System Cornerstone for both units:

- 125 Volt battery
- 250 Volt battery
- reactor core isolation cooling

c. Issues and Findings

There were no issues or findings associated with this activity.

1R05 Fire Protection (71111-05)

a. Inspection Scope

The inspectors performed fire impairment walk-downs of the following areas related to the Mitigating System Cornerstone:

- 4160 Volt switchgear - both units
- Reactor feed pumps - both units
- Feedwater regulating valve - both units
- Auxiliary electric and cable spreading rooms
- Impairments related to transformer deluge

The inspectors reviewed fire protection impairments and compensatory actions taken for the removal from service of fire protection detection and deluge systems associated with transformers T2, T21, T22, and the removal from service of turbine bearing deluge systems on Unit 2. The inspectors also spoke with operations personnel and the site fire marshal.

b. Issues and Findings

The inspectors did not identify any issues or findings from this inspection activity.

1R06 Flood Protection (71111-06)

a. Inspection Scope

The inspectors toured the facility and reviewed the Updated Final Safety Analysis Report. The inspectors reviewed annunciator response and testing procedures used to verify proper operation of plant internal flooding detection and mitigation systems. The inspectors also reviewed corrective actions (Problem Identification Form Q1999-02092) for an inspector-identified external flood deficiency.

b. Issues and Findings

There were no issues or findings associated with this activity.

1R11 Licensed Operator Requalification (71111-11)

a. Inspection Scope

The inspectors observed simulator training of Operating Crew “E” on April 26, 2000, and Operating Crew “D” on May 3, 2000. The inspectors assessed communications, procedure adherence, and implementation of emergency operating procedures. In addition, event classification and reporting actions were also evaluated.

b. Issues and Findings

There were no issues or findings for this inspection activity.

1R12 Maintenance Rule (71111-12)

a. Inspection Scope

The inspectors reviewed licensee-established performance criteria for the residual heat removal and station blackout diesel generator systems. The inspectors also reviewed the following problem identification forms for proper maintenance rule classifications:

Q2000-00925	“Valve 2-1001-16A Would not Close During Surveillance Testing,”
Q2000-01522	“Unit 2 Station Blackout Diesel Failed to Start,”
Q2000-00102	“Unit 2 Number 3 Stop Valve Failed to Close When Tested,”
Q2000-01130	“Sticking Turbine Solenoid Valves,”

Q2000-01360	"Power Load Unbalance Regarding Failed Main Turbine Fast Acting Solenoids,"
Q2000-01491	"Immediate Modification to Increase Voltage to the Main Turbine Solenoid Valves,"
Q2000-01449	"Unit 2 Turbine Generator Solenoid Valve Failures,"
Q2000-01630	"Turbine Valve Stroking Failure During QCOS 5600-05," and
Q2000-01821	"Unit 2 Turbine CIVs Fail to Operate as Required on the First Attempt."

b. Issues and Findings

Based on Unit 2 turbine testing, the licensee identified approximately 36 failures related to solenoid valve operation since March 12, 2000. Some failures appeared to be related only to the testing portion of the turbine control system. However, some of the valves which have failed had a turbine overspeed protective function. The licensee identified low voltage at the solenoid valves as a potential contributor to the problem. The vendor recommended that the solenoid valves be operated at 120 Volts with a 10 percent tolerance band, or 108 Volts minimum. Some voltages measured at the Unit 2 turbine solenoid valves were as low as 103 Volts. Other potential contributors included inadequate solenoid valve replacement frequency.

The inspectors determined that these valves were not specifically modeled in the station Individual Plant Examination. Additionally, engineers informed the inspectors that the turbine overspeed protective function and failure of this function were not included in the risk model either. The inspectors requested additional information to determine the risk significance of the failures and whether the Unit 2 turbine solenoid valves should be modeled in the Individual Plant Examination since the valves had exhibited a significantly higher failure rate during recent testing. Licensee engineers were obtaining more risk information at the end of the inspection period. This is considered an **unresolved item (50-265/2000005-01)** pending further risk information from the licensee which will be used in the Significance Determination Process.

Fast acting solenoids may be required for turbine overspeed protection. Without that protection, there is an increased risk of turbine failure. An increased failure probability of the turbine has the potential to adversely affect turbine building equipment directly or increase the risk of turbine building fires (which have occurred in the industry due to turbine failures). Safe shutdown equipment such as safety-related switchgear are located on the turbine floor, and therefore could be affected by a turbine failure.

Although the valves were part of the electro-hydraulic control system, which is classified as a Maintenance Rule a.1 system for other reasons, the action plan for correcting the solenoid valve deficiencies had not yet been approved. The inspectors continued to work with licensee engineers at the end of the inspection period to establish the specific failure modes for the solenoid valves and the risk associated with those failures. At the end of the period, corrective actions to improve voltage at the solenoid valves and to replace aging solenoid valves had not been completed. Four valves were replaced, and the testing frequency of the valves had been increased to weekly.

1R13 Maintenance Risk and Emergent Work (71111-13)

.1 Maintenance Risk

a. Inspection Scope

The inspectors evaluated risk considerations for planned and emergent work on/in the following systems/areas:

Mitigating Systems Cornerstone

- Unit 1 station blackout diesel generator,
- Unit 2 high pressure coolant injection system and switchyard,
- Work week reviews for various systems.

Initiating Events Cornerstone

- Unit 2 reactor water cleanup relief valves lifting.

b. Issues and Findings

There were no findings or issues identified during this inspection.

.2 Safe Shutdown Makeup Pump

a. Issues and Findings

The inspectors also evaluated the risk due to internal fire initiating events given the Unit 2 safe shutdown makeup pump injection valve failure that occurred on January 19, 2000. The emergent work associated with this valve failure and the failure itself were previously evaluated in Inspection Report 50-254/2000001; 50-265/2000001.

Background

In Inspection Report 50-254/2000001; 50-265/2000001, the inspectors documented the results of their review of the safe shutdown makeup pump discharge valve failure for risk significance using the Significance Determination Process developed for internal initiating events. At that time, the failure was considered to be of very low risk (GREEN). Subsequently, the NRC determined that the failure needed to be evaluated in terms of increase in core damage frequency due to fire initiating events. This determination was based on the fact that the baseline core damage frequency due to fires was higher than the baseline core damage frequency due to other initiators, and the safe shutdown makeup pump system was known to be an important mitigating system for fire initiating events. The failure itself is described in the previous inspection report.

Safety Impact

The safe shutdown makeup pump system is one of three high pressure systems that can be used to inject water into the reactor during accident conditions. The safe shutdown makeup pump was classified as a risk significant system under the Maintenance Rule and was credited as an injection source in the Quad Cities Appendix R Safe Shutdown Analysis.

Risk Analysis/Considerations

The inspectors determined that information required to perform the Phase 2 fire Significance Determination Process was not readily available. The inspectors were unable to determine for what fire scenarios the safe shutdown makeup pump is used as a mitigating system, and also what other mitigating systems would remain available under those same fire scenarios. As a result of being unable to perform the Phase 2 analysis, the inspectors entered a Phase 3 analysis. With the assistance of regional and headquarter's risk analysts, the inspectors used available information to estimate the risk significance of the safe shutdown makeup pump injection valve failure.

Calculations

The inspectors used the risk achievement worth (RAW) of the safe shutdown makeup pump system and the core damage frequency (CDF) from the licensee's latest Individual Plant Examination for External Events to estimate a change in risk. The safe shutdown makeup pump was assumed to be unavailable for one month. The risk increase was then estimated using the following equation:

$$1/12[CDF(RAW-1)]$$

Using the reported core damage frequency of 6.6E-5 per year and risk achievement worth of 3.9, the inspectors calculated a risk increase of 1.5E-5 per year. The risk increase of 1.5E-5 is near the threshold of a white/yellow issue.

Conclusions/Recommendations

Using the approach described above, the inspectors concluded that this failure represented a potentially safety significant issue. The inspectors therefore recommended that the NRC continue to evaluate this issue via the Phase 3 Significance Determination Process.

1R14 Non-routine Plant Evolutions (71111-14)

a. Inspection Scope

The inspectors reviewed events surrounding the scram on Unit 2 on May 5, 2000. The inspectors reviewed the licensee's sequence of events report, operator logs, and corrective actions.

b. Issues and Findings

On May 5, 2000, an instrument technician caused a Unit 2 reactor trip during performance of Quad Cities Instrument Surveillance 0200-17, “Main Steam Line High Flow Calibration and Functional Test.” The instrument technician was adjusting pressure input to an instrument on the “A” channel group 1 primary containment isolation system, while mistakenly adjusting the trip set point for an instrument on the “B” channel. A full group 1 isolation signal resulted, which caused all main steam isolation valves to close and a subsequent reactor trip. Operators responded in accordance with procedures, and safety systems functioned normally. Minor problems with reactor recirculation pump speed control circuitry were experienced and corrected.

Operators restarted the unit on May 6, and returned the unit to full power operation on May 7. Some problems with control rods being hard to move were encountered, but these did not adversely affect safety. This event was reported to the NRC via the emergency notification system. The licensee is still reviewing corrective actions for this event and plans to document their actions on a licensee event report.

1R15 Operability Evaluations (71111-15)

a. Inspection Scope

The inspectors reviewed equipment operability issues in the Mitigating Systems Cornerstone including 125 Volt battery concerns and a concern with the Unit 1 average power range monitor system. Concerns with battery operability are discussed in Section 1R16 of this report. Problem Identification Form Q2000-00146 was written to address concerns that maintenance technicians had installed unqualified parts in the average power range monitor system on January 12, 2000. The inspectors reviewed the problem identification form and the root cause report dated March 2, 2000, and discussed the issue with station management.

b. Issues and Findings

The licensee discovered that maintenance technicians had installed unqualified parts in the Unit 1 average power range monitoring system on January 12, 2000. The parts were unqualified in that they had not been purchased as safety related. The technicians knowingly installed the unqualified parts because they were thought to be acceptable for testing. However, subsequent to the installation, operators placed the Number 1 average power range monitor into service, providing input into the reactor protection system. Criterion VIII of 10 CFR 50, Appendix B, “Identification and Control of Materials Parts and Components”, requires measures be established for the identification and control of material parts and components and that these measures be designed to prevent the use of incorrect material, parts and components. Use of unqualified parts in the average power range monitor was considered a violation of 10 CFR 50, Appendix B. This violation is being treated as a **non-cited violation (50-254/2000005-02)**, consistent with Section VI.A.1 of the May 1, 2000, Enforcement Policy. The inspectors determined that the initial root cause report was inadequate in that it did not address why operators placed the system into service with unqualified parts. The licensee wrote a subsequent problem identification form (Q2000-01882) to address this specific issue. This violation was entered into the licensee's corrective action program as Problem Identification Forms Q2000-01882 and Q2000-00146. The risk significance of this issue was determined to be very low (GREEN) because other average power range monitors could have provided redundant reactor protection system signals, if needed.

1R16 Operator Work-arounds (71111-16)

a. Inspection Scope

The inspectors continued to review the requirements for load stripping of the 125 Volt direct current (Vdc) and 250 Vdc batteries that were initially discussed in Inspection Report 50-254/2000003; 50-265/2000003 and documented as an unresolved item.

b. Issues and Findings

This inspection focused on the load stripping requirements for the 125 Vdc battery system. Quad Cities Abnormal Operating Procedure 6900-07, "Loss of A.C. [alternating current] Power to the 125 VDC Battery Chargers With Simultaneous Loss of Auxiliary Electrical Power," requires operators to initiate load shedding immediately if the alternating current feed to the 125 Vdc battery chargers can not be restored. The loads are required to be stripped within 30 minutes. The procedure directs the operator to shed 55 loads at 4 separate distribution panels. Although many of the loads are non-safety related, several of the loads are safety related and other non-safety related loads can affect accident mitigation and recovery. These loads include:

- Division A core spray and residual heat removal logic;
- reactor core isolation cooling system logic and speed governor control;
- control room annunciator Panels 901-53,54,56, and 912-2,7,8;
- plant sirens;
- outboard main steam isolation valves; and
- inboard main steam isolation valve dc solenoids and position indication.

During review of the load shedding procedures, the licensee provided correspondence between ComEd and the NRC from 1984 when the procedures were initially developed. At that time, there were concerns over battery capacity. As a result, the NRC issued a Confirmatory Action Letter on May 7, 1984, that confirmed implementation of procedures to reduce the 125 Vdc loads to below 62.3 amperes within 30 minutes on loss of the associated chargers. This commitment and a written justification for interim operation of the plant based on a battery profile analysis were determined to be sufficient to permit the startup and interim operation of Quad Cities Unit 2. On June 16, 1984, the NRC sent a request for additional information to the licensee. The NRC asked the licensee to explain why each of the loads which was shed was no longer needed after the first half-hour of the event. The licensee responded with a list of loads that "may be manually tripped" and stated that the loads ".. are all nonsafety-related and are not used to mitigate the consequences of an accident." This battery loading issue was also reported to the NRC in Licensee Event Report 84-08. The report stated that long-term corrective actions would include replacement of the non-seismic battery chargers with seismically-installed chargers of greater capacity and replacement of the 125 Vdc batteries with greater capacity batteries.

The inspectors developed several concerns regarding the load shedding of the 125 Vdc battery system. Specifically:

- Load shedding appeared to be an interim solution with the long-term solution being replacement of batteries with a larger capacity. The batteries and battery chargers were replaced, but the load shedding within 30 minutes remained as an assumption in the battery sizing calculation and the testing profile.

- The licensee reported to the NRC that the load shedding involved only nonsafety-related loads. However, a comparison of the current procedure, which included safety-related loads, showed that it was essentially the same as the original procedure and that safety-related loads were being shed.

- In the 10 CFR 50.59 evaluation for the load shedding, the licensee concluded that there was no unreviewed safety question. However, the evaluation did not include an analysis of the impact of operator errors with the addition of these manual actions. Specifically, it did not evaluate both the operator failure to implement load shedding which potentially could result in the battery not performing its' function and the operator improperly performing load shedding which could impact both the battery and other equipment. The 50.59 evaluation also did not review the impact on the plant of shedding these loads.

This issue will continue to be tracked as an **unresolved item (50-254/2000003-01; 50-265/2000003-01)** pending further review by the NRC and the licensee regarding the adequacy of the 50.59 reviews, corrective actions, and the risk associated with the implementation of these procedures.

1R19 Post-Maintenance Testing (71111-19)

a. Inspection Scope

The inspectors reviewed and/or observed the following post maintenance tests:

Mitigating Systems Cornerstone

- Unit 2 high pressure coolant injection pump and system tested on April 20
- shared safe shutdown makeup system

Barrier Systems Cornerstone

- "A" standby gas treatment system testing on May 3

b. Issues and Findings

There were no findings or issues identified during this inspection.

1R22 Surveillance Testing (71111-22)

a. Inspection Scope

The inspectors reviewed the results of the Unit 1 high pressure coolant injection system logic functional test and Unit 2 traversing in-core probe calibrations, both under the Mitigating Systems Cornerstone.

b. Issues 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. This issue was documented in Inspection Report 50-254/200003; 50-265/200003.

The inspectors used Phase 2 of the Significance Determination Process and determined that the risk from internal events of the high pressure coolant injection system being inoperable for automatic starts for approximately one year was very low (GREEN). The risk due to external events, specifically fires, was also evaluated using the Phase 2 significance determination process. Using a core damage frequency for fires of 6.6E-5, risk achievement worth for the high pressure coolant injection system of 1.04, and approximately 11 months of unavailability time, the inspectors calculated a risk factor of 2.42 E-6 for the external event risk of the system unavailability. Further work to refine the assumptions of the Significance Determination Process was halted because this issue was the overriding contributor to a YELLOW performance indicator for equipment availability. The regulatory actions for this indicator will encompass the actions for the inspection finding.

Emergency Preparedness (EP)

1EP1 Drill, Exercise, and Actual Events (71114-06)

a. Inspection Scope

The inspectors observed simulator training of Operating Crew “E” on April 26, 2000, and Operating Crew “D” on May 3, 2000. The inspectors observed the crews attempt to mitigate the accident scenarios. The inspectors evaluated communications, procedure adherence, and implementation of emergency operating procedures. In addition, event classification and reporting actions were also assessed.

b. Issues and Findings

The licensee identified that Operating Crew “E” did not accurately complete the state notification portion of the Nuclear Accident Reporting System form. Two of three other opportunities (to classify and to make proper protective action recommendations) were completed satisfactorily relative to the Drill/Exercise Emergency Preparedness performance indicator. The licensee had identified weaknesses in this area and was planning corrective actions (reference Problem Identification Forms Q2000-01730, and Q2000-01747). There were no issues or findings for this inspection activity.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

.1 Safety System Unavailability - Residual Heat Removal

a. Inspection Scope (71151)

The inspectors reviewed operator logs, performance indicator guidance, and licensee safety system performance sheets to verify the licensee's residual heat removal system unavailability performance indicator information for the first quarter of 2000.

b. Issues and Findings

The inspectors completed the verification inspection of residual heat removal unavailability for the first quarter 2000. The licensee retracted about 45 hours of system unavailability during a planned outage of all residual heat removal trains during the Unit 2 outage. The licensee based this retraction on a frequently asked question as documented in NEI (Nuclear Energy Institute) 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline." There were no issues or findings from this inspection activity.

4OA3 Event Follow-up (71153)

a. Inspection Scope

The inspectors reviewed corrective actions associated with licensee event reports.

b. Issues and Findings

(Closed) Licensee Event Report 50-265/00002-00: Emergency Diesel Generator Inoperable due to Inadvertent Movement of Ventilation Fan Power Select Switch. On January 28, 2000, the licensee identified that a power select switch for the Unit 2 emergency diesel generator was in the improper position. The licensee could not identify how or when the switch was mis-positioned. This issue and its risk significance were discussed in Inspection Report 50-254/00003; 50-265/00003(DRP). This item is closed.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Barnes and other members of licensee management listed below at the conclusion of the inspection on May 16, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Barnes, Station Manager
P. Behrens, Chemistry Manager
G. Boerschig, Engineering Manager
R. Gideon, Work Control Manager
M. McDowell, Operations Manager
M. Perito, Maintenance Manager
C. Peterson, Regulatory Assurance Manager
F. Tsakeres, Training Manager

NRC

M. Ring, Branch Chief, Division of Reactor Projects

Illinois Department of Nuclear Safety

R. Ganser, Resident Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-265/2000005-01	URI	Increased failure rate of turbine solenoid valves
50-254/2000005-02	NCV	Operability Evaluations

Discussed

50-254/2000003-01; 50-265/2000003-01	URI	Load Shedding 125 Vdc Battery Loads during Accident Conditions

Closed

50-254/2000005-02	NCV	Operability Evaluations
50-265/00002-00	LER	Emergency Diesel Generator Inoperable due to Inadvertent Movement of Ventilation Fan Power Select Switch

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
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-06	Flood Protection Measures	1R06
71111-11	Licensed Operator Requalification	1R11
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Work Prioritization & Control	1R13
71111-14	Non-routine Evolutions	1R14
71111-15	Operability Evaluations	1R15
71111-16	Operator Work-Arounds	1R16
71111-19	Post Maintenance Testing	1R19
71111-22	Surveillance Testing/Post Maintenance Testing	1R22
71114-06	Drill, Exercise, and Actual Events	1EP1
71150	Plant Status	
71151	Performance Indicator Verification	4OA1
71153	Event Follow-up	4OA3
(none)	Management Meetings	4OA6

LIST OF ACRONYMS AND INITIALISMS USED

AC	Alternating Current
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
LER	Licensee Event Report
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
PERR	Public Electronic Reading Room
QCOS	Quad Cities Operating Surveillance
RAW	Risk Achievement Worth
TBD	To Be Determined
URI	Unresolved Item
Vdc	Volts Direct Current