November 7, 2005

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INTEGRATED INSPECTION REPORT 05000237/2005010;
05000249/2005010

Dear Mr. Crane:

On September 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 20, 2005, with Mr. D. Bost and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and 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.

This report documents one NRC-identified finding and three self-revealed findings of very low safety significance (Green). Each of these findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as Non-Cited Violations (NCVs) consistent with Section VI.A.1. of the NRC Enforcement Policy. Additionally, two licensee-identified violations which were determined to be of very low safety significance (Green) are listed in Section 4OA7 of this report.

If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Dresden Nuclear Power Station.

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Sincerely,

*/RA/*

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000237/2005010; 05000249/2005010
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
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REGION III

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| Docket Nos: | 50-237; 50-249 |
| License Nos: | DPR-19; DPR-25 |
| Report No: | 05000237/2005010; 05000249/2005010 |
| Licensee: | Exelon Generation Company |
| Facility: | Dresden Nuclear Power Station, Units 2 and 3 |
| Location: | 6500 North Dresden Road<br>Morris, IL 60450 |
| Dates: | July 1 through September 30, 2005 |
| Inspectors: | D. Smith, Senior Resident Inspector<br>M. Sheikh, Resident Inspector<br>D. Melendez-Colon, Acting Resident Inspector, Region III<br>C. Phillips, Senior Operating License Examiner<br>W. Slawinski, Senior Radiation Specialist<br>J. Cassidy, Radiation Specialist<br>L. Ramadan, Inspector, Region III<br>M. Gryglak, Reactor Inspector, Decommissioning Branch<br>M. Jordan, NRC Consultant<br>R. Schulz, Illinois Emergency Management Agency |
| Approved by: | Mark Ring, Chief<br>Branch 1<br>Division of Reactor Projects |

Enclosure

## SUMMARY OF FINDINGS

IR 05000237/2005010; IR 05000249/2005010; 07/01/2005 - 09/30/2005; Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; Post Maintenance Testing, Radiation Protection, and Identification and Resolution of Problems.

The report covered a 3-month period of baseline resident inspection; announced baseline inspections on occupational radiation safety radiological access control, independent spent fuel storage installation activities, and Followup to Temporary Instruction 2515/163, “Operational Readiness of Offsite Power.” The inspection was conducted by Region III inspectors and the resident inspectors. Four Green findings, which all involved Non-Cited Violations, were identified. The significance of most findings is indicated by their color (greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

### A. <u>Inspector-Identified and Self-Revealed Findings</u>

**Cornerstone: Mitigating Systems**

- Green. A self-revealing finding involving a Non-Cited Violation (NCV) of Technical Specification 5.4 “Procedures,” was identified on April 1, 2005, due to the licensee’s failure to ensure the post-maintenance test procedure contained proper instructions from the Vendor Equipment Technical Information Program Manual regarding actions to take on a reverse pressurization event of the reactor recirculation pump seals. The lack of procedural guidance in the maintenance procedure resulted in returning the 3B reactor recirculation pump to service with a seal which had a displaced O-ring and a cocked rotating face. This condition caused degradation of the pump seal after approximately four months of operation. The degradation of the seal challenged plant operators and increased the risk of a loss of coolant accident.

  This finding was considered more than minor because it affected the Initiating Event cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because the 3B reactor recirculation pump #2 seal continued to perform its intended function of maintaining the reactor pressure boundary and controlling leakage to within the Technical Specification limits. Corrective actions by the licensee included revising the maintenance procedure to incorporate the Vendor Equipment Technical Information Program (VETIP) Manual guidance on proper actions to take for a reverse pressurization on the reactor recirculation pump seals, and installing a new reactor recirculation pump seal. This finding was related to the cross-cutting issue of human performance because the licensee failed to have pertinent information from the VETIP Manual in the post-maintenance procedure. (Section 1R19.1)

- Green. A self-revealing finding involving a Non-Cited Violation of Technical Specification 5.4, “Procedures,” was identified on April 15, 2005, when control room operators were unable to remotely trip the 2B service water pump from the control room. The inability to trip the pump from safety related 4160 Volt bus 24 was due to the performance of poor maintenance on the pump’s breaker and inadequate post-maintenance testing. The inability to trip the breaker had the potential to render all other loads on bus 24 inoperable, including one division of the containment cooling service water system, or add an additional unanalyzed load on the emergency diesel generator.

  The finding was greater than minor because, if left uncorrected, it could become a more significant safety concern because inadequately performed breaker maintenance could render additional safety-related systems inoperable. The finding impacted the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. As a result of this event, the licensee replaced the trip coil, verified the installation of all the applicable trip coils on both units, revised the work order instructions, and evaluated post maintenance testing of 4 KV breakers. The finding was of very low safety significance because the other division of the containment cooling service water system was available and the licensee was able to trip the breaker locally at the bus. This finding was related to the cross-cutting issue of human performance because electricians failed to properly reinstall the trip coil for the 2B service water pump breaker per the work instructions and the work instructions failed to specify an adequate post maintenance test. (Section 1R19.2)

- Green. A finding involving a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” was identified by the inspectors on July 25, 2005, for the licensee’s lack of timely actions to promptly identify and correct out-of-tolerance lift setpoints for the main steam safety valves and the main steam safety/relief valves (Target Rock valves). The licensee’s actions lacked prioritization in performing Technical Specification required surveillance testing on the Unit 2 and Unit 3 Target Rock safety/relief valves, in determining the cause of the surveillance test failures on the Target Rock valves, and in not assigning corrective actions to determine the cause of the 4G safety valve Technical Specification surveillance test failure. The licensee’s lack of timely actions resulted in the delayed issuance of a Licensee Event Report following the discovery of degradation of the Unit 2 Target Rock valve during disassembly of the valve.

  The finding was greater than minor because, if left uncorrected, the lack of prioritization of the licensee’s actions could lead to the valves not meeting the safety function of preventing over-pressurization of the reactor coolant system. The finding could also lead to the licensee unknowingly operating the units with inoperable safety-related equipment. The finding impacted the Mitigating System cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. The finding was of very low safety significance because the ability of the main steam Target Rock safety/relief valves and the 4G main steam safety valve to function to prevent over-pressurization of the reactor coolant system was not invalidated by the inability of the valves to lift at the prescribed setpoint. In addressing this issue, the licensee discontinued in-plant Technical Specification testing after obtaining approval from the NRC, submitted an analysis to the NRC for determining that the drift condition of the valves was still bounded by the analysis for over-pressurization events, and installed refurbished valves in December 2004. This finding was related to the cross-cutting issue of problem identification and resolution because the licensee's actions were untimely and unfocused. (Section 4OA2.2)

**Cornerstone: Occupational Radiation Safety**

- Green. On June 8, 2005, a self-revealing finding of very low safety significance and an associated violation of NRC requirements were identified for the failure to adequately secure/lock the gate to a posted locked high radiation area (LHRA) and physically challenge the access to verify closure and proper latching in accordance with radiation protection procedures. As a result, access to a posted LHRA was unsecured for a period of approximately 24-hours.

  The issue was more than minor because it was associated with the Program/Process and Human Performance attributes of the Occupational Radiation Safety cornerstone in that the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation was impacted. The issue represents a finding of very low safety significance because it did not involve ALARA planning or work controls, no unauthorized entry into the posted locked high radiation area occurred so there was no overexposure or substantial potential for an overexposure, nor was the licensee's ability to assess worker dose compromised. A Non-Cited Violation of Technical Specification 5.4.1 was identified for the failure to comply with the radiation protection procedure that governs the control of access into high radiation areas. Corrective actions following the identification of the problem included tailgate training for radiation protection staff, development of enhanced pre-job briefing forms for high radiation area entry, performance of an additional physical verification to ensure barriers are secure following work in a locked high radiation area, and plans for additional training specific to high radiation area controls intended for all station radiation workers. Since the principal cause of the problem was a human performance deficiency, the finding also relates to the cross-cutting area of human performance. (Section 2OS1.7)

## B. <u>Licensee Identified Findings</u>

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

## REPORT DETAILS

### Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On July 2, 2005, power was reduced to 657 MWe to perform turbine valve testing, control rod scram time testing, and a control rod pattern adjustment, and the unit was returned to full power the same day.
- On August 20, 2005, power was reduced to 812 MWe to perform control rod pattern adjustment, and the unit was returned to full power the same day.
- On August 30, 2005, the unit was manually shutdown due to unexpected gassing of the main power transformer. A control rod pattern adjustment was performed during the shutdown. The unit returned to full power on September 17, 2005.
- On September 18, 2005, power was reduced to 793 MWe to perform control rod pattern adjustment, and the unit was returned to full power the same day.

Unit 3 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On August 19, 2005, power was reduced from 906 MWe to 804 MWe due to degrading offgas system performance caused by main condenser air inleakage. The unit returned to full power on August 27, 2005.
- On September 24, 2005, power was reduced to 665 MWe to perform turbine valve testing and a control rod pattern adjustment. The unit was returned to full power the same day.

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R01 Adverse Weather (71111.01)

a. Inspection Scope

Summer Readiness: The inspectors reviewed the licensee's process to prepare for reliable operation during summer conditions and verified that the process was followed in accordance with corporate work control procedure WC-AA-107, "Seasonal Readiness," Revision 1. The inspectors walked down equipment and systems to verify proper alignment in accordance with the following licensee procedures to remove plant equipment and systems used for cold weather operations from service at the end of cold weather season.

- DOS 0010-24, “Securing from Cold Weather Operations for Unit 2,” Revision 7;"
- DOS 0010-27, “Securing from Cold Weather Operations for Unit 3,” Revision 5;"
- DOS 0010-30, “Securing from Cold Weather Operations for Radwaste,” Revision 9;” and
- DOS 0010-33, “Securing from Cold Weather Operations at the Lift Station, Goose Lake Pump Station, Security Diesel Building, and Cooling Towers,” Revision 8”

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train to determine that the system met the design of the Updated Final Safety Analysis report. Piping and instrumentation diagrams were used to determine correct system lineup and critical portions of the system configuration were verified. 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.

The inspectors performed partial equipment alignment walkdowns of the:

- Unit 2 ‘B’ train emergency diesel generator system;
- Unit 2 station blackout diesel generator system;
- Unit 2 high pressure coolant injection system; and
- Unit 2/3 ‘A’ train standby gas treatment system.

This represented four inspection samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured plant areas important to safety to assess the material condition, operating lineup, and operational effectiveness of the fire protection system and features to ensure compliance with the station's Fire Hazard Analysis Report. The review included control of transient combustibles and ignition sources, fire suppression systems, manual fire fighting equipment and capability, passive fire protection features, including fire doors, and compensatory measures. The following areas were walked down:

- Unit 3 transformer (TR 31) area, elevation 517', Fire Zone 18.1.1;
- Unit 2 transformer (TR 21) area, elevation 517', Fire Zone 18.2.2;
- Unit 2/3 cribhouse circulation water pump area, elevation 490', Fire Zone 11.3;
- Station blackout building, east area, first floor, Fire Zone 18.6; and
- Unit 2/3 isolation condenser pump house - north cubicle, elevation 517', Fire Zone 18.7.1.

This represented five inspection samples.

b. Findings

No findings of significance were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report flood analysis documents and reviewed the licensee's high pressure coolant injection (HPCI) flooding analysis calculation (calculation number 0591-576-001) to ensure proper protection from a flooding event. In addition, the inspectors walked down the Unit 2 HPCI room to verify the integrity of flood barriers.

This represented one inspection sample for internal flooding.

b. Findings

No findings of significance were identified.

.2 External Flooding

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report flood analysis documents and reviewed the licensee's procedures to ensure the site was properly protected from an external flooding event. The inspectors reviewed the licensee's procedures for external flooding for ensuring proper safe shutdown of the plant, and reviewed the licensee's previously implemented corrective actions for deficiencies associated with flood protection.

This represented one inspection sample for external flooding.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors assessed the implementation of the licensee's Maintenance Rule program to evaluate maintenance effectiveness for the selected systems in accordance with 10 CFR 50.65, Maintenance Rule. The following systems were selected based on being designated as risk significant under the Maintenance Rule, being in the increased monitoring (Maintenance Rule Category a(1)) group, or due to an inspector's identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Local power range monitor system.

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, 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, functional failure determinations associated with the condition and issue reports reviewed, and current equipment performance status.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's maintenance risk program with respect to 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 in accordance with 10 CFR 50.65, Maintenance Rule. The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors also verified that equipment necessary to complete planned contingency actions was staged and available. The inspectors completed evaluations of maintenance activities on the:

- Isolation Condenser System Functional Testing

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations (OE) to ensure that operability was properly justified and the component or system remained available, such that any non-conformance conditions were in compliance with Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." The review included issues involving the operability of:

- XL 3 device 51-17, "2/3 Channel 2/3 diesel generator cooling water pump trouble alarm" (IR 349705);
- OE 05-004, "Unit 2 Emergency Diesel Generator/High Fuel Oil Pressure," Revision 0;
- OE 04-001, "Main Steam Safety Valve 2(3)-0203-4A, B, C, D, E, F, G, H and Target Rock safety/relief valve 2(3)-0-0203-3A of Each Unit Drift Problems," Revisions 1 and 2 (IR 200174);
- EC 357092, "Assess the Effect on the Plant Due to Leakage in Service Water Line";
- OE 05-006, "Non-Safety Related Flex Hose Installed Between Unit 2 Emergency Diesel Generator Air Start Motors," Revision 0;
- ATI 344849-03, "Evaluation of Fire Protection System after Failed Tri-Annual Flow Test,"; and
- IR 343019, "Operator Struck in Head by Falling Light Diffuser."

This represented seven inspection samples.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Around (71111.16)

Quarterly Review

a. Inspection Scope

The inspectors assessed the following operator workaround issue to determine the potential effects on the functionality of the corresponding mitigating system:

- Operator Workaround # 45, “U2 (EHC) Pressure Regulator Continues to Drift”

During this inspection, the inspectors reviewed the technical adequacy of the workaround documentation against the Updated Final Safety Analysis Report and other design information to assess whether the workaround conflicted with any design basis information. The inspectors compared the information in abnormal or emergency operating procedures to the workaround information to ensure that the operators maintained the ability to implement important procedures when needed. Multiple entries into the corrective action program were also reviewed to ensure that the operator workarounds had been entered into this process.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria in Technical Specifications or other design documents. The inspectors also reviewed the tests to determine if the systems were restored to the operational readiness status consistent with the design and licensing basis documents. The inspectors reviewed post-maintenance testing activities associated with the following:

- Unit 3, Work Order 99059664, replacement of 3B reactor recirculation pump seal;
- Unit 2, Work Order 00733304-01, replacement of scram pilot solenoid valves for hydraulic control unit 50-11; and

- Unit 3, Work Order 99081998-01, replacement of solenoid valve for the Unit 3 drywell torus vent to main chimney air operated valve, 3-1601-92.

This represented three inspection samples.

b. Findings

.1 Inadequate Post-Maintenance Procedure for Testing Reactor Recirculation Pump

Introduction: A Green finding involving a Non-Cited Violation of Technical Specification 5.4 was self-revealed when the 3B reactor recirculation pump #2 seal pressure unexpectedly rose to 840 pounds per square inch. The licensee failed to perform an adequate post-maintenance test on the 3B reactor recirculation pump seals when Vendor Equipment Technical Information Program (VETIP) manual information, regarding what actions to take for an overpressurization event of the reactor recirculation pump seal, were not included in the post-maintenance procedure. This procedural deficiency resulted in the premature degradation of the Unit 3B reactor recirculation pump seal. The degradation of the seal challenged plant operators and increased the risk of a loss of coolant accident. The finding was determined to be of very low safety significance because the #2 seal was capable of sustaining full reactor coolant system pressure.

Description: On April 1, 2005, the 3B reactor recirculation pump seal experienced an abnormal condition, in that, the seal experienced a substantial pressure transient. The #2 seal pressure rapidly rose from 513 psig to 840 psig and then slowly decreased to 685 psig over several days. The seal degradation continued and the #2 seal pressure increased again, reaching 750 psig prior to the licensee deciding to shut down the unit on April 26, 2005, to replace the seal. Each recirculation pump was equipped with a mechanical seal assembly designed to prevent reactor coolant system leakage past the pump shaft. The seal assembly consisted of two sets of sealing surfaces; each set consisted of a rotating surface and a stationary surface. The #1 seal was designed for exposure to reactor coolant system pressure, while the #2 seal was designed for normally operated pressure of approximately 500 psig. The pump seal assembly had been replaced previously during the October 2004 refuel outage with a new design.

The licensee initiated an investigation into the degradation of the 3B reactor recirculation pump seal and determined that the root cause was due to the lack of procedural guidance in the Dresden Maintenance Procedure (DMP) 0202-01, “Recirculation Pump Seal Replacement and Pump Leak Test,” Revision 14. Also, a major contributor to the seal pressure transient event was inappropriate corrective action taken by the licensee in not replacing the control rod drive (CRD) system hydrostatic supply line relief valve which had a history of lifting prior to reaching the set point.

The root cause report documented that the seal failure was due to partial displacement of the 1st stage rotating face O-ring. The O-ring displacement was caused by reverse pressurization of the seal assembly during the post-maintenance pump hydrostatic test performed in the October 2004 refuel outage. During this test, the CRD hydrostatic supply line relief valve lifted below its set point. The resultant effect was the rapid depressurization of the lower seal assembly. This rapid depressurization of the lower seal assembly (below the 1st stage) prior to the depressurization of the upper seal assembly (between the stages) resulted in the O-ring being partially displaced. When the O-ring was in the displaced position, it caused the cocking of the 1st stage rotating face and subsequent uneven wearing of the seal surfaces. The licensee determined that the O-ring returned to its normal position during the vessel hydrostatic test which pressurized the O-ring outer face and pushed the O-ring back to its normal position.

The investigation also determined that station workers followed DMP 0202-01. However, the procedure lacked guidance from the VETIP manual. The VETIP manual contained a CAUTION note addressing what actions to take for a reverse pressurization transient during testing of the reactor recirculation pump seal. The lack of procedural guidance in DMP 0202-01 resulted in the return to service of the 3B reactor recirculation pump with a seal which had a displaced O-ring and a cocked rotating face. These abnormal conditions caused degradation of the pump seal after approximately four months of pump operation. The inspectors determined that the licensee's failure to include pertinent information from the VETIP manual, regarding the actions to take for a reverse pressurization event of the reactor recirculation pump seal, in DMP 0202-01, was a performance deficiency.

Analysis: The inspectors determined that the licensee's failure to include essential VETIP manual information in DMP 0202-01 regarding actions to take for a reverse pressurization transient of the seal was a performance deficiency warranting a significant evaluation. Using Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," issued on September 30, 2005, the inspectors determined that this finding was more than minor because it impacted the Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. The failure to maintain adequate procedures for working on equipment can result in degrading equipment or rendering equipment inoperable. This condition caused degradation of the pump seal after approximately four months of operation, challenged plant operators, and increased the risk of a loss of coolant accident. Although the #1 seal failed, the second seal maintained the reactor pressure boundary and controlled leakage within the Technical Specification limits. This finding was related to the cross-cutting issue of human performance.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated December 1, 2004. The inspectors determined that the finding impacted the Initiating Events cornerstone. The inspectors answered "No" to all three questions under the Initiating Events cornerstone column (the finding did not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and did not affect other mitigation systems; did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment functions would be unavailable; and did not increase the likelihood of a fire or internal/external flood). Therefore, the issue screened as having very low safety significance (Green).

Enforcement: Even though the reactor recirculation pump system is not considered a safety-related system, the recirculation pump seals are safety-related components. Therefore, the inspectors determined that the failure to include proper post-maintenance testing instructions, to verify the pump seal had been properly returned to service after a reverse pressure transient, through the incorporation of VETIP manual information in the post-maintenance procedure was a violation of Dresden Technical Specification Section 5.4, “Procedures.” Section 5.4 states, in part, that written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, issued February 1978. Procedures addressing post-maintenance testing are recommended in this regulatory guide. Dresden Maintenance Procedure (DMP) 0202-01, “Recirculation Pump Seal Replacement and Pump Leak Test,” Revision 14 was the procedure established by the licensee for post maintenance testing of the recirculation pump seals. Contrary to the above, on November 17, 2005, the licensee’s post maintenance testing procedure, DMP 0202-01, was inadequately established, in that, the licensee failed to include pertinent information from the VETIP Manual regarding actions to take for a reverse pressurization event on the reactor recirculation pump seal in procedure DMP 0202-01. This failure resulted in the premature degradation of the 3B reactor recirculation pump seal. This event was entered into the licensee’s corrective action program as IR# 329888. Corrective actions by the licensee included revising maintenance procedure DMP 0202-01 to incorporate the VETIP manual guidance on proper actions to take for a reverse pressurization transient of the reactor recirculation pump seal, and installing a new reactor recirculation pump seal. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000249/2005010-01)**

.2 <u>(Closed) Unresolved Item (URI) 05000237/2005008-01</u> "Inability to Trip the 2B Service Water Pump from the Control Room”

<u>Introduction:</u> A Green finding involving a Non-Cited Violation of Technical Specification 5.4 was self-revealed when control room operators were unable to remotely trip the breaker for the 2B service water pump. The inability to trip the breaker from safety related 4160 Volt bus 24 was due to the performance of poor maintenance on the pump’s breaker. The inability to trip the breaker had the potential to render all other loads on bus 24 inoperable, including both pumps of Division 2 of the containment cooling service water system, or place an additional unanalyzed load on the emergency diesel generator. The finding was determined to be of very low safety significance because the other divisional containment cooling service water pumps were available, and the licensee was able to trip the breaker locally at the bus.

<u>Description:</u> On April 15, 2005, onshift operators began swapping service water(SW) pumps by starting the 2A SW pump and attempting to secure the 2B SW pump by placing the control switch in the normal-after-trip position. The pump did not trip as indicated by the motor amperage reading and the tripped light indication of the pump did not illuminate. Subsequently, the onshift operator placed the control switch in pull-to-lock, but the 2B SW pump continued to run. A non-licensed operator was dispatched to bus 24 and the operator tripped the pump with the local trip pushbutton.

The licensee‘s investigation into this issue revealed that maintenance was performed on the 2B SW pump on February 15, 2005, under work order #00727085-01. The work order instructions were to clean and inspect the close latch reset mechanism on the 4KV breaker for hardened lubricant. The work order instructions included the appropriate information from the vendor manual on how to perform the work. Licensee electricians, performing this type of work for the first time, failed to re-install the trip coil mechanism at the same mounting bolt location, per Step 4.9 of the WO instructions, from where it had been removed to ensure the trip coil plunger rod was in alignment with the trip paddle bar. In addition, the post-maintenance testing requirements were inadequate, in that, the instructions only had the electricians test the breaker’s ability to trip from the breaker’s local trip button. The instructions did not test the remote tripping capability from the control room because this task was an optional step (5.5.4). Because this step was not required to be performed, the post-maintenance test did not identify that the SW pump would not trip from the control room due to the misalignment of the trip coil plunger rod and the trip bar paddle upon reinstallation.

The licensee subsequently inspected the installation of the trip coil mechanism for all the potentially affected breakers, on both units, for safety-related buses 23, 24, 33, and 34 and did not identify any other incorrectly installed trip coil mechanisms. The inspectors determined that the failure of the electricians to properly install the trip coil mechanism, as well as the failure to have adequate work order instructions for performing proper post-maintenance testing on the 2B SW pump was a performance deficiency.

<u>Analysis:</u> The inspectors determined that the failure of the electricians to properly install the trip coil mechanism and the inadequate work order instructions for performing proper post-maintenance testing on the 2B SW pump constituted a performance deficiency warranting a significance evaluation. These failures had the potential to render a division of the containment cooling service water system inoperable or add an unanalyzed load on the emergency diesel generator. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on September 30, 2005. The inspectors concluded that the finding, if left uncorrected, could become a more significant safety concern because inadequately performed maintenance and subsequent inadequately performed post maintenance testing could render safety-related systems inoperable. The finding impacted the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Although the work instructions were not followed and the work instructions did not properly test the remote tripping capability of the breaker, the licensee was able to locally trip the pump, so the emergency diesel generator’s ability to respond to accident conditions was not compromised. Also, the other division of the containment cooling service water system was available. This finding was related to the cross-cutting issue of human performance.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, dated December 1, 2004. The inspectors concluded that the finding impacted the Mitigating Systems cornerstone. The inspectors answered ‘No’ to all five questions under the Mitigating Systems cornerstone column: the finding was not a design or qualification deficiency confirmed not to result in loss of function; the finding did not represent a loss of system safety function; the finding did not represent actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; the finding did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk-significant; and the finding did not screen as risk significant due to a seismic, flooding, or severe weather initiating event. Therefore, the issue screened as having very low safety significance (Green).

Enforcement: Technical Specification 5.4 requires that written procedures be established, implemented and maintained covering the applicable procedures in Regulatory Guide 1.33 (Revision 2), Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include procedures for performing maintenance, in that, maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to the above, on February 15, 2005, licensee personnel failed to implement maintenance instructions, in that, electricians failed to install the 2B service water pump breaker's trip coil mechanism as specified by work order instruction #00727085-01, and the work order instructions failed to require verification of the remote tripping capability of the breaker from the control room. As a result of this event, the licensee replaced the coil, verified the proper installation of all the applicable trip coils for both units, revised the work order instructions, and evaluated post maintenance testing of 4 KV breakers. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR324995 and 326406, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.
**(NCV 05000237/2005010-02**)

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

Unit 2 Forced Maintenance Outage

The licensee conducted a forced maintenance outage on Unit 2 from August 30 to September 14, 2005, due to significant gassing of the Unit 2 main transformer. During the outage the licensee replaced the main transformer, performed an overhaul of the 2A condensate pump, replaced the A electro hydraulic control (EHC) pressure regulator card connection and potentiometer, and performed an open face relay modification on the high pressure coolant injection (HPCI) and drywell purge relays.

The inspectors verified that the licensee effectively conducted the shutdown, managed elements of risk pertaining to reactivity control during and after the shutdown, and implemented decay heat removal system procedure requirements in accordance with Technical Specifications and other plant procedures.

The inspectors performed the following activities daily:

- attended control room operator turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance;
- ensured that the licensee appropriately considered risk factors during the development and execution of planned activities;
- monitored licensee's troubleshooting efforts for emergent plant equipment issues;
- performed plant walkdowns to observe ongoing work activities;
- conducted in-office reviews of selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance;
- observed control rod withdrawals and initial transition to criticality;
- monitored mode switch changes and observed portions of power ascension.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

## 1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and reviewed test results. The inspectors assessed whether the selected plant equipment could perform its intended safety function and satisfy the requirements contained in Technical Specifications. Following the completion of each test, the inspectors determined that the test equipment was removed and the equipment returned to a condition in which it could perform its intended safety function.

The inspectors observed surveillance testing activities and/or reviewed completed packages for the tests, listed below, related to systems in the initiating event, mitigating systems, and barrier integrity cornerstones:

- Unit 2/3 DOS 6600-01, "Diesel Generator Surveillance Tests," Revision 89;
- Unit 2(3), Appendix A, "Reactor Coolant System Leakage," Revision 99;
- Unit 2 DOS 6600-08, "Diesel Generator Cooling Water Pump Quarterly and Comprehensive/Preservice Test for Operational Readiness and In-Service Test (IST) Program," Revision 35;

- Unit 3 DOS 6600-08, “Diesel Generator Cooling Water Pump Quarterly and Comprehensive/Preservice Test for Operational Readiness and In-Service Test (IST) Program,” Revision 35;
- Unit 2/3 DIS 1300-01, “Sustained High Reactor Pressure Calibration,” Revision 23;
- DOS 0300-14, “Control Rod Drive Scram Testing at Power,” Revision 06; and
- Unit 2/3 DOA 6500-12, “Low Switchyard Voltage,” Revision 09, Appendix X.

This represented seven inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modification (71111.23)

a. Inspection Scope

The inspectors screened two active temporary modifications and assessed the effect of the temporary modifications on safety-related system functions as specified in the Updated Final Safety Analysis Report and Technical Specifications. The inspectors also determined if the installations were consistent with system design:

- Engineering Change 355653, “Defeat Fuel Pool Cooling Pumps Low Suction Pressure Trips,” Revisions 0 and 1; and
- TCCP Engineering Change 353888, “Isolate U2 Containment Coolant Service Water Pump Vault Cooling Coil 2-57-00-30B to Keep Coil 2-5700-30A Operable.”

This represented two inspection samples.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

**Cornerstone: Occupational Radiation Safety**

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed licensee event reports, corrective action documents, electronic dosimetry transaction data for radiologically controlled area egress, and data reported on the NRC’s web site relative to the licensee’s occupational exposure control performance indicator to determine whether or not the conditions surrounding any actual or potential performance indicator (PI) occurrences had been evaluated, and identified problems had been entered into the corrective action program for resolution. In particular, the inspectors reviewed two incidents which occurred in 2005 that involved locked high radiation area barrier security to determine whether these incidents represented performance indicator occurrences. Also, performance indicator data collection and analysis methods were evaluated overall by the inspectors as described in Section 4OA1.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns/Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified recently completed and ongoing work performed within high and locked high radiation areas of the plant and other potentially exposure significant work activities and selectively reviewed radiation work permit (RWP) packages and radiation surveys for these areas. The inspectors evaluated the radiological controls to determine if these controls, including postings and access control barriers, were adequate.

The inspectors reviewed active and recently closed RWPs and work packages which governed activities in radiologically significant areas to identify the work control instructions and control barriers that had been specified. For these activities, electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant procedures.

The inspectors walked down and surveyed (using an NRC survey meter) radiologically significant area boundaries and other radiological areas in the Unit 2/3 Reactor, Turbine, and Radwaste Buildings to determine if the prescribed radiological access controls were in place, that licensee postings were complete and accurate, and that physical barricades/barriers were adequate. During the walkdowns, the inspectors challenged access control boundaries to determine if high radiation area, locked high radiation area (LHRA) and very high radiation area (VHRA) access was controlled in compliance with the licensee's procedures, Technical Specifications, the requirements of 10 CFR 20.1601, and were consistent with Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The inspectors selectively reviewed RWP and post-job review documents for selected activities completed during approximately the 8-month period that preceded the inspection to determine if barrier integrity and engineering controls performance (e.g., filtered ventilation system operation) were adequate and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent. The inspectors reviewed the licensee's procedures and its methods for the assessment of internal dose as required by 10 CFR 20.1204, to ensure methodologies were technically sound and included assessment of the impact of hard to detect radionuclides such as pure beta and alpha emitters, as applicable. The inspectors reviewed internal dose assessment results and associated calculations for selected workers that had intakes between October 2004 through June 2005. No worker internal exposures greater than 50 millirem committed effective dose equivalent occurred for the period reviewed by the inspectors.

The inspectors reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within the spent fuel pools. Specifically, radiation protection (RP) and fuel handling procedures were reviewed, RP staff were interviewed, and a walkdown of the refuel floor was conducted. In particular, the radiological controls for non-fuel materials stored in the spent fuel pools were evaluated to ensure adequate barriers were in-place to reduce the potential for the inadvertent movement of these materials, and to assess compliance with the licensee's procedures and for consistency with NRC regulatory guidance.

These reviews represented six inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the results of RP department self-assessments related to the radiological access control program, nuclear oversight department field observations of various radiological activities, and the issue report (IR) database along with individual IRs related to the radiological access and exposure control programs to determine if identified problems were entered into the corrective action program for resolution. In particular, the inspectors reviewed radiological problems which occurred over the 15-month period that preceded the inspection including the review of any high radiation area (HRA) radiological incidents (non-PI occurrences identified by the licensee in high and locked high radiation areas) to determine if follow-up activities were conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

1. Initial problem identification, characterization, and tracking;
2. Disposition of operability/reportability issues;
3. Evaluation of safety significance/risk and priority for resolution;
4. Identification of repetitive problems;
5. Identification of contributing causes; and
6. Identification and implementation of corrective actions.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization, and determined if problems were entered into the corrective action program and were being resolved in a timely manner. For potential repetitive deficiencies or possible trends, the inspectors determined if the licensee's self-assessment activities were capable of identifying and addressing these deficiencies, if applicable.

The inspectors reviewed the licensee's documentation for all potential PI events occurring since the last radiological access control inspection performed in April 2004 to determine if any of these events involved dose rates greater than 25 Rem/hour at 30 centimeters or greater than 500 Rem/hour at 1 meter or involved unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5 Rem shallow dose equivalent or greater than 1.5 Rem lens dose equivalent). None were identified.

Additionally, the inspectors reviewed the circumstances surrounding a recent locked high radiation area (LHRA) barrier integrity problem identified through a licensee surveillance and a recent problem that involved the failure to secure a posted LHRA in a radwaste building storage area, a condition which prompted an apparent cause evaluation. Specifically, the licensee's evaluation reports were reviewed and the details were discussed with RP staff, the actual and potential radiological impact of the incidents were independently assessed using the NRC's significance determination process for the occupational radiation safety cornerstone, and the adequacy of the licensee's problem identification, evaluation, and corrective actions were examined. The details associated with one of these incidents is described in Section 2OS1.7.

These reviews represented four inspection samples. Specifically, the samples pertained to the licensee's self-assessment capabilities, the problem identification and resolution program for radiological incidents, a review of the licensee's ability to identify and address repetitive deficiencies, and a review of those radiological incidents and potential PI occurrences of greatest radiological risk.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas</u>

a. <u>Inspection Scope</u>

The inspectors accompanied radiation protection and maintenance staff into the Unit 3 reactor water cleanup pump room (a posted LHRA) and evaluated the radiological controls, job coverage and radiation worker practices during maintenance on an auxiliary pump. The inspectors also accompanied radiation protection staff into the Unit 3 traversing in-core probe (TIP) room (a posted VHRA) and observed radiation surveys being performed in preparation for work planned in the area later that day. Radiation survey information to support these work activities was reviewed by inspectors and the radiological job requirements and the access control provisions for these areas was assessed for conformity with Technical Specifications and with the licensee's procedures. The inspectors also attended the pre-job briefings for these activities to assess the adequacy of the information exchanged.

Job performance was observed to determine if radiological conditions in the work area were adequately communicated to workers through the pre-job briefings and area postings. The inspectors also evaluated the adequacy of the oversight provided by the radiation protection staff including the completion of confirmatory radiological surveys, the work oversight provided by the radiation protection technicians (RPTs), and the administrative and physical controls used over ingress/egress into these areas.

The inspectors also reviewed the licensee's procedures and discussed with RP staff its practices for access into high to very high radiation areas and for areas with the potential for changing radiological conditions such as steam sensitive areas, the drywell, and the TIP rooms to determine the adequacy of the radiological controls and hazards assessment associated with such entries. Work instructions provided in RWPs and in pre-entry briefing documents were discussed with RP staff to determine their adequacy relative to industry practices and NRC Information Notices.

The inspectors also reviewed the licensee's procedure and generic practices associated with dosimetry placement and the use of multiple whole body dosimetry for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201 and applicable industry guidelines. Additionally, previously completed work in areas where dose rate gradients were subject to significant variation, such as during reactor cavity decontamination and work on the reactor head, were reviewed to evaluate the licensee's practices for dosimetry placement.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, LHRA and VHRA Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures and RP job standards and evaluated RP practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas). The inspectors assessed compliance with the licensee's Technical Specifications, procedures, the requirements of 10 CFR Part 20, and the guidance contained in Regulatory Guide 8.38. In particular, the inspectors evaluated the RP staff's control of keys to LHRAs and VHRAs, the use of access control guards during work in these areas, and methods and practices for independently verifying proper closure and locking of access doors upon area egress. The inspectors selectively reviewed key issuance/return and door lock verification records and key accountability logs for selected periods in 2005 to determine the adequacy of accountability practices and documentation. The inspectors also reviewed selected records and evaluated the RP staff's practices for radiation protection manager and station management approval for access into Level 2 LHRAs and VHRAs and for the use of flashing lights in lieu of locking areas to verify compliance with procedure requirements and those of 10 CFR 20.1602.

The inspectors discussed with RP staff the controls that were in place for areas that had the potential to become high radiation areas during certain plant operations to determine if these plant operations required prior communication with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards. In particular, revisions to radwaste and reactor operations procedures and RP guidance developed to identify process piping flow paths and vulnerable areas subject to changing radiological conditions were discussed with RP supervisory staff.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to numerous LHRAs in the Unit 2/3 Reactor and Turbine Buildings and the common Radwaste Building, and for VHRAs (TIP rooms and Drywell airlocks).

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During work in the reactor water clean up pump room, the inspectors evaluated radiation worker performance for conformity with radiation protection work requirements and to determine whether workers were aware of the radiological conditions, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present.

The inspectors also reviewed radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this matched the corrective action approach taken by the licensee to resolve the identified problems. Section 2OS1.7 below documents a radiation worker performance deficiency that contributed to a high radiation area access control problem.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician (RPT) Proficiency

a. Inspection Scope

During job observations and general plant walkdowns, the inspectors evaluated RPT performance with respect to radiation protection work requirements, conformance with procedures and those requirements specified in the RWP, and assessed overall proficiency with respect to radiation protection requirements, station procedures and health physics practices.

The inspectors reviewed selected radiological problem reports generated between April 2004 and June 2005 to determine the extent of any specific problems or trends that may have been caused by deficiencies with RPT work control, and to determine if the corrective action approach taken by the licensee to resolve the reported problems, if applicable, was adequate. In particular, the inspectors reviewed the circumstances associated with a June 2005 incident that involved the failure to ensure a gate to a posted LHRA was secured upon completion of work in the area. For that incident, the inspectors walked down the area, reviewed the licensee's apparent cause evaluation report along with the radiological surveys of the area, and interviewed those involved in the incident and in its followup investigation.

These reviews represented two inspection samples.

b. Findings

Introduction: A self-revealing finding of very low safety significance and an associated violation of NRC requirements were identified for the failure to ensure that a gate leading into a posted LHRA was locked/latched and secured following work in the area. The condition existed about 24 hours until the problem was identified by a RPT that noticed the unsecured gate.

Description: On June 9, 2005, a RPT performing routine surveys in the radwaste building found the gate to radwaste storage bay no. 11 unsecured. At that time, the bay housed 14 shielded 55-gallon drums and a bin of solid radioactive waste that were positioned side-by-side along the back portion of the area. The gate was posted as a LHRA given the radiation levels measured previously on the drums. The bay was intended to be secured with a padlock to prevent unauthorized entry as required by RP procedure and Technical Specification 5.7 for LHRAs. About 24 hours prior to the identification of the problem, another RPT and two members of the maintenance staff worked in the bay sorting and consolidating radioactive waste. Similar work was conducted in the bay by the same individuals on June 7, 2005, as part of an ongoing radwaste consolidation project.

Prior to commencement of work in the storage bay on June 8, 2005, the three person work crew participated in pre-job and ALARA briefings. The briefings focused on the work to be performed and the measures to reduce radiation exposure. High radiation area barrier and access control integrity were not part of the briefing discussions. The key for the padlock was assigned to the RPT that provided the job coverage for the work activity. Work initiated on the morning of June 8 and continued throughout the day in the radwaste building and within the storage bay until mid-afternoon.

Following completion of work that day, the RPT closed the gate, locked/secured the padlock and reestablished the LHRA posting across the front of the gate. One of the involved maintenance workers then performed a peer check of the lock which is required by RP procedure. Specifically, following completion of work in a LHRA, procedure RP-AA-460, "Controls for High and Very High Radiation Areas," requires that individuals exiting the area ensure the access is secured/locked by physically challenging the access to ensure closure and proper latching. Additionally, through a peer check process, the key custodian/RP individual and a second individual are required to physically challenge the access to ensure closure and proper latching. However, while both the RPT and one of the maintenance workers checked that the padlock was locked and secured to a wall mounted hasp through a physical challenge of the padlock, neither individual challenged the gate and consequently failed to recognize that the padlock was not engaged with the gate. As a result, the LHRA posted storage bay was unsecured/unlocked and therefore the principal physical barrier intended to prevent unauthorized entry was compromised.

The licensee's followup investigation correctly determined that the gate was not properly secured initially and that the verification was inadequate because both workers failed to physically challenge the gate/lock mechanism. The lack of adequate pre-job and ALARA briefings likely contributed to the human performance problem. The licensee's investigation found that radiation levels atop accessible areas of the 55-gallon drums approached, but did not exceed, 1000 mrem/hour at 30 centimeter distances while the bottom of one drum measured in excess 1000 mrem/hour. However, due to the weight of each drum which precluded their movement without a forklift coupled with the location of the one drum that exhibited dose rates in excess of 1000 mrem/hour, the NRC concluded that LHRA radiological conditions were inaccessible to personnel as provided in Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

Analysis: The failure to secure/lock the posted LHRA and physically challenge the access gate/padlock mechanism as required by the licensee's procedure represents a performance deficiency as defined in NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening." The inspectors determined that the issue was associated with the Program/Process and Human Performance attributes of the Occupational Radiation Safety cornerstone and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation. Therefore, the issue was more than minor and represented a finding which was evaluated using the Significance Determination Process (SDP).

Since the finding involved a radiological access control problem and the potential for unauthorized entry into a LHRA, the inspectors utilized IMC 0609, Appendix C, "Occupational Radiation Safety SDP," to assess its significance. The inspectors determined that the finding did not involve ALARA planning or work controls. Since no unauthorized entry into the area occurred while it was unlocked for approximately 24 hours, there was no overexposure. Also, given the radiological conditions and the inaccessibility of personnel to areas that exhibited elevated dose rates, there was no substantial potential for an overexposure. The licensee's ability to assess dose was also not compromised for this incident. Consequently, the inspectors concluded that the SDP assessment for this finding was of very low safety significance (Green).

As described above, the individuals that performed the work in the storage bay failed to secure the padlock to the hasp/gate and subsequently failed to verify that area access was secured through an adequate physical challenge. Consequently, these human performance deficiencies were the principal cause of the problem.

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33 (Revision 2), Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include RP procedures for access control to radiological areas, which are provided by licensee procedure RP-AA-460, “Control for High and Very High Radiation Areas.” RP-AA-460 requires that individuals exiting the area ensure the access is secured/locked by physically challenging the access to ensure closure and proper latching. Contrary to this procedure, following completion of work in the radwaste bay on June 8, 2005, workers failed to ensure and verify that the gate to this posted LHRA was closed, secured and latched through an adequate physical challenge of the access.

Corrective actions taken by the licensee included tailgate training for the RP staff, development of an expanded pre-job briefing checklist for high radiation area entry, additional physical verification to ensure barriers are secure following work in LHRAs, and plans for additional training specific to high radiation area controls intended for all station radiation workers. Since the licensee documented this issue in its corrective action program (Issue Report (IR)/Apparent Cause Evaluation No. 342650) and because the violation is of very low safety significance, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000237/2005010-03;05000249/2005010-03**)

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

**Cornerstone: Occupational Radiation Safety**

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicator (PI) listed below for the period April 2004 through June 2005. To determine the accuracy of the PI data reported during that period, PI definitions and guidance contained in Revision 3 of Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used. The following PI was reviewed:

- Occupational Exposure Control Effectiveness

For the time period reviewed, no reportable occurrences were identified by the licensee. To assess the adequacy of the licensee’s PI data collection and analyses, the inspectors discussed with RP staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm reports, the dose assignments for any intakes that occurred for the period and the licensee's IR database along with individual IRs generated during the period reviewed to verify there were no unrecognized occurrences. Additionally, as discussed in Section 2OS1, the inspectors walked down the boundaries of selected LHRAs and VHRAs to verify the adequacy of postings and physical barriers.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Quarterly Review

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. In addition, in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing daily issue reports and attending daily issue report review meetings.

b. Findings

No findings of significance were identified.

.2 Untimely Actions with respect to Testing and Inspection of Main Steam Safety/Relief Valves (Target Rock Valves) and Main Steam Safety Valves

a. Inspection Scope

The inspectors performed a detailed review of selected issues to determine if problem characterization was accurate and to verify that extent of condition reviews were adequately completed or were in the process of being performed. The inspectors selected and reviewed the licensee's actions involving the surveillance testing of the main steam Target Rock safety/relief valves and the main steam safety valves to verify compliance with Technical Specifications (TS).

b. Findings

Introduction: One Green finding involving a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI was identified by the inspectors. In addition, the licensee identified a related Non-Cited Violation of TS 3.4.3 which was determined to have very low safety significance. The licensee-identified violation is documented in Section 4OA7 of this report. The Green finding involved a lack of prioritization and untimely actions by the licensee in performing TS required surveillance testing of the main steam Target

Rock safety/relief valves on both units, in subsequently determining the cause of the failure of the safety/relief valves to lift at the required setpoint, and in not developing corrective actions to determine the cause of the failure of the 4G main steam safety valve to lift at the TS required setpoint. As a result, the licensee was delayed in issuing a Licensee Event Report (LER) upon discovering degradation of the Unit 2 Target Rock safety/relief valve during its inspection. This was the first time a main steam Target Rock safety/relief valve was determined to have been inoperable while in service because these valves were not required to meet the one percent tolerance until a TS amendment was issued on July 30, 2004. However, historical test results show that these valves have repeatedly been unable to meet a one percent tolerance of the nameplate lift pressure. The finding was determined to be of very low safety significance because all of the safety valves were still able to perform their intended safety function.

Description:

Lack of Prioritization and Untimely Actions in Performing TS 3.4.3.1 Surveillance Testing and Inspections

Each unit at Dresden has 8 safety valves and 1 Target Rock safety/relief valve. The licensee typically removes 4 safety valves and the Target Rock valve each refueling outage, replaces the valves with previously tested valves, and has the removed valves tested for lift setpoint accuracy. The inspectors reviewed two apparent cause evaluations, #303692 for Unit 2 and #303699 for Unit 3, which documented that the two main steam Target Rock safety/relief valves had exceeded the TS 3.4.3.1 "Safety and Relief Valves," surveillance testing requirement lift setpoint of 1135 pounds per square inch gage (psig) plus or minus one percent tolerance. The inspectors' review of this issue determined that during the Fall 2004 dual unit outages, the main steam Target Rock safety/relief valve was removed from each unit. Prior to replacing the Unit 2 valve in November 2004, the licensee had noted and been trending elevated tailpipe temperatures on the valve since its installation in November 2003. Although the valves were removed in November 2004; the licensee did not have the valves tested until February 17, 2005. At the time of testing, both valves failed in the conservative direction. The Unit 2 main steam Target Rock safety/relief valve, 2-203-3A, lifted at 1091 psig or 3.9 percent below the nameplate lift pressure. This was outside the TS one percent tolerance requirement as well as the American Society of Mechanical Engineers (ASME) code requirement tolerance of three percent. The Unit 3 main steam Target Rock safety/relief valve, 3-203-3A, lifted at 1119 psig or 1.4 percent below nameplate.

On July 7, 2005, the inspectors questioned the licensee on why an LER had not been issued on the failure of both valves to meet TS 3.4.3.1 surveillance requirements as a condition prohibited by TS. The licensee indicated that the valves would be disassembled in the future to determine if the condition of the valves met 10 CFR 50.73 reportability requirements. Although the licensee stated that scheduling issues with the valve's vendor had delayed the valve inspections, the licensee did not appear to place any urgency on completing this task since the valves had been removed in November 2004. After the inspectors questioned the timeliness of the inspections, the vendor disassembled and inspected the Unit 2 valve on July 25, 2005, and the Unit 3 valve on

August 5, 2005. The Unit 3 Target Rock safety/relief valve did not have any physical damage, therefore, the licensee determined that the valve's failure to meet TS surveillance testing requirements in February 2005 was not a reportable event, consistent with NUREG 1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." However, the Unit 2 valve's second stage disc and seat exhibited steam cutting which was determined to be the cause of the valve lifting outside of TS requirements. This was determined to be reportable as a condition prohibited by TS and the licensee issued an LER in September 2005.

The licensee believed that the physical damage to the Unit 2 Target Rock safety/relief valve was caused by foreign material lodged between the valve's second stage seat and disc which had been introduced into the valve during TS 3.4.3.1 in-plant testing with reactor coolant system steam in November 2003. The licensee discontinued this method of testing after obtaining a TS change from the NRC in October 2004. The current testing method involved shipping the valves to the vendor for testing.

Historical Performance of the Main Steam Target Rock Safety/Relief Valves

On October 10, 2002, Exelon submitted a license amendment request to modify TS 3.4.3 to require nine instead of eight safety valves based on implementing extended power uprate. The amendment was approved by the NRC on July 30, 2004. Prior to July 2004, the Target Rock safety/relief valves were not required to meet the one percent tolerance requirement of TS 3.4.3. The licensee always tested the valves to the ASME Code which required the valves to lift within three percent of the valve's nameplate lift pressure. The valves on both units would often lift greater than the plus or minus one percent of the nameplate lift pressure with a failure rate of about 70 percent. However, the valves had a history of generally meeting the three percent tolerance ASME Code requirement. Therefore, the licensee did not disassemble the valves prior to July 2004 and considered the as-found out-of-tolerance lift pressure condition to be normal drift. The Target Rock valves' history of failing to meet the plus or minus one percent tolerance indicated that the licensee would have a high probability of not meeting the one percent tolerance when implementing the July 2004 TS 3.4.3.1. change.

Lack of Timely Corrective Action for the Failure of the 4G Main Steam Safety Valve

The licensee also experienced difficulty with the main steam safety valves meeting the plus or minus one percent tolerance for lift setpoint. The historical performance of the each unit's eight main steam safety valves, in not meeting TS 3.4.3.1 surveillance testing requirements, showed that about one out of four valves failed to meet the lift setpoint tolerance during testing. In one case involving Unit 3, the licensee failed to assign corrective actions to determine the cause of the 4G main steam safety valve's failure to meet its TS 3.4.3.1 surveillance testing requirement. The valve had been taken out of service on October 26, 2004, and surveillance tested on November 10, 2004. The 4G valve failed to lift within plus or minus one percent of the nameplate lift pressure (1250 psig) when it lifted early at 1231 psig or 1.52 percent below nameplate. The licensee generated an IR for the 4G valve failure, but did not specify the need to determine the root cause of the failure of the valve. This determination was important because it provided the basis for whether a reportable condition was met. After 10 months from the valve's removal, the inspectors prompted the licensee on September 14, 2005, about the need to determine the cause of the valve's failure; therefore, the licensee generated IR# 379027.

Subsequently, upon contacting the vendor, the licensee discovered that the 4G valve had, in fact, been disassembled and inspected in November 2004. The vendor's inspection of the valve in 2004 noted some physical seat damage. The presence of seat damage appeared to indicate that the valve may have been inoperable while in service. However, upon further follow-up on the seat damage, the vendor stated that the damage would not have caused the valve to lift outside of the plus or minus one percent of nameplate lift pressure tolerance. Therefore, using the guidance of NUREG 1022, the licensee was finally able to conclude that the 4G valve lift setpoint test failure was not a reportable condition.

Analysis: The inspectors determined that the licensee's lack of prioritization and timely action in conducting the TS 3.4.3.1 surveillance testing on the main steam Target Rock safety/relief valves on both units, in subsequently determining the cause of each valve's failure during surveillance testing, and in not assigning corrective actions to determine the cause of the 4G safety valve TS surveillance failure were three examples of performance deficiencies warranting significance evaluations. The inspectors perceived the deficiencies as having the same basic theme, lack of timely corrective actions, and treated the issues as one finding. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 30, 2005, because if left uncorrected, the licensee's delayed actions could lead to the main steam Target Rock safety/relief valves and main steam safety valves not meeting their intended safety function of preventing over-pressurization of the reactor coolant system. The finding could also lead to the licensee unknowingly operating with inoperable safety related equipment. The finding impacted the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Although both Target Rock valves lifted outside the acceptable tolerance value and there was steam cutting on the Unit 2 valve, analysis showed that both units' Target Rock safety/relief valves were still able to function to prevent over-pressurization of the reactor coolant system, in part, because they lifted early. In addition, the 4G main steam safety valve was still capable of performing its intended safety function. The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution.

The inspectors completed a Phase 1 significance determination for this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated December 1, 2004. The inspectors determined that this finding impacted the Mitigating Systems cornerstone. The inspectors answered "No" to all five questions under the Mitigating System cornerstone column (the finding was not a design or qualification deficiency confirmed not to result in loss of function per GL 91-18; did not represent a loss of system safety function; did not represent actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours; did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event). Therefore, the issue screened as having very low safety significance (Green).

Enforcement: Appendix B of 10 CFR 50, Criterion XVI, requires in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, the inspectors identified that the licensee did not establish measures to assure that the out-of-tolerance lift setpoints for the main steam safety valves and the main steam safety/relief valves (Target Rock valves) were promptly identified and corrected. The licensee's actions lacked prioritization in performing TS surveillance testing for the Unit 2 and Unit 3 main steam Target Rock safety/relief valves and in pursuing the cause of the valves' surveillance testing failures. The licensee removed both valves from the system during the Fall of 2004. Although previously tested valves had a history of not lifting at the nameplate lift pressure plus or minus one percent, the licensee did not immediately schedule the valves for surveillance testing. Additionally, the licensee had noticed that tailpipe temperature had become elevated on the Unit 2 valve since May 2004. Notwithstanding, the licensee did not make a concerted effort to arrange for the vendor to promptly disassemble and inspect the valve for damage. The licensee's lack of prompt actions in determining the cause of the valves' failure to lift at the required setpoint led to a delay in issuing a Licensee Event Report on September 23, 2005, for the deficient condition of the Unit 2 valve even though the valve had been removed from service in November 2004. In addition, the licensee failed to assign corrective actions for determining the cause of the failure of the 4G main steam safety valve to meet TS surveillance testing requirements when it was tested in November 2004. In addressing this issue, the licensee discontinued in-plant TS surveillance testing, submitted an analysis for determining that the drift condition of the valves was still bounded by analysis for overpressurization events, and installed a refurbished valve in each unit in 2004. The inspectors determined that this issue was of very low safety significance and the issue was entered into the licensee's corrective action program with IRs 358734, 303692, and 303699. The licensee has also submitted a setpoint tolerance and uncertainty treatment methodology to the NRC and, pending approval of the methodology, plans to submit a license amendment to change the setpoint tolerance values. Therefore, this issue is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2005010-04; NCV 05000249/2005010-04)**

4OA3 Event Followup (71153)

a. Inspection Scope

The inspectors reviewed two licensee event reports (LERs) to ensure that the issues documented in the reports were adequately addressed in the licensee's corrective action program. The inspectors interviewed plant personnel and reviewed operating and maintenance procedures to ensure that generic issues were captured appropriately.

The inspectors reviewed operator logs, the Updated Final Analysis Report, and other documents to verify the statements contained in the LER and to close out the unresolved item.

b. Findings

.1 (Closed) Licensee Event Report (LER) 50-237/2005-004-00: "Unit 2 Main Steam Target Rock Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value"

This issue is discussed in Section 4OA2.2 and Section 4OA7.1, "Identification and Resolution of Problems" and "Licensee-Identified Violations" respectively of this report.

This LER is closed.

This represented one inspection sample.

.2 (Closed) Licensee Event Report (LER) 50-237;50-249/2005-001-00: "4160 Volt Relaying and Metering Single Failure Vulnerability for Units 2 and 3"

This issue is discussed in Section 4OA7.2, "Licensee-Identified Violations" of this report.

This LER is closed.

This represented one inspection sample.

4OA5 Other Activities

.1 Operation of an Independent Spent Fuel Storage Installation (ISFSI) (60855.1)

a. Inspection Scope

The inspector evaluated the root cause analysis report performed by the licensee as a result of the Cask Transfer Facility (CTF) gear failure that occurred on June 3, 2005, during the evolution to transfer the Multi-purpose Canister (MPC) to a Hi-Storm storage cask.

The inspector reviewed the 10 CFR 50.59 safety evaluation and the 10 CFR 72.48 safety screening to verify that the use of an alternative lifting device with four hydraulic lifting boom systems to transfer the MPC to a storage cask was bound by the conditions set in the Certificate of Compliance (CoC), the Technical Specifications, and the Final Safety Analysis Report (FSAR). The inspector evaluated the design analysis, "Structural Qualification of Hi-Storm Loading at DNPS Using Hydraulic Lifters," to confirm the alternate lifting device satisfied the design criteria specified in the American National Standards Institute recommendations, ANSI N14.6, and contained the redundant drop protection features. In addition, the inspector reviewed the analysis to verify that the hydraulic lifting assembly conformed to the stress limits specified in the American Society of Mechanical Engineers Code, ASME Section III, and the beam deflection limits contained in the Crane Manufacturers Association of America specifications, CMAA # 70. The inspector also verified that there was adequate concrete strength to support the weight of the lifting beams and the cask.

The inspector reviewed the regular procedure used to transfer an MPC to a Hi-Storm and the special procedure which governed the use of the hydraulic lifting device in lieu of the CTF. The inspector observed the licensee complete the transfer of an MPC to a Hi-Storm using the hydraulic lifting system.

The inspector reviewed the licensee's response to a 10 CFR 21 Notification issued on January 6, 2005, by the Whiting Corporation. The Notification identified an overstress condition on hoist equalizer plates and welds. The inspector reviewed the initial crane inspection results and the final modification records including work instructions to install reinforcing plates. The inspector also reviewed the engineering analysis that demonstrated the structural integrity of the modified equalizer arms to sustain loads up to 125 tons.

b. Findings

No findings of significance were identified.

.2 Operational Readiness of Offsite Power (Temporary Instruction 2515/163)

The objective of Temporary Instruction (TI) 2515/163, "Operational Readiness of Offsite Power," was to confirm, through inspections and interviews, the operational readiness of offsite power (OSP) systems in accordance with NRC requirements. The inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/163 with licensee personnel during the 2nd Quarter of 2005. The results of the inspectors' review were forwarded to Office of Nuclear Reactor Regulation (NRR) for additional review and evaluation.

Following review and evaluation by the NRR staff, several follow-up questions were sent back to the inspectors for discussion with licensee personnel. The results of the inspectors' review and discussion of the follow-up questions, performed during the 3rd Quarter of 2005, were again forwarded NRR for evaluation.

The completion of this TI was documented in NRC Inspection Report (IR) 05000237/2005008; 05000249/2005008, and represented one inspection sample. The follow-up questions the inspectors discussed with licensee personnel during this inspection period were considered a part of the original inspection sample, and did not constitute an additional inspection sample for this TI.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. D. Bost, and other members of licensee management on October 20, 2005. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Occupational radiation safety radiological access control inspection with Mr. D. Wozniak and other licensee staff on July 15, 2005.

- Independent spent fuel storage installation with Mr. D. Wozniak, Mr. M. Mikota and other licensee staff on August 25, 2005.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

.1 Technical Specification 3.4.3, "Safety and Relief Valves," requires the safety function of all nine main steam safety valves to be operable in Modes 1, 2, and 3. Technical Specification Limiting Condition for Operation, Action 3.4.3.B, requires operators to place Unit 2 in Mode 3 within 12 hours and in Mode 4 within 36 hours with one or more safety valves inoperable. The Unit 2 main steam Target Rock safety/relief valve failed its lift setpoint surveillance test on February 17, 2005. Subsequent inspection of the valve, on July 25, 2005, revealed steam cutting on the second stage seat and disc and the licensee determined that the valve had been inoperable beyond the TS allowed outage time during the time the valve was in-service. The inspectors determined that the licensee's failure to comply with TS 3.4.3.B Limiting Condition for Operations was a violation.

The inspectors determined the violation to be of very low safety significance in evaluating the issue using the significance determination process because the valve was capable of performing its intended safety function. This issue was entered into the licensee's corrective action program with IRs 358734, 303692, and 303699. Additional information regarding this issue and the performance of the Target Rock safety/relief valves is described in Section 4OA2.2 of this report.

.2 Appendix B, Criterion III, Design Control, of 10 CFR 50, states 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 the structures, systems and components..." Contrary to the above, on February 15, 2005, the licensee identified that the 4160 V non-dash buses protective relaying circuitry, associated with the unit auxiliary transformers and the reserve auxiliary transformers, was inappropriately designed during the upgrade of these non-dashed buses to a safety-related classification. An open circuit condition in the common electric protective relaying circuitry had the potential to isolate all power sources to each non-dash bus, including the emergency diesel generators. Two of the buses feed the Division 1 and Division 2 containment cooling service water systems.

This condition was evaluated using the Significance Determination Process, Phases 1 and 2 and determined to be of very low safety significance because although emergency power would not be available to the containment cooling service water systems, emergency power would be available to all other emergency core cooling systems. In addressing this issue, the licensee immediately installed a temporary modification which removed the lockout function from the emergency diesel generator breakers on both units. The licensee planned to design and install a permanent modification to the circuitry to eliminate the single failure vulnerability and conducted preliminary engineering reviews for similar existing latent design deficiencies which did not find any similar deficiencies. In addition, the licensee has configuration change procedures for the installation of new designs which would identify this type of issue.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

D. Bost, Site Vice President
D. Wozniak, Plant Manager
H. Bush, Radiation Protection, Radiological Engineering Manager
R. Conklin, Radiation Protection Supervisor
J. Fox, Design Engineer
R. Gadbois, Operations Director
D. Galanis, Design Engineering Manager
V. Gengler, Dresden Site Security Director
J. Griffin, Regulatory Assurance - NRC Coordinator
P. Salas, Regulatory Assurance Manager
A. Khanifar, Nuclear Oversight Director
S. Kroma, Reactor Services Project Manager
T. Loch, Supervisor, Design Engineering
M. McGivern, System Engineer
M. Mikota, Dry Cask Project Manager, Dresden
M. Overstreet, Lead Radiation Protection Supervisor
N. Spooner, Site Maintenance Rule Coordinator
J. Strmec, Chemistry Manager
B. Surges, Operations Requalification Training Supervisor
G. Bockholdt, Maintenance Director
S. Taylor, Radiation Protection Manager

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1

IEMA

R. Schulz, Illinois Emergency Management Agency
R. Zuffa, Resident Inspector Section Head, Illinois Emergency Management Agency

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000249/2005010-01 | NCV | Failure to Perform Post-Maintenance Test on the 3B Reactor Recirculation Pump Seals |
| 05000237/2005010-02 | NCV | Unable to Remotely Trip the 2B Service Water Pump from Control Room from Safety Related 4160 Volts bus 24 |
| 05000237/2005010-03<br>05000249/2005010-03 | NCV | Failure to Ensure That a Gate to a Posted LHRA was Secured Following Work in the Area |
| 05000237/2005010-04<br>05000249/2005010-04 | NCV | Lack of Prioritization for Performing TS 3.4.3.1 Surveillance Testing and Valve Inspections for Target Rock Valves and Corrective Action Assignments for the 4G Valve |

Closed

| | | |
|---|---|---|
| 05000249/2005010-01 | NCV | Failure to Perform Post-Maintenance Test on the 3B Reactor Recirculation Pump Seals |
| 05000237/2005010-02 | NCV | Unable to Remotely Trip the 2B Service Water Pump from Control Room from Safety Related 4160 Volts bus 24 |
| 05000237/2005010-03<br>05000249/2005010-03 | NCV | Failure to Ensure That a Gate to a Posted LHRA was Secured Following Work in the Area |
| 05000237/2005010-04<br>05000249/2005010-04 | NCV | Lack of Prioritization for Performing TS 3.4.3.1 Surveillance Testing and Valve Inspections for Target Rock Valves and Corrective Action Assignments for the 4G Valve |
| 05000237/2005008-01 | URI | Inability to Trip the 2B Service Water Pump from the Control Room |
| 50-237/2005-004-00 | LER | Unit 2 Main Steam Target Rock Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value |
| 50-237;50-249/<br>2005-001-00 | LER | 4160 Volt Relaying and Metering Single Failure Vulnerability for Units 2 and 3 |

Discussed

None

## LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather

-IR 343789; Summer readiness check-in assessment issue; dated June 14, 2005
-IR 348415; Potential issue to cool generator H2; dated June 28, 2005
-IR 357558; NOS identified missed opportunity - summer readiness; dated July 28, 2005
-WC-AA-107, "Seasonal Readiness," Revision 1
-Dresden's Archival Operations Narrative Logs for July 22, 2005
-DOS 0010-24, Revision 7, "Securing from cold weather operations for Unit 2"
-DOS 0010-27, Revision 5, "Securing from cold weather operations for Unit 3"
-DOS 0010-30, Revision 9, "Securing from cold weather operations for Radwaste"
-DOS 0010-33, Revision 8, "Securing from cold weather operations at the lift station, Goose Lake pump station, security diesel building and cooling towers"

1R04 Equipment Alignment

-DOP 6620-E1, "Unit 2 Station Blackout Electrical Checklist," Revision 03
-DOP 6620-M1, "Unit 2 Station Blackout Mechanical Checklist," Revision 07

1R05 Fire Protection

-IR 346582; Fire barrier for penetrations in U2 DG room inop; dated June 22, 2005
-IR 349049; Weaknesses noted in fire drill; dated June 30, 2005
-IR 350676; Two cells showed low voltage; dated July 6, 2005
-IR 350838; XL-3 device 84-27 for U2 SBO; dated July 7, 2005
-IR 352703; NOS ID's repeat trans. comb. permit administration issues; dated July 13, 2005
-IR 353121; Secondary hose line too short to support attack team; dated July 14, 2005
-IR 353366; NOS ID required manual action not included in pre-fire plan; dated July 15, 2005
-IR 355415; Concerns during fire drill performance; dated July 21, 2005
-IR 356524; Fire on pole SE side of property; dated July 25, 2005
-IR 356542; ISFSI pad housekeeping issues; dated July 25, 2005
-IR 360312; EC 349539, Rev. 3 design summary section 4.1.34 inadequate; dated August 5, 2005
-IR 361241; Weakness noted in fire drill; dated August 9, 2005

1R06 Flooding

-IR 350601; IEMA identifies lack of flood analysis for Hi-Tract cask; dated July 6, 2005
-IR 363274; PAR errors during training drill; dated August 15, 2005
-IR 363469; Too few EP Dep opportunities in operator training; dated August 16, 2005
-IR 366892; Medical drill - IS security access for ambulance STD W/BW; dated August 25, 2005
-UFSAR 2.4 Hydrologic Engineering, Revision 01A
-UFSAR 3.4 Water Level (Flood) Design, Revision 4
-IR 111005; NRC identifies weakness in external flood procedure; dated June 11, 2003
-IR 246038; DOA 0010-04 Flooding procedure potentially inadequate; dated July 27, 2004
-IR 261167; DOA 0010-04 Flooding; dated October 7, 2004
-IR 379227; NRC Identifies poor quality closure - CA 246038-09; dated September 28, 2005
-IR 380075; NRC Identifies Discrepancies in DOA 0100-004, FLOODS; dated September 30, 2005
-DOA 0010-04, Flood, Revision 20
-DOA 0040-02, Localized Flooding in Plant, Revision 16
-DOS 1300-04, Operation of the Isolation Condenser External Flood Emergency Make-Up Pump, Revision 04
-PMRQ 166188, D2/3 6Yr Emergency Diesel Pump (Flood Pump) Operation, dated March 2, 2004
-—23, Diagram of Fire Protection Piping, Revision I
-—28, Diagram of Isolation Condenser Piping, Revision LK
-—75, Diagram of Fire Protection Piping, Revision H
-HPCI Flooding Analysis, Calculation No. 0591-576-001
-UFSAR 3.4.1.2, "Internal Flooding Protection Measures"
-Drawings No. FL-1, FL-32 sheets no. 1 & 2, FL-37 sheets 1 & 2

1R11 Operator Requalification

-IR 350738; Ops training evaluation has subjectivity; July 6, 2005
-IR 352411; Two instructors taught class without proper qualifications; dated August 5, 2005
-IR 352607; SOER 90-03 NI miscalibration eff. review - training comments; dated July 13, 2005
-IR 358227; Individual LORT weekly exam grade <80% crew clock reset; dated July 29, 2005

1R12 Maintenance Effectiveness

-IR 343999; EHC system requires maintenance rule (A)(1) eval; dated June 14, 2005
-IR 346162; NOS IDs finding with Dresden implementation of maint rule; dated June 21, 2005
-IR 347980; IR review backlog identified in maint. rule database; dated June 27, 2005
-IR 352313; Maint rule functional failure identified from IR 305692; dated July 12, 2005

-IR 352721; ‘A’ ISCO makeup pump unavailability not documented in MR; dated July 13, 2005
-IR 353596; EHC system maintenance rule (A)(1); dated July 15, 2005
-IR 354419; Unit 3 CCSW system runs ESHIP yellow; dated July 19, 2005
-IR 355086; AEER HVAC system at Max. failure rate in maintenance rule; dated July 20, 2005
-IR 360147; AEER HVAC failed to maintain temperature resulting in MRFF; dated August 4, 2005
-IR 360496; Maintenance rule backlop completion dates not established; dated August 5, 2005
-IR 366647; IEMA inspector ID’s error in past main rule quarterly eval; August 25, 2005

1R15 Operability Evaluations

-IR 349705; XL3 device 51-17 alarm; dated July 2, 2005
-IR 353345; Historical operability of loose bolt on flange IR 352219; dated July 15, 2005
-IR 353638; NOS ID’s deficiency in Op Eval 05-004; dated July 15, 2005
-IR 354104; Eval of cumulative foreign material in reactor vessel; dated July 18, 2005
-IR 360109; HPCI injection valve open permissive SW not properly installed; dated August 4, 2005
-IR 360387; Target Rock test results applicability to Op Eval 04-001; August 5, 2005
-IR 360396; No degradation found of Target Rock SRV; dated August 5, 2005
-ATI 344849-03, “Evaluation of Fire Protection System after Failed Tri-Annual Flow Test,” dated September 21, 2005
-WO 00468178, OP D1/2/3 3Y TS Fire Water System Triennial Flow Test, dated June 3, 2005
-DFPS 1423-08, Fire Water System Flow Test, Revision 15
-IR 344849; High D/P Found During Tri-Annual FP Sys Water Flow Test; dated June 16, 2005

1R22 Surveillance Testing

-WO 00825379-01, DIS 1300-01, “Sustained High Reactor Pressure”
-IR 357427; MSIV indicator reading high out of spec; dated July 27, 2005
-IR 357430; H2O2 monitor out of tolerance; dated July 27, 2005
-DOA 6500-12, “Low Switchyard Voltage,” Revision 09
-Unit 2(3), Appendix X, Revision 25, “Technical Specification Action Statement Initiated Surveillances”
-Technical Specifications 3.8.1, “AC Sources - Operating”
-Operator’s log for September 13, 2005, 1605 entry
-DOP 2000-23, “Drywell Sump Operation,” Revision 13
-Unit 2(3), Appendix A, Revision 99, “Unit NSO Daily Surveillance Log

1R23 Temporary Modification

-M31, Diagram of Fuel Pool Cooling Piping, Revision BL
-Operability Evaluation OPD 97-112, “Concern with Reactor Building Temperatures Post-Loca”
-WO 00814613-01, “Defeat Fuel Pool Cooling Pumps Low Suction Pressure Trips.”

-IR 350179; NOS IDs DAN 2223-6 G-4 implements TCCP outside of CC-AA-112; dated July 5, 2005
-TCCP EC 353888, “Isolate U2 CCSW Pump Vault Cooling Coil 2-57-00-30B to Keep Coil 2-5700-30A Operable,” Revision 000
-Calculation No. VV-14, CCSW for CCSW Cooler Performance and Coil Effectiveness Curve, dated 6/12/1998
-EC Eval #341365, Revision 1
-WO Task 00762027; TMOD - Check for Leaks and Flowrate Per TCPP 353888, dated 03/20/2005
-WO Task 00784151; TMOD353933 Run 3B CCSW PMP Check for leaks and Flowrate, dated 04/13/2005
-UFSAR 9.2 Water Systems, Revision 6
-UFSAR 3.4.1.2.1.2 Isolation of the Containment Cooling Service Water Pump from Flood Water, Revision 6
-IR 379598; NRC Insp Questions Acceptance Criteria in EC TCCP 353888; dated September 29, 2005

2OS1 Access Control to Radiologically Significant Areas

-RWP 10004863; Unit 3 Reactor Core Detector System Maintenance; Revision 0
-RP-AA-460; Controls for High and Very High Radiation Areas; Revision 7
-RP-DR-460-1001; Additional High Radiation Exposure Controls; Revision 0
-RP-AB-460; Transversing In-Core Probe (TIP) Area Access Controls; Revision 0
-IR 342650; Apparent Cause Evaluation, Radwaste Bay 11 Locked High Radiation Area (LHRA) Gate Found Unlocked; Undated Draft
-ATI 342650-02; Prompt Investigation Report for Radwaste Bay 11 Locked High Radiation Area (LHRA) Gate Found Unlocked; Undated Draft
-RWP 10004865 (and associated ALARA files); Unit 3 RWCU Aux Pump Seal Repair and Shaft Work; Revision 0
-RP-AA-210; Dosimetry Issue, Usage, and Control; Revision 5
-RWP 10005146 (and associated ALARA files); U2 Cavity Drain Down/Cavity Decon; Revision 0
-RWP 10004165; D3R18 Reactor Steam Dryer Modification Diving Activities; Revision 0
-Focus Area Self-Assessment Report; Evaluation of Access Control to Radiologically Significant Areas; dated June 22, 2005
-RP-AA-301; Radiological Air Sampling Program; Revision 0
-RP-DR-301-1001; Radiological Air Sampling Program Site Specific Guidance; Revision 0
-NOSA-DRE-05-06; NOS Audit - Health Physics Functional Area; June 20 to July 1, 2005
-IR 00329964; Maintenance Moved RP Barrier/Posting; IR generated in 2005
-IR 00289555; Locked High Radiation Area Gate/Door Repairs (and associated Work Request 00165855); IR generated in 2005
-IR 000342957; Locked High Radiation Door Deficiencies (and associated Work Request 00181172); IR generated in 2005
-IR 00320797; Unit 2 Main Turbine Floor Gate Needs Gate/Frame Enhancement; dated April 4, 2005
-DRS 5600-01; Quarterly High, Locked High and Very High Radiation Posting and Door Checks; Revision 8

-High, Locked High and Very High Radiation Area Boundary and Posting Surveillance Records; January - June 2005
-RP-DR-3001; Guidelines for Monitoring Resin or Fluid Transfer Evolutions; Revision 1
-RP-AA-220; Bioassay Program; Revision 2
-RP-AA-222; Methods for Estimating Internal Exposure From In-Vivo and In-Vitro Bioassay Data; Revision 1
-TID-2005-002; Report of Annual Bioassay Program Review; dated February 21, 2005
-Internal Dose Investigation Results Summary from D3R18 Outage, Associated Whole Body Count Data and Associated Dose Calculations; October and November 2004
-DFP 0800-39; Control of Material/Equipment Hanging in Units 2/3 Spent Fuel Pools; Revision 14
-Dresden Units 2 and 3 Spent Fuel Pool Inventory; June 2005

4OA1 Performance Indicator Verification

-IR 00218264; Electronic Dosimetry Alarm in Unit 2 Drywell; dated May 2, 2004
-LS-AA-2140; Monthly Data Elements for Occupational Exposure Control Effectiveness; Revision 4
-Reports of Electronic Dosimetry Dose Rate and Accumulated Dose Alarms; Reports Generated in May 2004 thru June 2005
-RP-DR-TEC-005; Radiation Exposure Investigation Logs, Attachment B; Records selected reviewed for June 2004 thru June 2005

4OA2 Identification and Resolution of Problems

-AR303699; As Found Setpoint Outside of Tech Spec Allowable
-AR303692; As Found Setpont Outside of Technical Specifications
-AR358734; The Second Stage Disc/Seat was Found Degraded (Steam Cut)

4OA3 Event Follow-up (71153)

-IR 340608; EHC leak at turbine front standard; dated June 2, 2005

4OA5 Other Activities

-Certificate of Compliance (CoC), No 1014, Amendment 2
-Safety Evaluation Report (SER), Amendment 2
-Final Safety Analysis Report (FSAR), Revision 3
-10 CFR 50.59 Evaluation, “Hi-Track to Hi-Storm MPC Transfer at the CTF (ATI #340904-15) & Structural Qualification of Hi-Storm Loading at DNPS Using Hydraulic Lifters,” Revision 1, dated August 19, 2005
-10 CFR 72.48 Screening, “Hi-Track to Hi-Storm MPC Transfer at the CTF (ATI #340904-15) & Structural Qualification of Hi-Storm Loading at DNPS Using Hydraulic Lifters,” Revision 4, dated August 8, 2005
-Calculation Package, “Structural Qualification of Hi-Storm Loading at DNPS Using Hydraulic Lifters (Hi-2053414),” Revision 2, dated August 22, 2005
-Loading Procedure, “Hi-Storm Processing at the CTF,” Revision 17
-Special Procedure, “Hi-Tract to Hi-Storm MPC Transfer at the CTF (ATI # 340904-15),” dated August 22, 2005

-Engineering Evaluation, EC 353243, “10 CFR 21 Notification-Reactor Building Crane,” dated January 28, 2005
-Work Planning Instructions, “EC (DCP) 353340, Rev. 0: WO 773133-01, Design Change Package to Upgrade Main Hoist Equalizer Assembly Reactor Building Crane (10 CFR 21 Notification),” dated February 8, 2005
-Design Analysis, “Reactor Building Crane: 10 CFR 21 Notification-Equalizer Arm Modification (Whiting Calculation),” dated March 16, 2005
-Root Cause Investigation Report, “Cask Transfer Facility (CTF) Lift Stopped with Loaded Cask Due to Lift System Gearing Failure,” dated August 5, 2005

## LIST OF ACRONYMS USED

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|---|---|
| ADAMS | Agencywide Documents Access and Management System |
| CCSW | Containment Cooling Service Water System |
| CFR | Code of Federal Regulations |
| CoC | Certificate of Compliance |
| CTF | Cask Transfer Facility |
| DIS | Dresden Instrument Surveillance |
| DOA | Dresden Operating Abnormal Procedure |
| DOS | Dresden Operating Surveillance |
| FSAR | Final Safety Analysis Report |
| HPCI | High Pressure Coolant Injection System |
| IEMA | Illinois Emergency Management Agency |
| IMC | Inspection Manual Chapter |
| IR | Inspection / Issue Report |
| LER | Licensee Event Report |
| LHRA | Locked High Radiation Area |
| MPC | Multi-purpose Canister |
| MWe | megawatts electrical |
| NCV | Non-Cited Violation |
| NRC | Nuclear Regulatory Commission |
| OSP | Offsite Power |
| PARS | Publicly Available Records |
| PI | Performance Indicator |
| RP | Radiation Protection |
| RPT | Radiation Protection Technician |
| RWP | Radiation Work Permit |
| SDP | Significance Determination Process |
| TI | Temporary Instruction |
| TIP | Traversing In-Core Probe |
| TS | Technical Specification |
| TSO | Transmission System Operator |
| UFSAR | Updated Final Safety Analysis Report |
| VHRA | Very High Radiation Area |
| WO | Work Order |