July 30, 1999

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: NRC INSPECTION REPORT 50-254/99014(DRS); 50-265/99014(DRS)

Dear Mr. Kingsley:

On July 15, 1999, the NRC completed the pilot baseline biennial heat sink inspection at your Quad Cities Nuclear Station. The results of this inspection were discussed on July 15, 1999, with Mr. Bohlke and other members of your staff. The enclosed report presents the results of this inspection.

The inspection consisted of a review of documents related to heat exchanger testing and performance as well as interviews with responsible personnel.

Based on the results of this inspection, NRC identified an issue which was categorized as being of low risk significance. This issue has been entered into your corrective action program. In addition, two previously identified issues were evaluated under the risk significance determination process and were determined to be of low risk significance, although regulatory requirements were violated. Based on the review of these two issues, one Non-Cited Violation was identified and is discussed in the subject inspection report.

If you contest 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 20555-0001, with copies to the Regional Administrator, Region III and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

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 (PDR).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/s/ J. M. Jacobson

John M. Jacobson, Chief
Mechanical Engineering Branch

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

Enclosure: Inspection Report 50-254/99014(DRS); 50-265/99014(DRS)

cc w/encl: D. Helwig, Senior Vice President
H. Stanley, PWR Vice President
C. Crane, BWR Vice President
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

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-254; 50-265
License Nos:	DPR-29; DPR-30
Report No:	50-254/99014(DRS); 50-265/99014(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Quad Cities Nuclear Power Station Units 1 and 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	July 13 - 15, 1999
Inspector:	Patricia Lougheed, Reactor Engineer
Approved by:	John M. Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Quad Cities Nuclear Power Station, Units 1 & 2
NRC Inspection Report 50-254/99014(DRS); 50-265/99014(DRS)

This report covers the pilot baseline inspection for the biennial review of heat sink performance. The heat sink performance inspection covers an inspectable area under the Initiating Events and Mitigating Systems cornerstones for which there is no performance indicator. Adequate or superior performance is not reported. Findings are assessed according to their potential risk significance and are categorized within color coded bands based on this assessment. The green band indicates those issues of low risk significance which can be turned over to the licensee for corrective action. The white band indicates issues with some increased risk to safety which would require additional regulatory and licensee concern. The yellow band is indicative of more serious issues with higher potential risk to safety. No individual finding is indicative of either acceptable or unacceptable performance. The findings will be combined with other inspection findings and with performance indicators in a separate assessment process to determine overall plant performance.

Mitigating Systems

- Green: The surveillance procedure for evaluating thermal performance of the residual heat removal heat exchangers contained errors which resulted in the licensee overestimating the heat removal capability of the 1A heat exchanger. The heat exchanger was still capable of removing its design heat load.

- Green: A non-cited design control violation with multiple examples was identified during close out of two unresolved items from the architect-engineer inspection (50-254/265-98201). The issues dealt with ensuring adequate net positive suction head for the emergency core cooling system pumps, ensuring the residual heat removal service water piping was analyzed for its design condition, and determining the adequacy of a thermal relief valve. All the examples in the Non-Cited Violation resulted from original design deficiencies.

1. REACTOR SAFETY

1R07 Heat Sink Performance

.1 Surveillance Procedure Errors

a. Inspection Scope

The inspector reviewed the surveillance procedure for thermal performance testing of the 1A residual heat removal heat exchanger. The inspector also reviewed the preventive maintenance (pre-defined work) instructions for inspection and cleaning of the high pressure injection pump room cooler and the emergency diesel generator heat exchanger. These heat exchangers were chosen for review as they were representative of three of the top four risk significant heat exchangers at the station.

b. Observations and Findings

The inspector identified that the residual heat removal thermal performance test procedure, QCOS-1000-29, contained incorrect information, which resulted in a nonconservative calculation of actual heat exchanger performance.

During review of the completed test for the 1A residual heat removal heat exchanger, the inspector noted that the heat exchanger performance had increased dramatically over previous tests (from having less than 2 percent margin to having over 300 percent). The licensee attributed the increase to their ensuring that adequate differential temperatures existed between the suppression pool and the river prior to starting the test. However, the inspector ascertained that the temperature differential was insufficient to account for the magnitude of the change.

Upon further review, the inspector determined that the factor that had changed most was the shell side film coefficient. This factor is somewhat dependent upon the shell side temperatures, but is mostly affected by the velocity of the water going across the tubes. The velocity across the tubes is a function of the heat exchanger shell side cross-sectional area and the flow rate. Neither the area or the flow rate normally changed between tests.

The licensee had revised the procedure in October 1998, based on receipt of a revised data sheet from the heat exchanger manufacturer. The revised data sheet gave a design velocity that was 37 percent lower than the previous one. The licensee used this velocity to calculate the cross-sectional area, which showed a corresponding increase. The inspector questioned the validity of the decreased velocity, as the heat exchanger had not physically changed. A licensee engineer performed a rough calculation of the cross-sectional area, based on the tube pitch and spacing, and found it to be closer to the previous data sheet information, rather than the revised values.

Both the licensee and the inspector independently calculated the test results using the previous heat exchanger design velocity and determined that the heat exchanger performance was acceptable.

The inspector also noted that the revised data sheet appeared to give an incorrect value for the gross tube area. The gross tube area was based on 2278 tubes, which was the number of tubes the licensee credited for design conditions. The heat exchanger actually contained 2415 tubes, which would have to be factored into shell side performance, even under design conditions (as they took up physical space inside the heat exchanger.) In one procedure step, the licensee was using the incorrect gross tube area instead of the correct effective tube area of the larger number of tubes.

Because the heat exchanger results were independently determined to be acceptable and the heat exchanger could meet its design function, this issue was determined to be of low risk significance and was categorized as "Green." It was entered into the licensee's corrective action program as problem identification form Q1999-02347.

.2 (Closed) URI 50-254/98201-01; 50-265/98201-01: Residual Heat Removal and Core Spray Systems Pump Net Positive Suction Head

This issue concerned the assumptions used and conclusions reached in the design basis post-accident net positive suction head calculations regarding containment overpressure. The licensee recalculated the long term post-accident calculation and determined that adequate net positive suction head existed for the emergency core cooling system pumps as long as credit was taken for containment overpressure. As this did not meet their licensing basis, the licensee recognized that NRC approval was necessary and submitted the appropriate documents for review by the Office of Nuclear Reactor Regulation (NRR). At the time of the inspection the licensee was working with NRR to resolve concerns as well as performing short term post-accident calculations.

The licensee's analyses showed that the pumps were operable. Therefore, this issue screened out of the significance determination process as “Green.” This item is in the licensee's corrective action program. 10 CFR Part 50, Appendix B, Criterion III, “Design Control” requires that design inputs be correctly translated into design documents. The failure to adequately control design inputs at the time of original plant design regarding reliance on containment overpressure to ensure adequate net positive suction head is an example of a violation of Criterion III and will be treated as a Non-Cited Violation. (NCV 254/265-99014-01(DRS)).

.3 (Closed) Unresolved Item (URI) 50-254/98201-15; 50-265/98201-15: Residual Heat Removal Service Water Design Pressure and Overpressure Protection.

The unresolved item encompassed two issues: (1) a pressure below the design pressure was used in portions of the piping stress analyses for the residual heat removal service water system and (2) the residual heat removal service water relief valve was not sized to handle the pump shutoff head, although the original General Electric design specification required it to handle that volume of water. These issues were entered into the licensee’s corrective action program.

In regard to the first issue, the licensee reanalyzed the piping and concluded that the stresses were acceptable. The inspectors reviewed the issue against the significance determination process criteria and determined that the issue was “Green.” However, the failure to adequately control design inputs by using design pressures in the piping stress analyses which were inconsistent with actual plant design is a second example of the Criterion III Non-Cited Violation.

In regard to the second issue, the licensee credited operational administrative controls as adequate corrective actions to ensure compliance to the ASME welding code of recor (B31.1). Although this solution was not an accepted method for resolving such code discrepancies; the inspector reviewed the issue against the significance determination process criteria and determined the risk significance was low, because the piping most probably could withstand the higher pressure. The failure to incorporate the original design requirement into the plant design is a third example of the Criterion III Non-Cited Violation.

4 OTHER ACTIVITIES

4OA4 Other

.1 (Closed) URI 50-254/98201-03; 50-265/98201-03: Inclusion of Emergency Core Cooling System Flow Measurements

This unresolved item dealt with a concern over what uncertainty factors for flow measurements needed to be included. This issue is under review by the Office of Nuclear Reactor Regulation (NRR) as a generic concern. Once NRR has completed its review, the licensee will be informed of any necessary actions by separate correspondence. Therefore this item is closed.

.2 (Closed) Violation 50-254/98019-01; 50-265/98019-01: Inadequate Test Control. In Inspection Report 50-254/265-98019, the inspectors noted that the licensee had taken adequate corrective actions to the violation and that no response was necessary. Therefore, this violation is closed.

.3 (Closed) Violation 50-254/98019-02; 50-265/98019-02: Inadequate Corrective Action. In Inspection Report 50-254/265-98019, the inspectors noted that the licensee had taken adequate corrective actions to the violation and that no response was necessary. Therefore, this violation is closed.

.4 (Closed) Violation 50-254/98019-03; 50-265/98019-03: Inadequate Design Control. In Inspection Report 50-254/265-98019, the inspectors noted that the licensee had taken adequate corrective actions to the violation and that no response was necessary. Therefore, this violation is closed.

40A5 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management in an exit meeting on July 15, 1999. The licensee acknowledged the information and findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Barnes, Station Manager
W. Bohlke, Engineering Vice President, Nuclear Generation Group
M. McDonald, Operations Manager
C. Peterson, Regulatory Assurance
D. Wozniak, Engineering Manager

NRC

J. Caldwell, Deputy Regional Administrator, Region III
L. Collins, Resident Inspector
J. Jacobson, Chief, Mechanical Engineering Branch, Division of Reactor Safety
S. Reynolds, Deputy Division Director, Division of Reactor Safety
M. Ring, Chief, Division of Reactor Projects, Branch 1

INSPECTION PROCEDURE USED

IP 71111.07 (draft) Heat Sink Performance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

254/265/99014-01	NCV	Three Examples of Design Control Relating to Original Plant Design

Closed

254/265/98201-01	URI	Residual Heat Removal and Core Spray Systems Pump Net Positive Suction Head
254/265/98201-03	URI	Inclusion of Emergency Core Cooling System Flow Measurements
254/265/98201-15	URI	Residual Heat Removal Service Water Design Pressure and Overpressure Protection
254/265/98019-01	VIO	Inadequate Test Control
254/265/98019-02	VIO	Inadequate Corrective Action
254/265/98019-03	VIO	Inadequate Design Control
254/265/99014-01	NCV	Three Examples of Design Control Relating to Original Plant Design

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically so stated in the body of the inspection report.

M-37	Diagram of Residual Heat Removal Service Water Piping, Revision AP
NSP-ER-3014	Generic Letter 89-13 Program Implementing Procedure (and Instructional Guide), Revision 0
NDIT QDC-98-230	Nuclear Design Information Transmittal of Revised Residual Heat Removal Heat Exchanger Datasheet Based upon Reissued Data Sheet from Heat Exchanger Original Equipment Manufacturer, August 20, 1998
NED-M-MSD-47	Calculation of Quad Cities Residual Heat Removal Heat Exchanger Performance Requirements as a Function of River Temperature, Revision 0

Problem Identification Forms:

Q1998-01716	River Water Leaking into ½ Emergency Diesel Generator Coolant, April 6, 1998
Q1998-05492	Unit 2 High Pressure Coolant Injection Room Cooler West End Baffle Plate Gaskets Found Missing During Inspection, December 7, 1998
Q1999-02269	Nuclear Operator Identifies Untracked Commitment to Generic Letter 89-13, July 06, 1999
Q1999-02345	Residual Heat Removal Service Water Pump Testing and Safety Margin, July 15, 1999
Q1999-02347	Residual Heat Removal Heat Exchanger Test (QCOS-1000-29) Methodology, July 15, 1999

Procedures:

QCCP-1000-05	Residual Heat Removal Service Water Heat Exchanger Predefined Inspection, Attachment A: For Unit 1 Diesel Generator Heat Exchanger, March 19, 1998
QCOS 1000-29	Residual Heat Removal Heat Exchanger Thermal Performance Test, Revision 1 with Completed Procedure for the 1A Heat Exchanger, December 11, 1998
QCTP-0820-10	Heat Exchanger and Room Cooler Inspection, Revision 0 with Completed Attachment A: For Unit 2 High Pressure Coolant Injection Room Cooler, December 7, 1998