Dresden 3
2Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Review of Unit 2/3 Condensate Storage Tank Overflow Event
A performance deficiency involving a non-cited violation of Technical Specification (TS) 5.4.1, was self revealed on November 20, 2006, when during the Unit 3 hotwell drain down to support Unit 3 startup, operators allowed the 2/3 condensate storage tank (CST) to overflow into the Unit 2 turbine building equipment drain system, which eventually overflowed onto the condensate/condensate booster pump room floor. Corrective actions by the licensee included implementing a standard pre-job brief to address risk and contingencies for this infrequently performed evolution. Additional actions involved revising procedures to include appropriate actions if the 2/3 CST high water level alarm is exceeded and limitations and precautions associated with proper 2/3 CST level, hotwell level, and emergency reject operations.

The finding was considered more than minor because it could be reasonably viewed as a precursor to a significant event. The finding was determined to be of very low safety significance because the event did not affect credited items for shutdown safety. This finding was related to the cross-cutting issue of human performance (resources) because the licensee did not provide complete, accurate and up-to-date procedures to plant personnel.
Inspection Report# : 2006011 *(pdf)*

Significance: G Oct 06, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Inappropriate Basis in 10 CFR 50.59 Evaluation for Temporary Modification
The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 “Changes, Tests, and Experiments,” having very low safety significance (Green) for the licensee’s failure to perform an adequate safety evaluation review for changes made to the facility per Temporary Modification EC TCCP 354622. Specifically, the licensee failed to appropriately evaluate the installation of a temporary jumper at the Electro-Hydraulic Control (EHC) Card 2-5640-A37 to bypass the function of the “A” Main Steam Pressure Regulator (MSPR). The licensee’s 10 CFR 50.59 safety evaluation 2005-01-001 failed to provide a basis as to why the activity which bypassed one of the two MSPRs did not present more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR).
Inspection Report# : 2006012 *(pdf)*

Significance: G Sep 07, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Identify and Correct Issues With the Operation and Testing of the Diesel Driven Pump Used to Respond to External Flooding
On September 7, 2006, the inspectors identified a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to promptly identify and adequately correct issues with the operation and testing of the isolation condenser emergency make-up pump until prompted by the inspectors. The pump is designed to ensure an adequate supply of make-up water to the isolation condenser during flood conditions to prevent core damage. Specifically, the licensee failed to ensure adequate corrective actions were taken to test the pump to its design limits, and failed to identify deficiencies with the suction hose to eliminate friction losses in the suction line. The licensee missed an opportunity to re-evaluate the losses in the suction line when the required flowrate was increased to 350 gpm in 2003 as a result of extended power uprate and consequently, failed to

ensure that equipment associated with the pump was sized adequately for execution of the pump’s safety function. The licensee’s corrective action for this issue included increasing the margin in net positive suction head (NPSH) by increasing the size of the pump suction line to a 30-foot length of 4-inch diameter hose, eliminating restrictions in the flowpath to the isolation condenser, and planning to review DOA 0010-04 and revise the procedure as appropriate to address the height above the water to which the pump could be elevated to maintain NPSH.

This finding was more than minor because it affected the equipment performance and procedure quality attributes of the Mitigating Systems cornerstone, and affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The issue was of very low safety significance based on the low initiating event probability and, because of the slow onset of the flooding and the reduced decay heat in the reactor core at the time recovery actions would be necessary, the licensee would be able to reasonably perform recovery actions that would prevent core damage. The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution (corrective action program) because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity (P.1(d)).
Inspection Report# : 2007003 *(pdf)*

Mitigating Systems

Significance: SL-IV Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump Room High Energy Line Break
The inspectors identified a finding having very low safety significance and an associated Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 for the licensee’s failure to perform an adequate safety evaluation review for changes made to the facility per safety evaluation 2005-02-001. Specifically, the licensee failed to provide an adequate basis as to why changes that credited the isolation condenser for decay heat removal in lieu of the automatic depressurization system and low pressure coolant injection (LPCI)/containment cooling service water and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI during a postulated high pressure coolant injection (HPCI) room high energy line break (HELB) did not require a license amendment. The licensee entered this issue into its corrective action program.

Because the issue potentially impacted the NRC’s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity implemented per 10 CFR 50.59 safety evaluation 2005-02-001, which adversely affected systems important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC’s Inspection Manual Chapter (IMC) 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and answered “no” to all of the questions under the Mitigating Systems cornerstone. Based upon this Phase 1 screening, specifically this issue did not represent a loss of function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation, the inspectors concluded that the issue was of very low safety significance (Green). The inspectors determined there was not a cross-cutting aspect to this finding. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.
Inspection Report# : 2007003 *(pdf)*

Significance: G Jun 12, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Restore Fire Brigade Equipment to a Ready Status in a Timely Manner after the Performance of a Fire Drill
The inspectors identified a Non-Cited Violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee’s failure to restore fire brigade equipment to a ready status in a timely manner after the performance of a fire drill on June 12, 2007. Corrective

actions by the licensee included the restoration of the fire brigade equipment to a ready status at 8:22 p.m. on June 13, 2007.

The inspectors determined that this finding was more than minor because the failure to restore the fire brigade equipment to a ready status, if left uncorrected, would become a more significant safety concern. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, dated March 23, 2007. The inspectors determined that the finding affected the Mitigating Systems cornerstone and the fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 1, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection Significance Determination Process,” and require management review. Therefore, the finding was reviewed by NRC management, and was determined to be a finding of very low safety significance (Green) because the condition existed for slightly more than 24 hours, the delay in getting additional self-contained breathing apparatus would be a maximum of about 10 minutes, and the majority of the safety significant equipment in the turbine building is protected by an automatic fire suppression system. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the licensee failed to ensure adequate supervisory and management oversight of work activities (restoration of the fire brigade equipment), such that nuclear safety was supported (H.4(c)).
Inspection Report# : 2007003 *(pdf)*

Significance: G May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Acceptance Criteria in 125 VDC Station Battery Service Test Procedures
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” Specifically, the licensee failed to incorporate the 125 VDC system minimum required voltage value as the acceptance criteria for the minimum battery terminal voltage inservice test procedure DES-8300-28 “Unit 2 - 125 Volt Main Station Battery Service Test.” Following discovery, the licensee entered the issue into its corrective action program to revise the station batteries test procedures to include the minimum required voltage values.

This finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because, if the finding was left uncorrected it would become a more significant safety concern. Specifically, the failure to ensure that the battery terminal voltage during the battery discharge per the service test did not drop below the 125 system design input value could have affected the operability of safety-related equipment in the event of a design basis accident condition. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.”
Inspection Report# : 2007006 *(pdf)*

Significance: G May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Adequate Control Voltage for 4160 Breaker’s Closing Coil was not Assured
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee failed to assure and verify that the minimum available control voltage at the 4160 circuit breakers was adequate for the closing coils to close the breakers, following a design basis accident and loss of offsite power condition. Following identification of this issue, the licensee obtained a letter from the vendor (General Electric Nuclear Energy) suggesting that it was reasonable to conclude that the closing coils will operate at Dresden’s minimum available voltage (58 volt) level based on ageing testing conducted in 1999 and 2007 testing of one of Dresden’s breakers.

This finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Disposition Screening,” because the finding was associated with the Mitigated Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring capability and reliability of systems that respond to initiating events. Specifically, the failure to assure adequate control voltage was available to close the 4160 breakers would have affected the capability of emergency diesel generators and other safety-related equipment to respond to initiating events. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A,

"Significance Determination of Reactor Inspection Findings for At-Power Situations
Inspection Report# : 2007006 *(pdf)*

Significance: G May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Procedurally Control Regulatory Guide 1.97 Control Board Labeling
The inspectors identified a performance deficiency involving a Non-Cited Violation of Technical Specification (TS) 5.4.1 for the licensee's failure to provide procedural controls for the unique identification of Regulatory Guide (RG) 1.97 post-accident instrumentation to aid the control room operator. Specifically, the licensee failed to adequately control the labeling on both unit's control panels and the simulator, resulting in several improperly marked post-accident indicators.

The finding was greater than minor because, if left uncorrected, it could become a more significant safety concern. Inaccurately labeled control room indicators of RG 1.97 post-accident instrumentation could lead to confusion and hamper the response of operators if conflicting indications resulted due to accident conditions. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations."
Inspection Report# : 2007006 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform 50.59 Evaluation of Non-Code Conforming Buried HPCI Piping (Section 1R02)
The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.59(d)(1) for the licensee's failure to document an evaluation which provides the basis for the determination that a change, test, or experiment did not require a license amendment. Specifically, the licensee's 10 CFR 50.59 screening failed to provide an evaluation as to why the installation of the high pressure coolant injection (HPCI) suction piping, which did not meet USAS B31.1 Code requirements, did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a Structure, System, or Component (SSC) important to safety. The licensee entered this issue into the corrective action program and planned to do additional weld metal tensile and bend tests on a remnant piece of the non-conforming HPCI pipe. The licensee intended to perform this testing to demonstrate quality levels equivalent to that prescribed by the USAS B31.1 Code.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that this change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The licensee considered the nonconforming replacement pipe operable, based upon satisfactory hydrostatic tests of the installed pipe to demonstrate structural and leakage integrity at the time of installation. The inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated November 22, 2005, and answered "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV Violation.
Inspection Report# : 2007002 *(pdf)*

Significance: G Oct 06, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
EQ Binder Failed to Include Conductor Temperature Rise
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" having very low safety significance (Green) for the licensee's failure to evaluate and include the conductor temperature rise for the 5KV cables for the CS and LPCI pump motors in the Equipment Qualification Binder EQ-04D. The EQ Binder used the cable design limit of 194 degrees F in calculating the qualified life of the 5KV cables instead of the sum of the

conductor temperature rise and the ambient temperature, during and post accident, which together exceeded the cable design limit.
Inspection Report# : 2006012 *(pdf)*

Barrier Integrity

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Licensee's Failure to Develop a Pre-fire Plan for Fire Zone 8.2.6.A, Elevation 534'
A performance deficiency involving a non-cited violation of the Dresden Nuclear Power Station Renewed Facility Operating License was identified by the inspectors due to the licensee's failure to develop a pre-fire plan. Specifically, on November 17, 2006, the inspectors identified that the licensee failed to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534'. The licensee has since developed a pre-fire plan for the Fire Zone 8.2.6.A, Elevation 534'.

This finding was considered more than minor because it involved the Barrier Integrity attribute of procedural quality for the control room ventilation system because the failure to develop a pre-fire plan for Fire Zone 8.2.6.A could have adversely impacted the fire brigade's ability to fight a fire. The finding was related to the performance of the fire brigade and was not suitable for SDP evaluation. Therefore, the finding was reviewed by NRC management and determined to be of very low safety significance because no safe shutdown equipment was located in this fire zone.
Inspection Report# : 2006011 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Sep 30, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Satisfy Technical Specification LHRA Access Requirements During Entry Into a Steam Sensitive Area at Power
A self-revealed finding of very low safety significance, and an associated violation of NRC requirements were identified for the failure to satisfy Technical Specification requirements for access into a high radiation area with dose rates in accessible areas greater than 1000 mrem/hour. As a result, a worker was allowed to enter a steam sensitive area at power that was controlled as a locked high radiation area (LHRA), without adequate recognition of the area radiological conditions and without positive radiological control over the activities within the area. The electronic dosimetry (ED) worn by the worker alarmed when significantly higher than expected dose rates were encountered, resulting in some unnecessary dose to that worker.

The issue was more than minor, because it was associated with the Program/Process attribute of the Occupational Radiation Safety Cornerstone, and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation. The issue represents a finding of very low safety significance because it did not involve ALARA Planning or work controls, there was no overexposure, nor did a substantial potential for an overexposure exist given the radiological conditions in the area and the worker's response to the ED alarm. Also, the licensee's ability to assess worker dose was not compromised. A Non-Cited Violation of TS 5.7.1 was identified for the failure to comply with the requirements for access into a high radiation area with dose rates accessible to personnel greater than 1000 mrem/hour. Corrective actions taken by the licensee included modification to the survey maps for steam sensitive areas, tagging of certain LHRA keys to remind radiation protection staff to coordinate entries

into these areas with operations staff, and plans to reevaluate the radiation protection department practices for entry into steam sensitive areas, and in general for entry into high radiation areas with the potential for significant dose rate gradients.
Inspection Report# : 2006010 *(pdf)*

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : August 24, 2007