# Dresden 3
# 4Q/2005 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Aug 12, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification Requirements for Position Verification Not Met**
The inspectors identified a Non-Cited Violation of Technical Specification Surveillance Requirement 3.7.2.1 regarding the failure to periodically verify the position of manual valves. Specifically, the licensee did not verify the correct position of 11 manual valves that were not locked, sealed, or otherwise secured in position in the diesel generator cooling water (DGCW) subsystem flow path associated with the DGCW pump motor coolers. The licensee's corrective actions included verifying and then locking the affected valves in the open position and revising operating procedures to reflect that the affected valves are locked in the open position.

This finding was more than minor because it was associated with the mitigating systems attribute of configuration control, which affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the DGCW system to respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance based on the licensee verifying the valves were in their correct position and screened as Green using the SDP Phase 1 screening worksheet.
Inspection Report# : 2005009*(pdf)*

**Significance:** G Aug 12, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Unanalyzed Diesel Loading Sequence in Operating Procedures**
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," due to the design basis emergency diesel generator (EDG) loading sequence during a loss of coolant accident/loss of offsite power not being correctly translated into procedures or instructions. Specifically, the loss of power procedure provided guidance to operate the plant outside the analyzed EDG loading sequence. The licensee's corrective actions included evaluating the effect of the procedure's unanalyzed load sequence and concluded that the EDG would have been capable of performing its safety function.

This finding was more than minor because it was associated with the attribute of procedure quality, which could have affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the EDGs to respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet.
Inspection Report# : 2005009*(pdf)*

**Significance:** G Jul 25, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**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**
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.
Inspection Report# : 2005010*(pdf)*

**Significance:** G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Modification to the Unit 3 Core Spray Piping**
On December 11, 2004, a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI and Criterion III was identified by the inspectors. The licensee failed to perform post-modification testing and to assure critical aspects of the core spray modification installation, which included obtaining gap measurement for mechanical joints, verifying the capability of the tooling to produce the required surface finishes on pre-fabricated components, and verifying that the pre-fabricated components were properly machined, met the leakage analysis specifications.

The finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, specifically the design control attribute. The finding was of very low safety significance because the licensee was able to demonstrate, with the assistance of General Electric, that there was reasonable assurance that the modification was installed properly. The licensee planned to revise CC-AA-107, "Configuration Change Acceptance Testing Criteria," and/or CC-AA-107-1001, "Post Modification Acceptance Testing." The procedure change would provide that the substitution for post modification testing would ensure quality at least equivalent to that specified in the original design bases. In addition, the licensee planned to confirm that the installed core spray modification had been installed with a level of quality equivalent to the original design basis.
Inspection Report# : 2005008*(pdf)*

**Significance:** G Jun 29, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Isolation Condenser Time Delay Relays Exceed TS Value**
On September 29, 2004, a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified by the inspectors. The licensee had implemented inadequate corrective actions for a deficient condition quality that occurred on September 6, 1996, to prevent recurrence of a similar deficient condition that occurred on September 29, 2004. Both events involved the failure of safety related time delay relays to meet acceptance criteria due to the use of a stopwatch as a tool for calibration of safety related equipment. The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution.

The finding was greater than minor because it impacted the mitigating system cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events and because it affected the reliability of a safety related component. As a result of the 2004 event, the licensee initiated issue report 258172, created an action item to review the root cause of the event, revised the isolation condenser initiation time delay relay calibration procedure to require the use of a strip chart recorder, and created an action item to evaluate the extent of condition. The finding was of very low safety significance because the isolation condenser system did not lose the ability to perform its safety function and all other mitigating systems were available.
Inspection Report# : 2005008*(pdf)*

**Significance:** G Apr 01, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Safe Shutdown Procedure Failed to Specify Correct Number of Turns for Opening Valve**
Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B requirements. The licensee failed to specify the correct number of turns in a hot shutdown procedure for partially opening a valve relied upon to mitigate a fire. The incorrect number of turns specified in the procedure could have caused a significant delay in performance of safe shutdown actions in the event of a fire. Once identified, the licensee entered the finding into their corrective action program to revise the affected procedures.

This finding was more than minor because the procedural error could have caused a significant delay in the performance of safe shutdown actions in the event of a fire. The issue was of very low safety significance because the licensee's analysis showed that sufficient margin remained for the performance of the safe shutdown actions. The finding was a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, which required procedures affecting quality to be of a type appropriate to the circumstances. (Section 1R05.5b)

Inspection Report# : 2005002*(pdf)*

**Significance:** G Apr 01, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Perform Post-Maintenance Test on the 3B Reactor Recirculation Pump Seals**
A self-revealing finding involving a non-cited violation 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.
Inspection Report# : 2005010*(pdf)*

**Significance:** G Jan 05, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Performance Deficiency While Performing Surveillance Procedure DIS 700-02, "APRM/RBM [average power range monitor/rod block monitor] Flow Instrumentation Total Drive Flow Adjustment," Revision 16**
On January 5, 2005, a performance deficiency involving a Non-Cited Violation of Technical Specification 5.4.1 was self revealed when instrument maintenance technicians were performing Dresden Instrument Surveillance 700-02, "APRM/RBM [average power range monitor/rod block monitor] Flow Instrumentation Total Drive Flow Adjustment," Revision 16. The technicians misadjusted the recirculation flow signal to the reactor protection system which required entry into Technical Specification 3.3.1.1 Limiting Condition for Operation A.1 and C.1 for Average Power Range Monitor Channels 1, 2, and 3 Flow Bias Trips. The instrument maintenance technicians were using the averaging function of a Fluke 189 digital multi-meter. The technicians had not been trained on how to use the function and the procedure did not provide instructions on how to use the multi-meter. The mis-use of the averaging function resulted in adjusting the recirculation flow converter signal too high.

The finding was greater than minor because it impacted the Mitigating System Cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events and because it affected the procedure quality of a surveillance test procedure. The finding was of very low safety significance because it impacted the reactor protection system for a time period of less than 1 minute. The surveillance test procedure was changed to include instructions on how to use the averaging function of a digital multi-meter and the instrument maintenance technicians were briefed on this event and trained on how to use the averaging function of the digital multi-meter. (Section 1R22)
Inspection Report# : 2005003*(pdf)*

**Significance:** W Jan 30, 2004
Identified By: NRC
Item Type: AV Apparent Violation
**Failure to properly evaluate extended power uprate for its impact on post scram reactor vessel water level to prevent water intrusion into HPCI steam supply line**
An apparent violation (AV) of 10 CFR 50, Appendix B, Criterion III, Design Control, having a preliminary low to moderate safety significance (White) was identified as a result of the inspectors' review of a January 30, 2004, scram event. Water intrusion into the high pressure coolant injection (HPCI) system turbine steam supply line occurred as a result of the scram and rendered the HPCI system inoperable. The inspectors determined that the licensee implemented extended power uprates on Unit 2 in 2001 and Unit 3 in 2002, but failed to verify the adequacy of design for the implementation of extended power uprate to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the HPCI steam supply line.

The finding was determined to be greater than minor because it impacted the mitigating systems cornerstone. The finding was preliminarily determined to be of low to moderate safety significance following the performance of a case-specific Phase 3 SDP evaluation. Corrective actions taken by the licensee included modifying the feedwater level control system post-scram level setpoints and dynamic modeling of the reactor vessel level response.
Inspection Report# : 2005014*(pdf)*

# Barrier Integrity

**Significance:** G Feb 08, 2005
Identified By: NRC
Item Type: FIN Finding
**Failure of the Refuel Floor Damper & Design Deficiency with the Standby Gas Treatment System**
On February 8, 2005, a performance deficiency was identified by the inspectors. The licensee failed to identify the failure of the refuel floor damper in the reactor building ventilation system in a timely manner which resulted in the late discovery of a design deficiency with the standby gas treatment system. The standby gas treatment system used reactor building ventilation ductwork before directing air flow to the standby gas treatment filters. The refuel floor damper would throttle down, per design, to ensure a local negative differential pressure in the reactor water cleanup heat exchanger rooms with respect to the refuel floor. As a result, air flow to the standby gas treatment system was significantly restricted and affected the standby gas treatment recovery time for the entire secondary containment. The damper failed prior to 2003, masking the design deficiency, and was unnoticed until February 2005. Also, inadequate inspections of the dampers in the reactor building ventilation system during operation of the standby gas treatment system contributed to the late discovery of this design issue. The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution.

The finding was greater than minor because, if left uncorrected, the failure to identify deficient plant equipment would become a more significant safety concern because important systems could be rendered inoperable and because it impacted the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. In addressing this issue, the licensee gagged each unit's refuel floor damper open to 80 percent to ensure adequate air flow to the standby gas treatment system. The finding was of very low safety significance because the standby gas treatment system was always able to restore secondary containment differential pressure within the Technical Specifications allowed outage time of four hours.
Inspection Report# : 2005008*(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Jun 08, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Ensure That a Gate to a Posted LHRA was Secured Following Work in the Area**
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.
Inspection Report# : 2005010*(pdf)*

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# Public Radiation Safety

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# Physical Protection

Physical Protection information not publicly available.

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# Miscellaneous

Last modified : March 03, 2006