January 29, 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 INSPECTION REPORT 50-237/00-21(DRP); 50-249/00-21(DRP)

Dear Mr. Kingsley:

On January 4, 2001, the NRC completed an inspection at Dresden Units 2 and 3. The enclosed report documents the inspection findings which were discussed on January 4, 2001, with Mr. Fisher and other members of your staff.

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

Based on the results of this inspection, the inspectors identified two issues for which no risk significance or color was assigned. Those two issues involved problem identification and resolution for radiological postings and human performance in three other areas. In addition, the inspectors identified three issues of very low safety significance (GREEN). The first issue involved the fire marshal's failure to determine that a fire drill had failed. The second issue involved the inadvertent opening of an electromatic relief valve due to an instrument maintenance technician's failure to follow procedure. The third issue is associated with a reactor scram that occurred due to a configuration control problem while working in the switchyard. Two of these issues were determined to involve 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at the Dresden 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 available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PAR) 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/

Mark Ring, Chief
Reactor Projects Branch 1

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

Enclosure: Inspection Report 50-237/00-21(DRP);
50-249/00-21(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: License Nos:	50-237; 50-249 DRP-19; DRP-25
Report No:	50-237/00-21(DRP); 50-249/00-21(DRP)
Licensee:	Commonwealth Edison Company (ComEd)
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	November 15, 2000 through December 31, 2000
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector R. Lerch, Project Engineer P. Pelke, Reactor Engineer P. Lougheed, Reactor Inspector G. O'Dwyer, Reactor Inspector R. Jickling, Emergency Preparedness Analyst R. Zuffa, 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

IR 05000237-00-21, IR 05000249-00-21; on 11/15 - 12/31/2000; Commonwealth Edison Company; Dresden Nuclear Power Plant; Units 2 and 3. Fire Protection, Non-routine Evolutions, Access Control to Radiologically Significant Areas, and Human Performance.

The inspection was conducted by resident, regional, and Illinois Department of Nuclear Safety inspectors. The inspection identified three GREEN findings, two of which were Non-Cited Violations. The inspection also identified a human performance cross-cutting issue and a problem identification and resolution cross-cutting issue with no color. The significance of most findings is indicated by their color (GREEN, WHITE, YELLOW, RED) using Inspection Manual Chapter 0609, “Significance Determination Process (SDP).” Findings for which the SDP does not apply are indicated by “NO COLOR” or by the severity level of the applicable violation.

Reactor Safety

Initiating Event

- GREEN — On November 11, 2000, an instrument maintenance technician failed to follow a surveillance procedure which resulted in the inadvertent opening of the 3B electromatic relief valve. Failure to follow the procedure while performing the surveillance was considered a Non-Cited Violation of Technical Specifications.

  The event had minimal safety significance because the valve was closed within 10 seconds of opening, reactor pressure control responded as expected with no reactor level or pressure transients, and the 3B electromatic relief valve was returned to operable status prior to exceeding the allowed time per Technical Specifications (Section 1R14).

- GREEN — On November 30, 2000, a scram occurred on Unit 2 due to deficient configuration control while performing work on a switchyard circuit breaker. Substation construction personnel manipulated twelve additional switches without procedure guidance. These actions resulted in improper restoration of the circuit breaker which caused the scram.

  The inspectors reviewed this issue using the significance determination process and determined that this issue was of very low risk significance because all the required emergency core cooling systems and risk significant equipment were available. (Section 1R14).

Mitigating Systems

- GREEN On November 20, 2000, the fire marshal failed to conclude that the unannounced fire drill was unsatisfactory after the drill's expected response time was not met. The time was exceeded when a fire brigade member could not don his gear and the licensee requested an alternate fire brigade member. Failure to follow the procedure in evaluating the drill was considered a Non-Cited Violation of Technical Specifications.

  The risk significance of this issue was minimal due to the absence of an actual fire (Section 1R05).

Cross-Cutting Issues:

Human Performance

- NO COLOR The inspectors identified human performance errors that affected or had the potential to affect plant operations during this period. These errors represented a continuation of human performance problems across various station departments. The fire marshal failed to determine that the unannounced fire drill had failed. An instrument maintenance technician caused the inadvertent opening of an electromatic relief valve. A Unit 2 scram occurred due to poor configuration control during work on a switchyard breaker.

  Although each individual issue was low in risk significance, the incidents indicated a performance trend of problems with control, review, and performance of activities. (Section 4OA4).

Problem Identification and Resolution

- NO COLOR On November 8, 2000, the inspectors identified that a contaminated area posting was not properly established. The licensee's subsequent corrective actions identified other deficient postings. The licensee's corrective actions were considered inadequate because the inspectors subsequently identified an additional deficient posting.

  The risk significance of this issue was minimal because there was no spread of contamination during the period the deficient postings existed (40A2).

Report Details

Summary of Plant Status

Unit 2 began the period at full power operations. On November 18, 2000, Unit 2 reduced load to 650 MWe to repack the 2A feedwater regulating valve and replace the associated joucomatic solenoid valve. Unit 2 was returned to full power operations later that night. On November 30, 2000, the unit scrammed on a load rejection due to station personnel improperly configuring the plant while working on a breaker in the switchyard. The unit was returned to service on December 2, and full power operations were reached on December 4, 2000. During the forced outage, the licensee performed the following work: replacement of main steam isolation valve pushbuttons, replacement of all four main turbine control valve electro-hydraulic control accumulators, replacement of main generator alterex brush holder cartridges, and plugging of a number of condenser tubes.

Unit 3 began the period at full power operations. On November 17, 2000, Unit 3 conducted a fast load drop to 550 MWe and then increased to 650 MWe to allow engineering personnel to monitor the performance of the reactor recirculation pump seal pressure. Also, the licensee replaced the joucomatic solenoid on the B feedwater regulating valve. Unit 3 returned to full power operations the following day.

1. REACTOR SAFETY

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

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the licensee's cold weather preparations and response to adverse snow and icing conditions.

b. Issues and Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors selected a redundant or backup system (listed below) to an out-of-service or degraded train, reviewed documents to determine correct system lineup, and verified critical portions of the system configuration. Instrumentation valve configurations and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status. Control room switch positions for the systems were observed. Other conditions, such as adequacy of housekeeping, the absence of ignition sources, and proper labeling, were also evaluated.

Mitigating System Cornerstone

Unit 2/3 Standby Gas Treatment System
Unit 2 Standby Liquid Control System

b. Issues and Findings

There were no findings.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured plant areas important to safety to assess the material condition, operational lineup, and operational effectiveness of fire protection systems and features. The walkdown assessed the control of transient combustibles and ignition sources, fire detection systems, fire suppression systems, manual fire fighting equipment and capability, passive fire protection features (including fire doors), and compensatory measures. Also, the inspectors evaluated the station's performance with respect to an unannounced fire drill. The following two areas were walked down:

Mitigating System Cornerstone

Unit 2 Turbine Trackway	Fire Zone 8.2.5.2
Unit 2 Battery Room	Fire Zone 7.0.A.1

b. Issues and Findings

Fire Drill Sequence

On November 20, 2000, the inspectors observed an unannounced fire drill for a fire in the Unit 2 emergency diesel generator room. After the fire alarm was initiated, all fire brigade members (FBMs) immediately arrived and began donning the required protective gear. One FBM could not don his gear. As a result of this problem, the fire brigade leader (FBL) directed the safety officer (SO) to obtain the radwaste control room operator as the alternate FBM. In directing the alternate FBM to respond to the fire, the SO instructed the FBM to wait for a relief, perform a turnover, then respond to the fire location. These instructions were contrary to the station's fire alarm response procedure. As a result of these directions, the alternate FBM did not arrive to the fire location until approximately 22 minutes after initiation of the drill. The objectives of the drill required the FBMs to respond within 15 minutes. Because the alternate FBM's response time exceeded the specified drill response time objective, the inspectors concluded the drill had failed to meet its objectives.

Once the alternate FBM donned the gear, the FBL briefed the full brigade complement and the drill scenario was completed by extinguishing the fire, checking for injured personnel, and checking for reflash.

During the critique of the drill, the inspectors had the following observations:

Performance of Drill Evaluators During Critique

1) The FM and the drill evaluator (DE) did not fail the drill. The inspectors were concerned that both individuals failed to adequately evaluate the drill. The FM informed the inspectors that the FBL waited for the full brigade complement before responding to the fire. Therefore, the drill was satisfactory per Procedure OP-AA-201-003, "Fire Drill Performance," Revision 1, despite the equipment deficiency identified with one FBM having the wrong gear. Procedural Step 4.19 of OP-AA-201-003, states, "satisfactory drill performance is determined by the fire marshal by considering the overall performance of fire brigade response. Overall performance may be judged satisfactory even though individual performance and /or equipment deficiencies are identified." The inspectors disagreed with the fire marshal's interpretation of this step because this deficiency contributed to the drill not meeting the expected response time objective. The failure of the FM to determine that the drill was unsatisfactory is a violation of Procedure OP-AA-201-003.

2) The FM and the DE demonstrated a lack of a questioning attitude because the alternate FBM was not questioned on his excessive response time. The inspectors were concerned that the cause of the alternate FBM's excessive response time was not initially identified and could adversely affect the FBM's response to an actual fire.

3) During the critique, the alternate FBM stated that the SO instructed him to wait for his relief, perform a turnover, then respond to the fire location. Neither the FM, DE, or FBL challenged the SO on this guidance. The inspectors were concerned that FBMs left the critique with the misunderstanding of what constituted the proper response to a fire. Procedural Step 3.2 of OP-AA-201-003, requires that fire brigade members are responsible for responding in accordance with fire alarm response procedures. Dresden Operating Procedure, DOP 2000-25, "Radwaste Control Room Operators Action For Station Assembly or Fire Brigade Member Duties," Revision 4, Procedural Step G.4, requires if a fire alarm has sounded, then pick up turn out gear and report to the fire location. The inspectors were later informed that the FBL briefed the crew on proper fire response.

4) The licensee did not capture the incorrect guidance given by the SO in the station's corrective action program. Again, the inspectors were concerned that all deficiencies which contributed to the failure of the drill were not captured. Therefore, the deficiencies would not be corrected and could adversely affect the station's response to an actual fire. After prompting by the NRC, the licensee generated Condition Report #D2001-00090 on January 5, 2001, which addressed this concern and documented that the FBL remediated the crew after the critique. Also, the operations manager sampled a number of FBLs to verify they understood proper fire response. In addition, the licensee verified that each FBM had the proper gear.

Dresden Technical Specification 6.8.A.1 states that procedures shall be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33, states, in part, that activities for implementing the fire protection program should be covered by procedures. Procedure Step 4.19 of OP-AA-201-003, requires the fire marshal to assess the drill.

The failure of the fire marshal to adequately assess the drill as required by OP-AA-201-003 is a violation of Technical Specification 6.8.A.1. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-237/00-21-01 (DRP))**.

<u>Station Assessment of Drill</u>

The operations manager ultimately determined that the drill was unsatisfactory after discussing the problems experienced during the drill with the fire marshall.

<u>Significance Determination Process</u>

The safety significance of this issue was minimal due to the absence of an actual fire. Therefore, these issues screened out as very low safety significance (GREEN) during the phase 1 evaluation. These issues were documented in Condition Reports D2000-06316, 06580, and D2001-00090.

1R11 <u>Licensed Operator Requalification (71111.11)</u>

a. <u>Inspection Scope</u>

The inspectors observed and assessed the performance of operators in the control room and in the simulator to identify deficiencies and discrepancies in performance and training. The inspectors assessed the performance of operating crew #3 in the simulator on December 7, 2000, for AE-P1. The scenario included an intermediate range monitor failure with a partial half scram actuation, loss of bus 24, anticipated transient without scram, loss of feedwater, and a loss of coolant accident in the drywell.

b. <u>Issues and Findings</u>

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors assessed the licensee's implementation of the maintenance rule by determining if systems were properly scoped within the maintenance rule.

The following systems were reviewed during the inspection period:

Mitigating System Cornerstone

Unit 3 Average Power Range Monitors
Unit 2 Containment Cooling Service Water Pumps

b. Issues and Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified how the licensee managed the risk. The inspectors also evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities.

The following risk significant activities were evaluated:

Mitigating Systems Cornerstone:

WR#990178028	Repair Fuel Oil Leak Downstream of the Unit 2/3 Diesel Generator Fuel Oil Priming Pump Discharge Check Valve
WR#990024719	2E Drywell Cooler Exhaust Damper Repair/Closure
WR#990037753	Repair/Replace 3A Low Pressure Coolant Pump Differential Pressure Switch (Failed Surveillance)
WR#990003115	Unit 2 Diagnostic VOTES test of 2-1501-13A
WR#990047447	Unit 2 4-year Preventive Maintenance of 4KV Bus 24-1 to 2B Core Spray Pump
WR#990218901	Unit 2 Repair of 2B Core Spray Lower Oil Bath Plug
WR#980124960	Unit 3 Replace ASCO Solenoid Valve on 3-1601-20B
WR#980132933	Unit 3 Replace ASCO Solenoid Valve on 3-1601-20A

WR#990233847	Unit 2/3 Repair Control Room/Reactor Building dP Indication with Standby Gas Treatment System (SBGT) Running
AR#9990116938	Unit 3 Replace Missing Pipe Support on Unit 3 Scram Air Header

b. Issues and Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors reviewed operator logs, problem identification forms, the sequence of events recorder, and alarm printer outputs associated with the inadvertent opening of an electromatic relief valve event on November 11, 2000 and a scram that occurred on November 30, 2000. The inspectors interviewed operations, maintenance, and engineering personnel concerning the cause of the failure and the resulting sequence of events.

b. Issues and Findings

.1 Inadvertent Opening of the 3B Electromatic Relief Valve

On November 11, 2000, during the performance of Dresden Instrument Surveillance DIS 0250-03, "Electromatic Relief Valve/Target Rock Valve Pressure Switches Calibration without Control Switch Functional Testing," Revision 31, a spurious opening of the 3B Electromatic Relief Valve (ERV) occurred while the reactor was at 100 percent power. The 3B ERV opened for approximately 10 seconds until closed by the Unit 3 nuclear station operator by placing the 3B ERV control switch in the OFF position. The plant responded as designed without any reactor pressure or level transients. The licensee terminated the work activity and initiated an investigation of the event.

The licensee's investigation of this event determined that while performing the surveillance an instrument maintenance technician performed procedural steps out of sequence. The technician connected a digital multimeter (DMM) in the OHM Mode to the 3B ERV pressure controller prior to the nuclear station operator placing the 3B ERV control switch to the OFF position which made up the valve's opening circuitry. Category 1 Procedure DIS 0250-03, Step I.3.c requires the technician to request the nuclear station operator to place the control switch for the ERV to be tested in the OFF position. DIS 0250-03, Step I.5.l(1) requires the technician to verify that the control switch for the ERV being tested is in the OFF position. The subsequent Step I.5.l(2) requires the technician to connect the DMM set to monitor resistance. However, the technician connected the DMM prior to verifying that the control switch for the 3B ERV was in the OFF position. Administrative procedure AD-AA-104-101, "Procedure Use and Adherence," Revision 0, Step 4.3.2 requires that all numbered steps in Category 1 procedures be performed in sequence. The failure of the technician to perform steps in sequence in DIS 0250-03 as specified by AD-AA-104-101 is a violation.

Dresden Technical Specification 6.8.A.1, states that procedures shall be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33, states, in part, that surveillance and calibration tests are typical safety-related activities that should be covered by procedures. Procedural Step 4.3.2 of AD-AA-104-101, requires that all numbered steps in Category 1 procedures be performed in sequence. The failure to perform procedural steps of DIS 0250-03 in sequence as required by AD-AA-104-101 is a violation of Technical Specification 6.8.A.1. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-249/00-21-02(DRP))**.

Significance Determination Process

Because the valve was closed within 10 seconds of opening and reactor pressure control responded as expected with no reactor level or pressure transients, the inspectors concluded that this issue was of very low safety significance (GREEN). The licensee documented this event in Condition Report # D2000-06116.

.2 Poor Plant Configuration Control Resulted In a Turbine and Reactor Trip

On November 30, 2000, Dresden Unit 2 scrammed from 100 percent power due to a main generator load reject which was followed by a turbine trip. The main generator load reject occurred during the return to service of switchyard circuit breaker OCB 2-7. During the transient all emergency core cooling systems remained available. However, following the transient, several components did not respond per design. Specifically, five control rod drives lost full core display position indications. The operators were able to verify that the control rods had fully inserted using other methods. Also, the operators received indication that two intermediate range monitors (13 and 14) and one source range monitor (22) did not fully insert into the core. Because intermediate range monitor 11 was rendered inoperable prior to the scram, this significantly delayed the operator in resetting the scram signal. Prior to restart of the unit, the licensee identified and repaired a limit switch problem with the intermediate and source range monitors and entered the position indication display problem into the corrective action program. The licensee initiated an investigation into the cause of the scram.

The licensee's preliminary investigation determined that the circuit breaker was taken out of service to perform planned maintenance. At the start of the evolution, substation construction personnel opened knife switches located in the upper portion of the circuit breaker cabinet as specified by procedure. Subsequently, substation construction personnel opened 12 current transformer cutout knife switches which were not part of the approved work instruction. As a result, the circuit breaker was returned to service with the knife switches inappropriately positioned. When the circuit breaker was closed, a high differential fault was sensed on the ring bus which caused OCB circuit breaker 2-3 to trip. When this breaker tripped, a phase differential on the ring bus was sensed which led to a load reject on the main generator and a turbine trip.

The licensee concluded that the cause of this event was the failure of maintenance personnel to maintain proper plant configuration control during work activities and improper verification practices. Additionally, the licensee failed to perform a post maintenance test. Had a post maintenance test been performed, the mispositioned switches may have been detected and the scram avoided.

Significance Determination Process

The inspectors reviewed the risk significance of this initiating event for both units using the Significance Determination Process. All mitigating equipment was available for Unit 2 following the trip, and this event was screened as having very low risk significance (GREEN). This event was documented in condition report #D2000-06464.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following post maintenance test results to confirm that the tests were adequate for the scope of the maintenance being performed, and that the test data met the acceptance criteria.

Mitigating Systems Cornerstone:

WR#990207341 01	Repair of the 2/3 Emergency Diesel Generator Fuel Line Coupler
WR#990236545 01	Intermediate Range Monitor #15 Failure
WR#990236551 01	Installed Jumper to Allow Intermediate Range Monitor 11 to Clear Rod Block
WR#990217217 01	Average Power Range Monitor #4 Joystick Replacement
WR#990217217 02	Average Power Range Monitor #4 K23 Relay Replacement

Initiating Event Cornerstone:

WR#990192422	3B Generator Starter Water Cooler Repairs

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment. The inspectors assessed whether the selected plant equipment could perform its intended safety functions and satisfy the requirements contained in Technical Specifications.

Mitigating System Cornerstone

WR#990186745	Unit 2A Low Pressure Coolant Injection Pump In Service Test, DOS 1500-10
WR#990198768	Unit 3 High Pressure Coolant Injection System, DOS 2300-03
WR#990005448	Division 1 Low Pressure Coolant Injection Containment Cooling System Logic System Functional Test, DIS 1500-27
WR#99022933	Electromatic Relief Valve/Target Rock Valve Pressure Switches Calibration Without Control Switch Functional Testing, DIS 0250-03
WR#990123385	High Pressure Coolant Injection Gland Seal Condenser Level Control/Alarm Switch Inspection and Functional Test, DIS 2300-15
WR#990003115-01	Differential Pressure Test of Unit 2 Low Pressure Coolant Injection Mini-flow Valve, 1501-13A, DEP 0040-38
WR 990228536	Unit 2/3 Monthly SBGT Surveillance and IST per DOS 7500-02
WR 990060625	Unit 2/3 SBGT Charcoal Sample per DTS 7500-07
WR 990056758	Unit 2/3 18-month Visual Inspection of 2/3 SBGT Filter Train B per DTS 7500-13
WR 990062670	Unit 2/3 4-year EQ Surveillance of Group B SBGT Discharge Valve per DMS 0040-02
WR 990062668	Unit 2/3 4-year EQ Surveillance of Group B SBGT Inlet Motor Operated Valve per DMS 0040-02
WR 990229969	Unit 2/3 Diesel Generator Monthly Operability Surveillance per DOS 6600-01
WR 990229973	Unit 2/3 Diesel Generator Lube Oil Sample from Crankcase per DOS 0040-04

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors screened active temporary modifications on systems ranked high in risk and assessed the effect of temporary modifications on safety-related systems.

Mitigating System Cornerstone:

#9900736	Cross-tie the Unit 3 Station Blackout Inverter to the Unit 2 Uninterruptable Power System Panel, Revision 0
#9900838	Blocked Open Unit 3 Reactor Recirculation Pump Motor Generator Set Ventilation Exhaust Damper 3-5772-20
#990796	Temporary Jumper of Unit 3 125Vdc Alternate Battery Cell No. 34

Initiating Event Cornerstone:

#9900699	Installed Gags on Shutdown Drain Volume Relief Valves 3-0301-158A & B

b. Issues and Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114-04)

a. Inspection Scope

The inspector reviewed Revision 9 to Sections 1.0, 4, 8,and 9, of the Generating Stations Emergency Plan, which was submitted by letter dated April 26, 2000, in order to determine whether the changes in Revision 9 might decrease the plan’s effectiveness pending future inspection of the implementation of these changes. This emergency plan revision was submitted in accordance with 10 CFR 50.54(q).

b. Issues and Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety [OS] Cornerstone:

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors walked down radiological postings to determine that the licensee had properly posted appropriate areas of the plant.

b. Issues and Findings

Inadequate Posting of Contaminated Area

On November 8, the inspectors identified a deficient contaminated area posting on the Unit 3 source range monitor/intermediate range monitor (SRM/IRM) landing in the Unit 3 reactor building. The posting was deficient because the contaminated area posting was missing. Specifically, the licensee had placed a contaminated area sign on a handrail. The sign specified that contamination existed behind the hand rail. However, the inspectors determined that approximately 30 inches of the hand rail had been removed which resulted in the accessability of a contaminated area. The inspectors informed RP personnel of this deficiency. The licensee immediately corrected the contaminated area posting by placing a stanchion, with a contaminated area sign, where the handrail was missing and initiated a condition report. Dresden Radiation Procedure, DRP 5010-01, “Radiation Posting and Labeling Requirements,” Revision 9, Step G.3.f, requires if an area has smearable contamination present at levels greater than or equal to 1000 dpm/100 cm^2 beta-gamma or 20 dpm/100 cm^2 alpha, then the area should be posted with a sign that states “caution, contaminated area.” The licensee’s failure to adequately post the Unit 3 SRM/IRM landing contaminated area was a violation of DRP 5010-01. By itself, this violation was minor and would not ordinarily be documented in the report.

The RP department performed an extent of condition review of radiological postings on November 13, 2000, and identified two additional deficient areas. The licensee subsequently placed the required posting in those areas. However, on December 5, 2000, the inspectors identified another deficient contaminated area posting on the Unit 2 SRM/IRM landing. This area was missed by RP personnel even though this area was identified on the SRM/IRM monthly survey. Therefore, the inspectors considered the licensee’s initial resolution to this issue to be inadequate. The licensee corrected the posting and initiated condition report #D2000-06563. The licensee subsequently performed an extent of condition review of all station postings and identified additional improvement opportunities.

Significance Determination Process

The inspectors determined that there was no spread of contamination during the time the postings were deficient, therefore, the individual posting deficiencies were considered to be of very low safety significance. These issues were documented in condition reports D2000-06119, 06563 and D2001-00050. The inadequate problem identification and resolution to ensure radiological postings were appropriate was considered a (NO COLOR) cross-cutting issue finding.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed a sample of plant records and data against the reported performance indicators in order to determine the accuracy of the indicators.

Initiating Events System Cornerstone

Unit 2 and Unit 3 Reactor Scrams With Loss of Normal Heat Removal Performance Indicator Verification (November 1999 through June 2000).

Unit 2 and Unit 3 Unplanned Power Changes (November 1999 through June 2000).

b. Issues and Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

As described in 2OS1 of this report, the inspectors identified an area of deficient radiological posting. Licensee corrective action efforts discovered additional examples of deficient postings. These efforts were considered inadequate, however, because the inspectors subsequently found an additional posting deficiency on the other unit in the same area as the original deficiency.

4OA3 Event Follow-up (71153)

a. Inspection Scope

.1 (Closed) Unresolved Item 50-237-98007-02 (DRS); 50-249-98007-02 (DRS): Evaluation of containment cooling service water (CCSW) pump performance with vent holes cut in the suction piping and the CCSW suction bay stoplogs in place.

The inspectors had two concerns. The first concern was that the CCSW suction piping vent holes at the 501 feet elevation may not have adequate submergence after a hypothetical dam failure. The licensee changed Dresden Operating Abnormal Procedure DOA 0010-01, “Dresden Lock and Dam Failure,” Revision 9, to require that the CCSW bay level be restored to 505 feet elevation instead of 501.5 feet elevation. The inspector determined that the submergence would be adequate at the 505 feet elevation. The inspector based the determination on the Hydraulic Institute Standards, ANSI/HI,1994 edition, Figure 1.66, which recommended 2.5 feet of submergence for a pump with a rated capacity of 3,600 gpm. Each CCSW pump has a capacity of 3,600 gpm. This concern was resolved because the submergence would be 4 feet. The second concern was that the “stoplog” made out of rough-hewn logs might leak excessively and prevent reflooding the CCSW intake bay. The licensee replaced the wooden “stoplog” with a metal “stoplog” that has a gasket seal surface similar to the other metal “stoplog.” This unresolved item is closed.

.2 (Closed) Inspection Followup Item 50-237-98007-03 (DRS); 50-249-98007-03 (DRS): Evaluation of the acceptability of a two-hour delay and reduced performance of the Unit 2/3 Diesel Fire Pump.

This item tracked the inspector’s concern that the Unit 2/3 Diesel Fire Pump (DFP) would not promptly provide makeup water to the isolation condenser after a postulated failure of the Dresden Dam. This concern resulted when the inspector determined that the intake water level after a dam failure would be too low to allow the Unit 2/3 DFP to promptly provide isolation condenser makeup water. The licensee changed the design analysis to take credit for the Unit 1 DFP promptly providing makeup water to the isolation condenser. Region III personnel consulted with Nuclear Reactor Regulatory personnel and consensus was made that the licensee could take credit for the Unit 1 DFP. The inspector verified that the Unit 1 DFP suction was below 492 feet elevation and would promptly provide isolation condenser makeup water if a dam failure reduced the intake water level to the 495 feet elevation. This inspection followup item is closed.

4OA4 Human Performance Issues

a. Inspection Scope

The inspectors reviewed human performance associated with several events and issues.

b. Issues and Findings

The inspectors identified human performance errors that affected or had the potential to affect plant operations during this period. These errors represented a continuation of human performance errors by station personnel. The fire marshal failed to determine that a fire drill had failed (See Section 1R05). An instrument maintenance technician caused the inadvertent opening on an electromatic relief valve (See Section 1R14). A Unit 2 scram occurred due to poor configuration control during work in the switchyard (See Section 1R14).

Although each individual issue was of very low risk significance, the incidents indicated a performance trend of problems with control, review, and performance of activities.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Fisher and other members of licensee management at the conclusion of the inspection on January 4, 2001. The licensee acknowledged the findings presented. No proprietary information was identified. On January 26, 2001, the inspectors again met with licensee management to discuss a change in the NRC’s characterization of the radiological posting issue.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Ambler, Regulatory Assurance Manager
P. Boyle, Chemistry Manager
P. Chabot, Site Engineering Manager
R. Fisher, Station Manager
A. Haeger, NLA - Regulatory Services
B. Hanson, Shift Operations Superintendent
J. Harlach, Industrial Safety and Hygiene Advisor
R. Kelly, NRC Coordinator
W. Liscomb, Training Manager
J. Moser, Radiation Protection Manager
M. Pacilio, Operations Manager
R. Peak, Design Engineering Manager
M. Riegel, Acting Nuclear Oversight Manager
R. Whalen, System Engineering Manager

NRC

B. Dickson, Dresden Resident Inspector
M. Ring, Branch Chief
D. Smith, Dresden Senior Resident Inspector
R. Jickling, Emergency Preparedness Analyst
R. Lerch, Project Engineer
P. Pelke, Reactor Engineer
G. O'Dwyer, Reactor Inspector

IDNS

R. Zuffa, Illinois Department of Nuclear Safety

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/00-21-01	NCV	failure of the fire marshal to properly evaluate fire drill.
50-249/00-21-02	NCV	inadvertent opening of an emergency relief valve due to a technician's failure to follow procedure.

Closed

50-237; 249/98007-02	URI	evaluation of CCSW pump performance with vent holes cut in the suction piping and the CCSW suction bay stoplogs in place.
50-237; 249/98007-03	IFI	evaluation of the acceptability of a two-hour delay and reduced performance of the Unit 2/3 diesel fire pump.
50-237/00-21-01	NCV	failure of the fire marshal to properly evaluate fire drill.
50-249/00-21-02	NCV	inadvertent opening of an emergency relief valve due to a technician's failure to follow procedure.

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 Number	Inspection Procedure Title	Report Section
71111-01	Adverse Weather Preparations	1R01
71111-04	Equipment Alignment	
71111-05	Fire Protection	1R05
71111-11	Licensed Operator Requalification	1R11
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk and Emergency Work	1R13
71111-14	Non-routine Evolutions	1R14
71111-19	Post Maintenance Testing	1R19
71111-22	Surveillance Testing	1R22
71111-23	Temporary Modifications	
71114-04	Emergency Action Level and Emergency Plan Changes	1EP4
71151	Performance Indicator Verification	4OA2
71121-01	Access Control to Radiologically Significant Areas	1OS1
71153	Event Followup	4OA3
(none)	Other	4OA4
(none)	Management Meetings	4OA5

LIST OF ACRONYMS USED

CCSW	Containment cooling service water
CR	Condition Report
DE	Drill Evaluator
DFP	Diesel Fire Pump
DMM	Digital Multi Meter
ERV	Electromatic Relief Valve
FAQ	Frequently Asked Question
FBL	Fire Brigade Leader
FBM	Fire Brigade Member
FM	Fire Marshall
IDNS	Illinois Department of Nuclear Safety
IFI	Inspection Followup Item
NCV	Non-Cited Violation
RP	Radiation Protection
SBGT	Standby Gas Treatment System
SDP	Significance Determination Process
SO	Safety Officer
SRM/IRM	Source Range Monitor/Intermediate Range Monitor
URI	Unresolved Item
VIO	Violation
WR	Work Request