February 6, 2001

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: DRESDEN - NRC INSPECTION REPORT 50-237/01-06(DRS); 50-249/01-06(DRS)

Dear Mr. Kingsley:

On January 25, 2001, the NRC completed an inspection at the Dresden Nuclear Generating Station, Units 2 and 3. The enclosed report presents the results of that inspection. The results were discussed on January 25, 2001, with Mr. P. Swafford and other members of your Dresden staff.

The inspection was an examination of activities conducted under your license as they relate to compliance with the Commission's rules and regulations and with the conditions of your license. The inspector reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). These issues were determined to involve two violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Dresden Nuclear Generating Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John M. Jacobson, Chief
Mechanical Engineering Branch
Division of Reactor Safety

Docket Nos. 50-237; 50-249
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 50-237/01-06(DRS); 50-249/01-06(DRS)

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
P. Swafford, Site Vice President
R. Fisher, Station Manager
D. Ambler, Regulatory Assurance Manager
M. Aguilar, Assistant Attorney General
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	50-237/01-06(DRS); 50-249/01-06(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Dresden Nuclear Generating Station, Units 2 and 3
Location:	6500 N. Dresden Road Morris, IL 60450
Dates:	January 22 - 25, 2001
Inspector:	M. Holmberg, Reactor Engineer
Approved by:	J. Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

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

IR 05000237-01-06(DRS), 050000249-01-06(DRS), on 01/22-25/2001, Commonwealth Edison Company, Dresden Nuclear Generating Station, Units 2 and 3. Heat Sink Performance Inspection.

The inspection was conducted by a Region III engineering specialist. The inspection identified two Green findings, which were Non-Cited Violations. The significance of findings is indicated by their color (Green, White, Yellow, Red) using IMC 609 “Significance Determination Process” (SDP).

A. <u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- Green. One Non-Cited Violation was identified for the licensee’s failure to appropriately evaluate test data associated with measuring the thermal performance of the isolation condensers to assure that test requirements had been satisfied.

  The safety significance of this finding was very low because the affected mitigation system remained operable. This issue was considered more than minor, because if left uncorrected, it could impact the ability of the licensee to detect degradation or loss of isolation condenser function (Section 1R07.b.1).

- Green. One Non-Cited Violation was identified for the licensee’s failure to enter in the corrective action system and appropriately evaluate a concern for the isolation condenser shell side integrity.

  The safety significance of this finding was very low because the affected mitigation system remained operable. This issue was considered more than minor, because it was not adequately evaluated since identification in 1996, and it had the potential to challenge accident mitigation associated with a tube rupture in the isolation condenser (Section 1R07.b.2).

Report Details

Summary of Plant Status: Unit 2 and 3 were at 100 percent power during this inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R07 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the documents associated with maintenance and thermal performance testing of the Unit 3 isolation condenser, the Unit 2 low pressure coolant injection system heat exchangers (HXs), and the Unit 2 high pressure coolant injection system pump lube oil cooler. These HXs were chosen for review based on a relatively high risk achievement worth. The inspector reviewed completed surveillances and associated calculations to confirm that these HXs met their design heat removal requirements or that licensee maintenance practices were adequate to assure design performance.

The inspector reviewed condition reports concerning heat exchanger or heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues. The inspector also evaluated the effectiveness of the corrective actions to the identified issues, including the engineering justification for operability, if applicable. The documents that were reviewed are included at the end of the report.

b. Findings

.1 Isolation Condenser Heat Removal Testing Deficiencies

The inspector reviewed DOS 1300-01, "Isolation Condenser Five Year Heat Removal Capability Test," performed in 1992 and 1997 to evaluate the heat removal capability of the Unit 3 isolation condenser. These tests were performed with the reactor at power. The heat removal capability was determined using an energy balance with inputs of reactor power and feed flow rates before and after system initiation. The calculated heat removal rate and system heat removal capability over 20 minutes was compared to values derived from the Updated Final Safety Analysis Report, Section 5.4.6. The inspectors identified the following deficiencies with this testing, which were nonconservative and applicable to both Units:

1) The 1992 and 1997, Unit 3 isolation condenser thermal performance tests were performed at approximately 70 percent power. With the plant at power, the reactor feed and steam flows create a differential pressure and inductor affect across the isolation condenser piping taps, which would change with the reactor shutdown.

The Unit 3 isolation condenser was placed in service following a 1999 scram and the tank level initially boiled off at approximately 1/2 of the rate observed during the at power tests. This data indicated that the test conditions at power do not accurately reflect system operating conditions with the reactor shutdown.

2) The 1992 and 1997 tests were intended to demonstrate that an adequate volume of water existed in the isolation condenser such that tank level would not reach the condenser tubes within the first 20 minutes of isolation condenser operation. During both of these tests, the shell side tank level end point desired for testing (3.0 feet) was not reached due to high radiation levels detected in the vent line radiation monitors caused by loss of shielding as the tank level decreased. The licensee estimated the additional time needed to reach the three foot tank level and used this value to confirm that sufficient water inventory existed to allow design heat removal for 20 minutes without uncovering the isolation condenser tubes. Based on design drawings, the inspector identified that the top of the tubes was actually at a tank level of 3 feet 4 inches and no allowance for tank level uncertainty was considered in this test.

In calculation BSA-D-95-07, "Dresden Isolation Condenser Performance," the licensee established a minimum level required in the tank at several shell side water temperatures to ensure the design thermal quench capacity. The test as performed did not consider or incorporate the results of this calculation, such that a bounding test condition was established with respect to meeting the design thermal quench capacity.

Because of these deficiencies and the relatively small margin available, the test did not adequately demonstrate that a sufficient tank water volume existed in the isolation condenser.

3) The calculated heat removal rate of the isolation condenser determined from these tests was not corrected or evaluated for the affects of instrument uncertainty, and the value derived from the test was compared directly with the minimum allowable design value. Specifically, the impact of instrument uncertainty from test data inputs for the average power range monitor and feedwater flow rate inputs had not been considered. Further, the heat removal rate varied in excess of five percent between the 1992 and 1997 tests and the licensee had not evaluated this condition, because test results were not trended. Lack of a margin for instrument uncertainty was nonconservative and potentially significant because of the relatively small margins involved.

4) These tests did not evaluate the impact of a valve positioning error band on measured system performance. The test results are very sensitive to the throttle position of the 12 inch gate type isolation valve (motor operated valve 1301-3) on the condensate return leg for the system. Therefore, the repeatability of the test results is heavily dependant on how accurately the motor operated valve can position the valve stem and disc. For example, during Unit 2 testing, a 1/4 inch change in this valve's position, changed the measured heat removal rate by 45 percent of the design heat removal rate. This was potentially significant because

the 1997, Unit 3 test demonstrated only a 12 percent margin over the minimum required design value.

Because test data had not been appropriately adjusted/evaluated to allow comparison with design acceptance criteria, the tests performed did not confirm that the isolation condensers met the minimum required design heat removal capacity. These testing deficiencies were considered more than minor, because if left uncorrected, they could impact the ability of the licensee to detect degradation or loss of isolation condenser function.

No changes had been made to the Unit 3 system since it adequately performed its function following a 1999 scram event and the Unit 2 isolation condenser performance was bounded by Unit 3 isolation condenser performance. Therefore, based on engineering judgement, the licensee considered that these testing deficiencies did not affect the operability of the isolation condensers.

This finding did have a credible impact on safety; however, since only the mitigation cornerstone is affected and the system remained operable, the finding screened by the SDP, is considered to be of very low safety significance (Green). The licensee failed to appropriately evaluate test data associated with the isolation condensers to assure that test requirements have been satisfied, which is a violation of 10 CFR Part 50 Appendix B, Criterion XI, “Test Control.” However, due to the very low safety significance of the item and because the licensee entered this item into the corrective action program (CR D2001-00451), this violation is a Non-Cited Violation (NCV 50-237/01-06-01; NCV 50-249/01-06-01) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

.2 Inadequate Evaluation of the Isolation Condenser Tank Vent Configuration

The isolation condenser system design specification GE 21A1608 section 5.2.6.3 requires that shell vents be sized to accommodate design conditions which include system initiation, concurrent with a 10 percent tube flow area rupture without exceeding the shell design pressure (25 psig). The licensee purchased the isolation condenser with two 24 inch vent openings in the tank shell. These two vent openings were then routed via 25 inch diameter piping to a single 32 inch diameter vent pipe which is routed outside the reactor building. With this vent configuration a potential choked flow condition could be developed which would invalidate the original system vent design.

In March of 1996, the licensee identified a concern for the potential choked vent flow condition in calculation BSA-D-95-07, “Dresden Isolation Condenser Performance.” The licensee chose not to evaluate this condition, because current station practices were considered sufficient to assure tube integrity. However, the licensee engineer involved in reviewing this issue in 1996, was unaware of the design specification requirements associated with tube rupture and shell side vent design. Therefore, an adequate basis for the decision to not enter and evaluate this condition adverse to quality in the corrective action program had not been established.

A postulated rupture of the shell side of the isolation condenser would cause in excess of 22,000 gallons of water to escape and potentially drain down through open gratings below the tank into the valve rooms which house the isolation valves for the isolation condenser. This could potentially affect the ability of operators to isolate the condenser and successfully mitigate a tube rupture. Therefore, this issue was considered more than minor, because it was not adequately evaluated, and it posed a challenge for accident mitigation associated with a tube rupture in the isolation condenser.

On January 25, 2001, the licensee performed a preliminary analysis, “Evaluation of Isolation Condenser Shell Pressurization During Operation with Tube Rupture,” which demonstrated that the vent design was adequate to prevent shell side pressure from exceeding design.

This finding did have a credible impact on safety; however, since only the mitigation cornerstone is affected and the system remained operable, the finding screened by the SDP, is considered to be of very low safety significance (Green). Upon identification in 1996, the licensee failed to enter and appropriately evaluate the concern for the isolation condenser shell side integrity (condition adverse to quality) in the corrective action system, which is a violation of 10 CFR Part 50 Appendix B Criterion XVI “Corrective Action.” However, due to the very low safety significance of the item and because the licensee subsequently entered this item into the corrective action program (CR D2001-00446), this violation is a Non-Cited Violation (NCV 50-237/01-06-02; NCV 50-249/01-06-02) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

4. OTHER ACTIVITIES

4OA5 Management Meetings

Exit Meeting Summary

The inspector presented the inspection results to Mr. P. Swafford, Site Vice President, and other members of licensee management at the exit meeting held on January 25, 2001. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Swafford, Site Vice President
R. Fisher, Plant Manager
K. Bowman, Operations Manager
R. Whalen, Engineering Manager
M. Mohammed, Design Engineering Manager
J. Reda, Design Engineering
M. Martivovich, System Engineering
J. Ellis, System Engineering
L. Lewandowski, System Engineering
P. Chennel, System Engineering

NRC

D. Smith, Senior Resident Inspector
B. Dickson, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened & Closed

NCV 50-237/01-06-01; NCV 50-249/01-06-01	Failure to appropriately evaluate thermal performance test data associated with the isolation condensers to assure that test requirements have been satisfied is a violation of 10 CFR Part 50 Appendix B, Criterion XI.
NCV 50-237/01-06-02; NCV 50-249/01-06-02	Failure to enter and appropriately evaluate the concern for the isolation condenser shell side integrity (condition adverse to quality) in the corrective action system is a violation of 10 CFR Part 50 Appendix B, Criterion XVI.

Discussed

None

LIST OF DOCUMENTS REVIEWED

Calculations

BSA-D-99-04, Revision 1, “Dresden Unit 2 and 3 Reconstitution of Isolation Condenser Design Bases with Respect to Decay Heat Loads and Long Term Makeup Requirements”
BSA-D-95-07, Revision 0, “Dresden Isolation Condenser Performance”
ATD-0141, Revision 0, “Transient Pressure Analysis of the Isolation Condenser for 10 CFR Part 50 Appendix R”
RSA-D-93-06, Revision 2, “Dresden IC Extended Heat Removal Capacity”
NED-I-EIC-306, Revision 0, “Isolation Condenser Level Loop Accuracy and High Level Setpoint”
000652-02, Revision A, “Data Reduction of Heat Transfer Test Data for the 3B LPCI/CCSW Heat Exchanger” - Draft Version
DR-029-M-001, Revision 1, “CCSW System Design Input for the Balance Heat Exchanger Performance Utility”
DRE 96-0162, Revision 0, “LPCI HX Performance Versus River Water Temperature”
T23-000740-03, “GE Final Report GENE-T23-00740-03 for Containment Analysis Utilizing a New Decay Heat Curve with a 2 Sigma Adder Based on ANS 5.1-1979 Decay Heat Standard”
GE-NE-T2300740-2, “Dresden Nuclear Power Station Units 2 and 3 Containment Analysis of the DBA-LOCA Based on Long Term LPCI/Containment Cooling System Configuration of on LPCI/Containment Cooling System Pump and 2 CCSW Pumps”
HPCI-06, Revision 0, “Supporting HPCI Oil Cooler Tube Repairs”

Drawings

66-2-5636C1, Revision 6, “Setting Plan 144" OD Isolation Condenser Type 144-IR-44-33-18-18-U-42-82H”
66-2-563D1, Revision 3, “Shell Details for 144" OD Isolation Condenser”
66-2-5637D1, Revision 0, “Tube Bundle Assembly for an Isolation Condenser”

Condition Reports

D1998-1825, “FME Found in Isolation Condenser”
D1999-00640, “Broken Supports for ISCO Internal Baffle Plates”
D1999-00715, “FME Concerns from Broken Supports for ICO Internal Baffle Plates”
D2000-00514, “Tube Leak in 3A LPCI Heat Exchanger”
D2000-01384, “3A LPCI HX Tube Leak-Rework”
D2000-06952, “LPCI Heat Exchanger Leaks”
D1998-00105, “Drawing M-51 Missing VLV For U2 HPCI Lube Oil Cooler”

Heat Exchanger Data Sheets and Design Specifications

21A1608, Revision 2, “Isolation Condenser”
Specification Sheet 205-92017, Revision 2, “Isolation Condenser”
6B-3222, Revision 3 “Containment Exchanger”
Specification 165A202EF, Revision 2 “Cooler-Oil”
CE-67-8418, “Vertical Heat Exchanger”
21A5778, Revision 0, “General Requirements for Auxiliary Steam Turbine Drives (HPCI System)”

Vendor Manuals

D1331, Struthers Wells, “Instruction Book for Isolation Condenser for Dresden”
6B-3222, Perfex Corporation, “Perfex Instruction Manual for GE Containment Cooling Heat Exhanger EPN 1503"

Procedures/Surveillances

DOS 1300-01, Revision 14 “Isolation Condenser Five Year Heat Removal Capability Test” Performed in 1997 for the Unit 3 Isolation Condenser
DOS 1300-01, Revision 7 “Isolation Condenser Five Year Heat Removal Capability Test” Performed in 1992 for the Unit 3 Isolation Condenser
DTS 1500-01, Revision 1 “Containment Cooling Heat Exchanger Thermal Test Data” Completed on 2/12/1998 for the 2B LPCI HX and on 1/21/99 for the 2A LPCI HX
DOS 2300-03, Revision 65 “High Pressure Coolant Injection System Operability Verification”
DAN 902(3)-3-F-10, Revision 4, “HPCI Oil CLR Disch Oil Temp Hi”

Work Requests

990019981
990007338
990012493
980038988
970075680
970075994
970010761
970043298
960023485
960034434
960023209
960111112

Correspondence

Memorandum dated November 4, 1996, “Low Pressure Coolant Injection (LPCI) Heat Exchanger Shell to Tube Pressure Boundary Integrity”
Letter dated November 25, 1996, “Overall Heat Transfer Coefficient, Dresden Unit 2 and 3 Containment Cooling Heat Exchangers”
Letter dated October 1, 1996, “Evaluation of Reduced LPCI HX Performance”
Memorandum dated May 20, 1998, “Systems Materials Analysis Department Report on the Evaluation of HPCI Lube Oil Cooler Tubes From Unit 2 at Dresden Station”

LIST OF INFORMATION REQUESTED

For heat exchangers (HXs) (Unit 3 Isolation Condenser, Unit 2 Low Pressure Injection HXs, Unit 3 Drywell Coolers) the following information is needed in the resident inspectors office before January 22, 2001, to support the biennial “Heat Exchanger Performance” inspection procedure 71111.07:

1. Copy of the two most recently completed tests confirming thermal performance of each HX. Include documentation and procedures that identify the types, accuracy, and location of any special instrumentation used for these tests. (e.g., high accuracy ultrasonic flow instruments or temperature instruments). Include calibration records for the instruments used during these tests.

2. Copy of the evaluations of data for the two most recent completed tests confirming the thermal performance of each HX.

3. Copy of the calculation which establishes the limiting (maximum) design basis heat load which is required to be removed by each of these HXs.

4. Copy of the calculation which correlates surveillance testing results from these HXs with design basis heat removal capability (e.g., basis for surveillance test acceptance criteria).

5. The clean and inspection maintenance schedule for each HX.

6. For the last two clean and inspection activities completed on each HX, provide a copy of the document describing the inspection results.

7. Provide a copy of the document which identifies the current number of tubes in service for each heat exchanger and the supporting calculation which establishes the maximum number of tubes which can be plugged in each HX.

8. Provide a copy of the document establishing the repair criteria (plugging limit) for degraded tubes which are identified in each HX.

9. Copy of the design specification and heat exchanger data sheets for each HX.

10. Copy of the vendor/component drawing for each HX.

11. Provide a list of issues with a short description documented in your corrective action system associated with these HXs in the past three years.

12. Provide a list of calculations with a short description which currently apply to each HX.

13. Provide HX performance trending data tracked for each HX.

If the information requested above will not be available, please contact Mel Holmberg as soon as possible at (630) 829-9748 or E-mail - msh@NRC.gov.