October 26, 2000

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

SUBJECT: DRESDEN INSPECTION REPORT 50-237/00-13(DRP); 50-249/00-13(DRP)

Dear Mr. Kingsley:

On September 30, 2000, the NRC completed an inspection at Dresden Units 2 and 3. The enclosed report presents the results of that inspection. The NRC discussed the results on September 26, 2000, with Mr. Fisher and other members of your staff.

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.

Based on the results of this inspection, the NRC identified one issue involving several human performance problems for which no risk significance or color was assigned. Additionally, the NRC identified five issues that were each evaluated under the significance determination process and determined to be of very low safety significance (GREEN). The first issue involved instrument mechanics removing the wrong fuse and causing a half scram. The second incident related to a materials engineer specifying the wrong part for installation into a breaker. The third issue related to a pipe replacement modification to the high pressure coolant injection system. The fourth issue involved the licensee's failure to use the design control process when changing the tap setting on an undervoltage relay. The fifth issue was related to the local leak rate test failures of several valves which required notification to the NRC. These issues have been entered into your corrective action program, and are discussed in the summary of findings and in the body of the enclosed inspection report. Issues one through four were determined to involve violations of NRC requirements. These violations are being treated as non-cited violations (NCVs), consistent with Section VI.A.1 of the Enforcement Policy. The NCVs are described in the subject inspection report. If you contest these non-cited violations, 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 Dresden Nuclear Power 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 (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/

Mark Ring, Chief
Reactor Projects Branch 1

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

Enclosure: Inspection Report 50-237/00-13(DRP);
50-249/00-13(DRP)

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REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	50-237/00-13(DRP); 50-249/00-13(DRP)
Licensee:	Commonwealth Edison Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	August 11, 2000, through September 30, 2000
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector E. Duncan, Senior Resident Inspector, LaSalle Station P. Louden, Senior Resident Inspector, Clinton Station D. Roth, Resident Inspector 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 05000237-00-13(DRP); 05000249-00-13(DRP); on 8/11 - 9/30, 2000; Commonwealth Edison Company; Dresden Nuclear Power Plant; Units 2 and 3. Maintenance Work Prioritization, Operability Evaluations, Outage and Refueling Activities, and Other Activities.

The inspection covered a 7-week period of resident inspections. The inspection identified five green issues, four of which were non-Cited Violations. The inspectors also identified a cross-cutting issue with no color. The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Initiating Events

- GREEN. On August 28, 2000, an instrument mechanic removed a fuse from the wrong reactor protection system motor-generator bus causing an unexpected half scram. The mechanic's failure to follow the procedure while performing this work was considered a Non-cited Violation (NCV) of Technical Specifications.

  The inspectors considered this event to be of very low safety significance because no actual loss of safety function occurred. (Section 1R13.1).

- GREEN. On May 8, 2000, a materials engineer specified an incorrect contactor coil to be installed in the reactor protection system motor generator set feed breaker. During subsequent testing on September 2, 2000, the breaker tripped and the breaker cubicle was damaged. The failure to properly implement design control measures to ensure installation of the correct contactor was considered a NCV of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

  The inspectors reviewed this issue using the significance determination process and determined that this event was of very low risk significance because failure of the contactor coil would not have prevented the motor-generator set from performing its safety function. (Section 1R13.2).

Mitigating Systems

- GREEN. On September 22, 2000, the inspectors' review of an operability evaluation for pipe replacement on the high pressure coolant injection system identified that the licensee had performed an inadequate like-for-like replacement parts evaluation in that yield strength differences were not considered. The failure to properly implement design control measures to ensure the equivalent replacement of safety-related piping was considered a NCV of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

The inspectors reviewed this issue using the significance determination process and determined that this event was of very low risk significance because the strength difference did not significantly change the seismic calculations and the operability of the high pressure coolant injection system was not impacted. (Section 1R15).

Containment Barrier

- GREEN. On September 27, 2000, the licensee discovered that the overvoltage coil tap setting for the undervoltage relays on Bus 34 (4160 KV) had been changed without using the station's design control process. This error would have resulted in the loss of the 'C' and 'D' containment cooling service water pumps during a loss of offsite power transient. The licensee's failure to implement sufficient design control measures for changing the overvoltage coil tap setting was considered a NCV of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

  The safety significance of this issue was minimal due to the availability of the 'A' and 'B' containment cooling service water pumps. (Section 1R20b.1).

- GREEN. The licensee determined that the 'A' feedwater header check valves and both the 'B' inboard and outboard main steam isolation valves had each failed local leak rate testing. The licensee reported the failures to the NRC on September 19 and 20, respectively.

  The inspectors reviewed this issue using the containment integrity significance determination process and determined that these failures were of very low risk significance because even though the valves leaked, they did shut, and therefore, did not constitute a large early release pathway. (Section 1R20.2).

Cross-Cutting Issues; Human Performance

- NO COLOR. The inspectors determined that several recent events and issues which affected plant operations were attributed to deficient human performance. An apparent adverse trend in human performance was evidenced by the following incidents. Instrument mechanics removed the incorrect fuse from the reactor protection system. A materials engineer specified the incorrect contactor for installation in the reactor protection system breaker. A materials engineer performed an inadequate parts evaluation during a high pressure coolant injection system piping modification. The license failed to use the design control process when changing the overvoltage tap settings on the undervoltage relay.

  While the risk of the individual events was very low, the number of incidents indicated a performance trend of problems with control, review, and performance of maintenance related activities. (Section 4OA4)

Report Details

Summary of Plant Status

Unit 2 began this period at full power. On September 24, 2000, the operators decreased power 720 MWe for turbine control valve testing. The unit was returned to full power operations later that day.

Unit 3 began this period at full power. On September 15, 2000, the unit was shut down to commence the beginning of the sixteenth refueling outage. Major work planned for the outage included the replacement of 20 condenser bellow assemblies, improvement to the Unit 3 electro-hydraulic control system, and upgrades to the moisture-separator vanes.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

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. Operational status of support systems was verified by direct observation of various parameters. 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 Containment Cooling Service Water System
Unit 2 Low Pressure Coolant Injection System
Unit 3 Shutdown Cooling System
Unit 2/3 Emergency Diesel Generator System
Unit 2 and 3 Main Control Room,
Unit 2 and 3 Emergency Diesel Generator Rooms
Unit 2 and 3 Essential 4160 Volt Switchgear
Unit 2 and 3 250 V dc Battery

b. Issues and Findings

There were no findings identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured plant areas important to safety to assess the material condition of fire protection systems and features, and the operational lineup and operational effectiveness. The review included 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. The tours included the following areas:

Mitigating System Cornerstone

Unit 2 High Pressure Coolant Injection System Pump Rooms ---- fire zones 11.2.3
Unit 3 High Pressure Coolant Injection System Pump Rooms ---- fire zones 11.1.3
Secondary Containment (Reactor Building) ----fire zone 1.2.1
125 volt Battery Room ---- fire zone 7.0.A.2
250 volt Battery Room ---- fire zone 7.0.A.3
Unit 3 Shutdown Cooling Pump Room ---fire zone 1.3.1
Unit 3 Shutdown Cooling Heat Exchanger Room ---- fire zone 1.1.1.3
Unit 2/3 Emergency Diesel Generator Room ----fire zone 9.0.C

Documents reviewed included the fire hazard analysis, Dresden Safe Shutdown Procedure 0100-E, Rev. 16, "Hot Shutdown Procedure - Path E," Dresden Safe Shutdown Procedure 0010-01, Rev. 08, "Determining Safe Shutdown Paths for Extensive Plant Damage," Dresden Safe Shutdown Procedure 0100-F, Rev. 15, "Hot Shutdown Procedure - Path F," and Chapter 9 of the Dresden Updated Final Safety Analysis Report.

b. Issues and Findings

There were no findings identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

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 crews during dynamic training exercises on August 28, 2000. The inspectors also assessed licensed operator performance for operating crew #5 during requalification simulator dynamics on August 31, 2000, and the evaluators' critiques following the dynamic examination. The scenarios included the following:

* Loss of bus 21 and the Unit 2 drywell pneumatics followed by an anticipated transient without a Scram (ATWS), and

* Loss of the ATWS level transmitter coincident with a loss of motor control center 29-1 and 29-9 followed by an unisolable steam leak from the high pressure coolant injection system steam supply and a damaged fuel element.

The inspectors also compared simulator board configurations with actual control room board configuration for consistency, especially with recent modifications implemented in the control room.

b. Issues and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements, including a review of scoping, goal setting, performance monitoring, short-term and long-term corrective actions, and current equipment performance status. The following systems were reviewed during the inspection period:

Mitigating System Cornerstone

Unit 2 and Unit 3 Primary Containment System
Unit 2 and Unit 3 Containment Cooling Service Water System
Unit 2 and Unit 3 Low Pressure Coolant Injection System
Unit 2 Core Spray System

Initiating Event Cornerstone

Unit 2 and Unit 3 Turbine Building Closed Cooling Water System

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Work Prioritization & Control (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 verified that, upon identification of an unforeseen situation, the licensee had taken the necessary steps to plan and control the resulting emergent work activities. The inspectors also verified that the licensee adequately identified and resolved maintenance risk assessments and emergent work problems.

Documents reviewed included WC-AA-103, Revision 2, "On-Line Maintenance," and WC-AA-104, Revision 2, "Review and Screening for High Production Risk Activities and Work Authorization."

Mitigating Systems Cornerstone

2A Low Pressure Coolant Injection/Containment Cooling Service Water Heat Exchanger Tube Replacement
WR 990203811, Repair of 1-inch opening in Reactor Building Ventilation Piping
WR 990210733, Replace Charging Water Inlet Valve on HCU 46-19
WR 990016485, Clean/Inspect/Hydro/Eddy Current Test 3B LPCI Heat Ex
WR 990202253, 2B LPCI Heat Exchanger Tube Leak Repair
WR 990095844, Replace Scram Discharge Volume Level Switch with Magnitrol

Initiating Event Cornerstone:

* 18-Month Planned Maintenance of the 3A Motor Generator Set (WR #990008955)

b. Issues and Findings

.1 Incorrect Fuse Removal

On August 28, 2000, while instrument mechanics were performing Dresden Technical Surveillance (DTS) 500-01, “Calibration of Reactor System Motor Generator Set Relays” Revision 02, on the Unit 3 “B” reactor protection system bus under Work Request (WR) #990008955-01, an unexpected half-scram was received in the main control room.

An investigation into this event by the licensee concluded that the unexpected half-scram occurred because a technician incorrectly removed a control power fuse from the “A” reactor protection system while performing the procedure. Step 12.a of DTS 500-01, specified the removal of fuse UA in panel 903-52A. However, the technician opened the panel door for 903-53B and pulled the UA fuse. This action resulted in the receipt of a half-scram on Unit 3 and a trip of the Unit 3 reactor water cleanup system.

The inspectors concluded that the technician failed to follow the procedure when he opened the wrong cabinet and removed the wrong fuse. Dresden Technical Specification (TS) 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.

Contrary to the above, on August 28, 2000, the technician incorrectly removed the UA fuse from panel 903-53B and an unexpected half scram occurred. This was a violation. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-249/00-013-01(DRP))**. The inspectors used the significance determination process (SDP) analysis to determine the safety significance of the event, and concluded that the finding did not represent an actual loss of safety function. The event was considered to be of very low safety significance (GREEN). The licensee documented this event in Condition Report #D2000-04713.

.2 Inadequate Parts Review

On September 2, 2000, a second human performance error resulted in the tripping of the 3A motor generator set feed breaker following planned maintenance activities under WR #990008955. When the operators attempted to start the 3A motor generator set by turning the control switch to the close position, the motor came up to speed then tripped before the control switch was released. The licensee inspected the breaker and initiated an investigation.

The licensee's inspection of the feed breaker at motor control center 38-3 found that the starting relay and the contactor were severely damaged. An investigation into this issue revealed that a 120 VAC contactor coil had been installed in the starting circuitry instead of the required 480 VAC contactor coil.

In addition, the licensee determined that the incorrect contactor installation was due to a materials engineer having done an inadequate part evaluation while preparing the work package. This inadequate review was not detected during the licensee's work package review and approval process.

Criterion III, Design Control, Appendix B, 10 Code of Federal Regulations, Part 50, states, "Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety related functions of the structures, systems and components."

Contrary to the above, on September 2, 2000, the licensee incorrectly installed a 120 VAC contactor coil instead of the 480 VAC contactor in the starting circuitry. The licensee's failure to ensure sufficient design control measures were in place, when replacing the contactor is a violation. This violation is being treated as an example of a non-cited violation (NCV), consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-237/00-013-02(DRP))**. The inspectors used the SDP to determine the safety significance of the event, and concluded that the finding did not represent an actual loss of safety function. The event was considered to be of very low safety significance (GREEN). The licensee documented this event in Condition Report #D2000-04810.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability evaluations (OEs) of degraded and non-conforming conditions affecting mitigating systems and barrier integrity to ensure that operability was properly justified and the component remained available, such that no unrecognized increase in risk had occurred. The following operability evaluations were reviewed:

Mitigating System Cornerstone

* Operability Evaluation #00-025 Small Bore Safety-Related Carbon Steel Piping Replaced With Chrome Moly

* Operability Evaluation associated with Condition Report #D2000-04806, "Drywell Average Temperature Exceeds Updated Final Safety Analysis Report Maximum," dated September 1, 2000.

* Operability Evaluation #00-038 Tube Leakage Unit 2 LPCI/CCSW "B" heat exchanger.

b. Issues and Findings

Operability Evaluation #00-025

The subject operability evaluation reviewed the identified condition that substitution of chrome moly piping for carbon steel piping in safety-related applications may not meet American Society of Mechanical Engineers (ASME) Code requirements. The inspectors reviewed the subject operability evaluation; Specification K-4080, "General Work Specification for Maintenance/Modification Work"; ComEd Letter 5289459, "Revision to Specification K-4080," dated December 17, 1996; ComEd Letter 5195908, "Evaluate Substitution of Chrome-Moly Material for Carbon Steel Piping," dated October 1, 1996; ComEd Letter 5196432, "Revision to Spec K-4080," dated October 1, 1996; 10 CFR 50.59 Safety Evaluation DFL-96132, "Update Updated Final Safety Analysis Report Tables 3.9-9, 3.9-7, 5.2-2, 6.1-1, 10.3-1, dated December 5, 1996; and Problem Identification Form (PIF) D2000-02489, "Specification K-4080 Chrome Moly Substitution for Carbon Steel," dated April 24, 2000.

To address this issue, the licensee completed OE 00-025 which concluded that in accordance with NES-MS-03.2, "Evaluation of Discrepant Piping and Support Systems," the installed configuration was acceptable in the near term. The inspectors reviewed this operability evaluation. No issues were identified. As part of their long-term corrective actions, action tracking item #28123 was generated to review and evaluate all safety-related carbon steel piping that was replaced with chrome moly as allowed per Specification K-4080 for the acceptability with respect to the ASME Code yield stress allowable. The licensee also planned to review the Updated Final Safety Analysis Report and determine if any changes were required.

The inspectors also reviewed documentation regarding WR #960034237 which controlled the replacement of the carbon steel Unit 3 High Pressure Coolant Injection (HPCI) drain pot line with chrome moly. In particular, the inspectors reviewed ComEd letter 5325911, "Dresden Unit 3, HPCI Drain Pot Line Replacement," dated February 5, 1997; and Alternate Replacement Evaluation D-1996-15-1, "HPCI Drain Pot Line Alternate Replacement," dated October 1, 1996. The inspectors identified that the replacement of the drain pot line was controlled as a like-for-like replacement activity and that the evaluation stated that all critical characteristics of the chrome moly piping was identical to that of carbon steel. The inspectors reviewed this evaluation, and identified that since the chrome moly had a minimum yield strength which was less than that of the carbon steel, the replacement activity should have been controlled as a design change instead of a like-for-like replacement.

10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires that measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of structures, systems, and components. The failure to adequately review the replacement of carbon steel piping in the HPCI drain pot line with chrome moly piping was an example where the requirements of 10 CFR Part 50, Appendix B, Criterion III, were not met and was a violation. However, this violation is being treated as an example of a non-cited violation (NCV), consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-237-00-13-03; 50-249-00-13-03(DRP))**. This issue was captured in the licensee’s corrective action program in Condition Report #D2000-05612.

The inspectors assessed this issue using the NRC’s Significance Determination Process (SDP). Since the difference in minimum yield strength was not significant and only impacted seismic calculations, the operability of the system was not impacted. As a result, this issue had very low risk significance (GREEN).

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed the licensee’s operator work-arounds that were open on August 23, 2000, to assess any potential effect on the functionality of mitigating systems. The inspectors reviewed ComEd procedure OP-AA-101-303, “Operator Work-Around Program,” Revision 0. Below are examples of the work-arounds:

#2-99-02	2-1501-21A/B Valve timer
#3-99-04	Leak by the 3A feedwater regulating valve

b. Issues and Findings

There were no findings associated with this inspection activity.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

Mitigating Systems Cornerstone

* Unit 3 Low Pressure Coolant Injection System 18A valve (torus spray)

b. Issues and Findings

There were no findings identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed and evaluated several outage activities during the ongoing refueling outage for Unit 3 to ensure that the licensee appropriately considered risk factors during the development and execution of planned activities.

The following is a list of the activities the inspectors reviewed:

* Shutdown of Unit 3
* Alternate Decay Heat Removal Evolution
* Out-of-Service Clearances
* Transition from Wide Range Level Indication to Shutdown Range during Reactor Flood-up Activities
* Reactor Cavity Drain Down Activities (briefing)
* Walkdowns of Protected Pathways for Electrical Power and Inventory Control
* Fuel Handling Activities.

b. Issues and Findings

.1 Incorrect Tap Setting on Undervoltage Relays

On September 25, 2000, while performing Division II undervoltage bus testing, undervoltage relay 127-2-B-34 failed on two separate occasions. In attempting to restore power to Bus 34, 4160V from cross-tie Bus 34-1, 4160V, the control room operator closed the bus cross-tie control switch. At that time, electricians noted that the undervoltage relay was smoking and contacted on shift personnel. As a result, the licensee initiated an investigation. The licensee's preliminary investigation results revealed that this relay has both an undervoltage and overvoltage coil, and that the tap setting for both the overvoltage coils for Bus 34 had been changed during the last refueling outage in February 1999 (D3R15).

Under normal conditions offsite power is supplied to Bus 34. Bus 34 then supplies power to Bus 34-1. Under an undervoltage condition (Loss of Offsite Power) on Bus 34, the Unit 3 emergency diesel generator which supplies power to Bus 34-1 would have been used to back feed Bus 34 through a cross tie breaker. Bus 34 supplies power to the 'C' and 'D' containment cooling service water (CCSW) pumps, the Unit 3 "C" and "D" low pressure injection pumps, and the 3B core spray pump.

The licensee's investigation revealed that the tap setting of the overvoltage coil had been improperly set to 0.2 amps verses the required setting of 2 amps. As a result, during the performance of the first test with the relay at this setting, the trip signal from the auxiliary relays to the crosstie breaker was present longer than expected and it re-opened the cross-tie breaker when the operator released the switch. The licensee's review of the sequence of events recorder indicated that the trip Bus 34 reset 13 minutes after the Bus 34-1 undervoltage alarm cleared. This time was outside the time limit allowed by the Updated Final Safety Analysis Report for providing CCSW. During the second test, the overvoltage coil failed completely which resulted in not allowing the undervoltage relay to reset. As a result, CCSW would not have been available from the 'C' and 'D' pumps.

The licensee successfully replaced and tested a new relay with the correct setting. The licensee performed a walkdown of the overvoltage coil tap settings for the undervoltage relays on all the other safety-related 4160V Buses and did not identify any additional problems. In addition, the licensee walked down the undervoltage coil tap setting for all the undervoltage relays for the 4160V Buses and identified one incorrect setting. The licensee determined that Bus 24 was set at 2.0 amps when it should have been set at 0.2 amps. The licensee performed an operability evaluation for this setting which concluded that the undervoltage relay could perform its intended safety function.

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, requires that measures should also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components, and . . . provide for verifying or checking the adequacy of design.

Contrary to the above, on February 12, 1999, the licensee performed an uncontrolled design change on the undervoltage relays by changing the overvoltage coil tap setting to an incorrect setting. The licensee's failure to establish sufficient design controls on the overvoltage coil tap settings is a violation. This violation is being treated as an example of a non-cited violation (NCV), consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-249/00-13-04(DRP))**. This issue was captured in the licensee's corrective action program in Condition Report #D2000-05400.

The inspectors assessed the issue using the SDP. The inspectors evaluated the unavailability of the 3C and 3D CCSW pumps during a loss of offsite power transient and determined that power would have been available to the 3A and 3B CCSW pumps by the Unit 2/3 emergency diesel generator. Therefore, the inspectors determined that this issue had very low safety significance (GREEN).

.2 Local Leak Rate Test Failures

On September 19, 2000, the licensee made an emergency notification system (ENS) call under 50.72 (b)(2)(i) to the NRC regarding local leak rate test (LLRT) failures of both the inboard and outboard 'A' header feedwater check valves (FWCVs). The licensee determined that the leakage for each valve was unquantifiable because the test volume would not hold pressure. As a result, this leakage exceeded the containment pathway leakage values allowed by the Primary Containment Leakage Rate Testing Program. The licensee made a follow-up ENS call to the NRC on September 20, 2000, after LLRT results on the 'B' inboard and outboard main steam isolation valves (MSIV) indicated that leakage from each valve also exceeded the TS maximum allowed leakage value of a total of 46 standard cubic feet per minute for all four main steam lines.

The licensee initiated repairs to each valve. The licensee replaced the 'B' inboard MSIV liner and replaced the valve's packing, and lapped the seat on the 'B' outboard MSIV; subsequently, both valves were tested satisfactorily. The licensee installed a modification on the 'A' header FWCV and the subsequent LLRT performed on the valve passed. The 'A' outboard FWCV had its seat lapped and later tested satisfactorily. The licensee initiated an investigation to determine the root cause of the LLRT failures and other LLRT failures that did not exceed TS limits. The licensee was concerned with the high number of LLRT failures after reviewing LLRT results from three previous outages which indicated that the leakage on each of the 'B', 'C', and 'D' outboard and the 'B' inboard MSIVs was insignificant.

The licensee preliminarily determined that the method used to remove the MSIVs from service was the cause for the LLRT failures. The licensee removed the MSIVs from service by cycling all MSIVs until each MSIV's associated accumulator had been depleted. At that point each MSIV was then closed to perform the scheduled LLRTs. With the accumulators empty, the springs on the MSIV were the only motive force available to fully seat the valves. If the licensee had not changed the method for removing the MSIVs from service, the valves would have been closed using air and spring force; this combination would have resulted in five times more motive force being applied to close the valves. The licensee had changed the manner in which the MSIVs were removed from service based of the implementation of corrective actions (CA) from an incident involving an air-operated valve. However, the licensee failed to fully assess the impact of these CAs on the performance of the LLRTs for the MSIVs.

The inspectors evaluated the LLRT failures using the containment integrity significance determination process. The inspectors evaluated how the leakage rates would have affected containment performance if a core damage event had occurred while the MSIVs and FWCVs were leaking. The inspectors assessed the affect on containment for the MSIVs and the FWCV failures separately. The inspectors determined that the leakage failures would be a "Type B" finding and the leakage should be treated as a suppression pool bypass concern since the release would be straight out the main steam and feedwater lines. Furthermore, the inspectors concluded that the leakage from these lines did not constitute a large early release because at least one of the following conditions had been met:

1) Either of the leaking MSIVs can be shown to have closed.

2) The turbine stop valve would have successfully closed.

3) The leakage control system would have been available and the main steam line leakage would have been within the capacity of the standby gas treatment system.

The inspectors confirmed that the MSIVs had closed during the shutdown and the turbine stop valves had properly closed; however, although the main steam line leakage control system had been available, the inspectors were unable to determine if the leakage was within the capacity of the standby gas treatment system. Based on this information, the result for the large early release SDP would be of very low safety significance (GREEN). The licensee has entered these issues into the corrective action program in Condition Reports #D2000-05145, 5092, 5093, 5094, 5141, 5142, and 5285.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and verified that systems selected could perform their intended safety function and that the surveillance tests satisfied the requirements of TSs, the Updated Final Safety Analysis Report, and licensee procedures. During surveillance testing observations, the inspectors verified that the test was adequate to demonstrate operational readiness consistent with the design and licensing basis documents, and that the testing acceptance criteria were clear. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written; and that all testing prerequisites were satisfied; the test data were complete, appropriately verified, and met the requirements of the testing procedure; and that the test equipment range and accuracy were consistent with the application, and the calibration was current. Following the completion of the test, the inspectors verified that the test equipment was removed, and the equipment was returned to a condition to perform its safety function. The review included the following surveillance testing activities:

Barrier Integrity Cornerstone

DOS 5750-04 “Control Room Emergency Ventilation and Air Filtration Unit Monthly Operability Surveillance,” Revision 22

DOS 1600-09 “Pressure Suppression Chamber to Drywell Vacuum Breaker Full Stroke Exercise Test,” Revision 18

Mitigating System Cornerstone

DOS 6600-01, Unit 3 Diesel Generator Monthly Operability Surveillance
DES 6600-05, WR 990186695, Unit 3 Diesel Generator Brush Inspection Surveillance
DOS 1100-04, WR 990184254, Unit 3B Standby Liquid Control Pump In Service Test
DOS 1100-04, WR 990184255, Unit 3B Standby Liquid Control Pump In Service Test
DOS 6600-01, “Emergency Diesel Generator Monthly Surveillance” Revision 12,
DES 0040-17, "Limitorque Environmental Qualification Surveillance" Revision 08,
DES 6700-09, "Inspection and Maintenance of General Electric MC-4.76 Horizontal Draw-Out Metal-Clad Switchgear” Revision 05,
DOS 7000-08, “Local Leak Rate Testing of Primary Containment Isolation Valves,” Revision 0, Unit 3 reactor water cleanup valve 3-1299-285,
DOS 7000-18, “Local Leak Rate Testing of Unit 2(3) Reactor Water Cleanup System Valves,” Revision 0.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA4 Human Performance Issues

a. Inspection Scope

The inspectors reviewed human performance errors associated with several events that were caused by deficient human performance.

b. Issues and Findings

The inspectors identified a number of events that affected operations and safety-related equipment which involved elements of human performance deficiencies. A failure of an instrument mechanic to remove the correct fuse from the reactor protection bus resulted in the receipt of an unexpected half-scram in the control room (See Section 1R13.1). Engineering personnel's incorrect selection of a 120 VAC contactor coil in the "B" reactor protection system motor-generator set breaker resulted in the motor-generator set failure when placing the equipment back in service (See Section 1R13.2). An incorrect overvoltage coil tap setting on the undervoltage relays on Bus 34 would have resulted in the loss of the 'C' and 'D' containment cooling service water pumps during a loss of offsite power transient (See Section 1R20.1). An engineer performed an inadequate parts evaluation for a piping replacement modification on the high pressure coolant injection system (See Section 1R15). All the events indicated a lack of self-check and/or peer-check.

While the risk of the individual events was very low, the number of incidents indicated a performance trend of problems with control, review, and performance of maintenance related activities. In accordance with the reactor oversight process, this issue is not assigned risk significance (NO COLOR).

Licensee management acknowledged a declining trend in human performance at the station and had conducted shop tailgate meetings and trend investigations in an attempt to enhance worker performance.

4OA6 Meetings, including Exit

The inspectors presented the inspection results to Mr. Fisher and other members of licensee management at the conclusion of the inspection on September 26, 2000, and again on October 5, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Chabot, Site Engineering Manager
R. Fisher, Dresden Station Manager
B. Hanson, Shift Operations Supervisor
P. Karaba, FIN Team Supervisor
W. Lipscomb, Training Manager
M. Pacilio, Operations Manager
P. Swafford, Dresden Site Vice President
R. Whalen, System Engineering Manger
R. Kelly, NRC Coordinator

NRC

B. Dickson, Dresden Resident Inspector
M. Ring, Branch Chief
D. Smith, Dresden Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-249/00-13-01	NCV	incorrect fuse removed
50-249/00-13-02	NCV	wrong contactor specified
50-237/249/00-13-03	NCV	high pressure coolant injection system pipe replacement modification
50-249/00-13-04	NCV	improperly changed tap setting

Closed

50-249/00-13-01	NCV	incorrect fuse removed
50-249/00-13-02	NCV	wrong contactor specified
50-237/249/00-13-03	NCV	high pressure coolant injection system pipe replacement modification
50-249/00-13-04	NCV	improperly changed tap setting

Discussed

None

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
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-11	Licensed Operator Requalification	1R11
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Work Prioritization & Control	1R13
71111-14	Nonroutine Evolutions	1R14
71111-15	Operability Evaluations	1R15
71111-19	Post Maintenance Testing	1R19
71111-20	Refueling and Outage Activities	1R20
71111-22	Surveillance Testing	1R22
(none)	Other	4OA4
(none)	Management Meetings	4OA6

LIST OF ACRONYMS AND INITIALISMS USED

ATWS	Anticipated Transient Without Scram
CA	Corrective Actions
CCSW	Containment cooling service water
DTS	Dresden Technical Surveillance
ENS	Emergency Notification System
FWCV	Feedwater Check Valves
HPCI	High Pressure Coolant Injection
IDNS	Illinois Department of Nuclear Safety
LLRT	local leak rate test
MSIV	main steam isolation valve
NCV	Non Cited Violation
OE	Operability Evaluation
SDP	Significance Determination Process
TS	Technical Specification
URI	Unresolved Item
VIO	Violation
WR	Work Request