Dresden 3
3Q/2009 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2009
Identified By: NRC
Item Type: FIN Finding
Failure to Identify and Replace CR120A Relays as Recommended by GE SIL 229 Supplement 1
A finding of very low safety significance was identified by NRC Inspectors for the licensee's failure to identify and replace several CR120A relays as recommended by GE SIL 229 Supplement 1. Specifically, the licensee failed to replace several CR120A relays associated with primary containment valve isolation logic which eventually resulted in a partial Group 2 logic isolation event. The licensee entered this issue into the corrective action program (CAP) as Issue Report 923691. The licensee plans to replace these CR120A relays. There was no enforcement action associated with this finding.
This finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affected the cornerstone's objective to limit the frequency of those events that upset plant stability and challenge critical safety functions during power operations. The relay failure caused an unplanned partial Group II primary containment isolation that impacted plant operations for several days. This issue was determined to be of very low safety significance since it did not contribute to both a reactor scram and loss of a mitigating function when evaluated as a Transient Initiator.
Inspection Report# : 2009004 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Instrument Air Isolation Valve Mispositioning on April 26, 2009
A finding of very low safety significance and associated Non Cited Violation of Technical Specification Section 5.4.1 was self revealed when the Unit 2 instrument air system had a significant pressure drop because a non licensed operator failed to follow procedure DOP 4700 01, "Instrument Air System Startup," Revision 46. The violation was placed into the licensee's corrective action program (CAP) in Issue Reports 911794 and 893376. The non licensed operator was relieved from duty. Both the non licensed operator and the unit supervisor were counseled for the failure to perform expected work practices. The licensee also found that this event was similar to other problems discussed in the licensee's Root Cause Report 893376, "Operations Cyclic Performance." Multiple corrective actions were assigned in Root Cause Report 893376 to address a lack of operations supervision enforcing department standards.
Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated December 4, 2008, the inspectors determined that the finding was more than minor because the finding could be reasonably viewed as a precursor to a significant event. Specifically, the failure to follow procedure resulted in an instrument air (IA) transient that could have resulted in a unit scram if the IA system had not been recovered in a timely manner. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 Initial Screening and Characterization of findings," Table 4a, for the Initiating Event Cornerstone. The inspectors determined that the finding represented an increase in the likelihood of a reactor trip and the likelihood that mitigation equipment would be unavailable because the finding increased the likelihood of a loss of instrument air (LOIA) event. Therefore, the finding required a phase 2 SDP evaluation. The duration of the condition was less than three days. Using the SDP usage rules from IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At Power Situations", the inspectors increased the initiating event frequency for the LOIA event by one order of magnitude for the three day exposure period. The result was an estimated change in core damage frequency of less than 1.0E 6/yr. As a result, the finding was determined to be of very low safety significance (Green) based on the phase 2 SDP evaluation. This finding had a cross cutting aspect in the area of Human Performance, Work Practices because the operator did not use the expected human performance techniques.
Inspection Report# : 2009003 *(pdf)*

Significance: G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Freeze Seal Established Prior to Meeting The Requirements of Procedure MA-AA-736-610
A finding of very low safety significance and associated Non Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified by NRC inspectors on November 12, 2008, when the licensee had declared a freeze seal established prior to meeting the requirements of procedure MA AA 736 610, “Application of Freeze Seal to All Piping,” Revision 3. The licensee took corrective actions that included counseling the first line supervisor and the engineer involved in the work.
The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, the inspectors determined that the licensee had determined the freeze seal to be acceptable before it was allowed by procedure. Had there been a problem with the freeze seal, there may not have been adequate time to react and implement any required contingency actions. The inspectors concluded this finding was associated with the Initiating Events Cornerstone. This finding has a cross cutting aspect in the area of Human Performance, H.1.b, because the licensee did not make a conservative assumption in decision making

Inspection Report# : 2008005 *(pdf)*

Mitigating Systems

Significance: W Jul 15, 2009
Identified By: NRC
Item Type: VIO Violation
Inadvertent Control Rod Movement While Shutdown
A finding that has preliminarily been determined to be White, a finding with low to moderate safety significance, was self-revealed on November 3, 2008, when the licensee failed to prevent inadvertent and uncontrolled control rod withdrawal by non-licensed operators. After the finding was self-revealed, the control rods were returned to the full-in position to ensure there was no immediate safety concern and the licensee implemented corrective actions, including conducting a prompt investigation. The finding is also associated with five apparent violations of NRC requirements specified by 10 CFR 50.54(j), Technical Specification 3.1.1, and Technical Specification 5.4.1.

The performance deficiency was determined to be more than minor because licensed operators did not maintain configuration control of the control rods when non-licensed operators were able to inadvertently cause control rods to move. Because probabilistic risk assessment tools were not well suited for this finding, the criteria for using IMC 0609, Appendix M, “Significance Determination Process Using Qualitative Criteria,” were met. Based on the additional qualitative circumstances associated with this finding, regional management concluded the finding was preliminary low to moderate safety significance (preliminary White).

The performance deficiency was determined to have resulted from several causes; however, the primary cause was determined to involve the ineffective use of operating experience. This finding has a cross cutting aspect in the area of problem identification and resolution, operating experience, because the licensee did not effectively implement and institutionalize operating experience through changes to station processes, procedures, and training programs. (P.2(b)) (Section 40A2)

Final Significance Determination letter (White) issued on 10/26/2009 with the following as NOV text:

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted from May 8 to July 15, 2009, violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the violations are listed below:

A. 10 CFR 50.54(j) requires that apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor, shall be manipulated only with the knowledge and consent of an operator or senior operator, licensed in accordance with 10 CFR Part 55 present at the controls.

Contrary to the above, on November 3, 2008, mechanisms other than controls which affected the reactivity of the reactor were manipulated without the knowledge and consent of a licensed operator or senior operator present at the controls. Specifically, non-licensed operators manipulated the control rod drive system hydraulic control unit insert riser isolation valves and the withdraw riser isolation valves, an action which affected the reactivity of the reactor in that the valve manipulations caused three control rods, D-7, E-7, and E-6 to move out of the core to positions 06, 18, and 16, respectively. The valve manipulations were accomplished without the knowledge and consent of a licensed operator or senior operator present at the controls.

B. Technical Specification 3.1.1 requires, in part, that the shutdown margin shall be = 0.38 ?k/k, with the highest worth control rod analytically determined or = 0.28 ?k/k, with the highest worth control rod determined by test. Technical Specification 3.1.1, Action Statement D, requires, in part, that if the shutdown margin is not within limits in Mode 4, then initiate action to fully insert all insertable rods immediately.

Contrary to the above, on November 3, 2008, with the reactor in Mode 4, the shutdown margin was not = 0.38 ?k/k and the licensee failed to initiate immediate actions to insert control rods. Specifically, based on the defined shutdown margin conditions of xenon free, temperature of 68°F, highest worth rod fully withdrawn and accounting for the reactivity worth of the actual control rod pattern, the reactor would have been critical.

C. Technical Specification 5.4.1, “Administrative Controls,” requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. RG Guide 1.33, Appendix A, Paragraph 4, “Procedure for Startup, Operation, and Shutdown of Safety-Related BWR Systems,” requires, in part, that instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation should be prepared, as appropriate, for systems, including the control rod drive system.

RG Guide 1.33, Appendix A, Paragraph 9, “Procedures for Performing Maintenance,” Item (a), requires, in part, that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Item (e) requires, in part, that general procedures should be prepared which should include information on areas such as the method for obtaining permission and clearance for operation personnel to work and for logging such work.

Contrary to the above, on November 3, 2008, maintenance that affected the performance of the control rods, which are safety related equipment, was performed in accordance with a written procedure that was not appropriate to the circumstances. Specifically, the maintenance activity informed the workers to use Procedure DOP 0500-05, “Discharging CRD Accumulators with Mode Switch in Shutdown or Refuel,” Revision 5, a procedure prepared in accordance with Regulatory Guide 1.33, Appendix A, Paragraph 4, to isolate each of the 177 hydraulic control unit (HCU) accumulators. This procedure was not appropriate to the circumstances, in that the procedure did not contain any guidance regarding monitoring of control rod drive (CRD) system pressure, did not contain any guidance for ensuring the control room operators were aware of the CRD accumulator activities, did not contain any precautions that the manipulation of HCU valves could affect reactivity, and did not specify how many HCUs could be isolated or whether a control rod drive pump should be operating. As a result, isolating all of the HCUs in accordance with the procedure caused the inadvertent withdrawal of three control rods.

D. Technical Specification 5.4.1, “Administrative Controls,” requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. RG Guide 1.33, Appendix A, Paragraph 1, “Administrative Procedures” lists “Authorities and Responsibilities for Safe Operation and Shutdown” as a subject which requires a written procedure. Procedure OP-AA-103-102, “Watch Standing Practices,” Revision 8, is the implementing procedure for ensuring authorities and responsibilities for safe operation and shutdown. Section 4.3.2 of Procedure OP-AA-103-102 requires operators to aggressively investigate annunciators and alarms to fully understand the reason for any alarm that comes in and to accept all alarms as correct until demonstrated otherwise.

Contrary to the above, on November 3, 2008, the control room operators failed to implement Section 4.3.2 of Procedure OP-AA-103-102 in that they did not aggressively investigate annunciators and alarms and did not accept the alarms as correct until demonstrated otherwise. Specifically, the control room operators did not aggressively investigate multiple rod-drift alarms to ensure they understood the reason for the alarms and failed to accept the alarms as correct until demonstrated otherwise until after three control rods had moved partially out of the full-in position.

E. Technical Specification 5.4.1, “Administrative Controls,” requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

RG Guide 1.33, Appendix A, Paragraph 6, “Procedures for Combating Emergencies and Other Significant Events,” lists “Inability to Drive Control Rods” as a subject which required a written procedure.
Contrary to the above, on November 3, 2008, the licensee failed to implement its written procedure which addressed the inability to drive control rods. Specifically, the control room operators verbally directed a non-licensed operator to open the affected HCU insert valve in order to cause the control rod to insert into the core, and then to re-shut the valve, without implementing a procedure.

These violations are associated with a White finding.
Inspection Report# : 2009009 *(pdf)*
Inspection Report# : 2009010 *(pdf)*

Significance: G May 22, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Diesel-Driven Fire Pump Discharge Valve Found Out of Position
A finding of very low safety significance and associated non-cited violation of license conditions 2.E and 3.G for Units 2 and 3, respectively, was identified by the inspectors for the failure to restore the Unit 1 diesel-driven fire pump to an operable condition within 7 days as required by Technical Requirements Manual (TRM) 3.7.i.A.1. Specifically, the Unit 1 fire pump discharge valve was found closed rendering the pump inoperable for greater than 7 days. Upon discovery of the valve in the closed position the licensee repositioned the valve in the correct locked open position and initiated Action Requests (AR) 922581 and 922585.
This finding is more than minor because the failure to provide the two required fire pumps could have resulted in a failure of the station’s water based fire protection system should the Unit 2/3 fire pump have been out of service at the same time. The finding screened as very low safety significance because the performance of the system was not affected by the closed valve as the Unit 2/3 diesel-driven fire pump remained operable to provide water to the station’s fire protection system, if required. This finding has a cross-cutting aspect in the area of human performance, work control because the licensee did not properly plan and coordinate activities consistent with nuclear safety. Specifically, the licensee failed to restore the Unit 1 diesel-driven fire pump to an operable condition within 7 days as required by TRM 3.7.i.A.1 as a result of ineffective communications between licensee personnel to verify that valve 1-4199-109 was in its correct locked open position prior to declaring the pump operable [H.3(b)].
Inspection Report# : 2009006 *(pdf)*

Significance: G Mar 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Develop a Pre Fire Plan for Fire Zone 18.6
The inspectors identified an NCV of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance for the licensee’s failure to develop a pre fire plan for Fire Zone 18.6. This issue was entered into the licensee’s CAP as issue reports 873977 and 875688. The licensee’s corrective actions included the development of a pre fire plan for Fire Zone 18.6.

The finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., fire), where the failure to develop a pre fire plan for Fire Zone 18.6 could have adversely impacted the fire brigade’s ability to fight a fire. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 0609.04. However, as discussed by Attachment 0609.04, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection SDP,” and require management review. The finding was reviewed by NRC management, and was

determined to be a finding of very low safety significance because no safe shutdown equipment was located in this fire zone. The inspectors determined that this issue also affected the cross cutting area of Problem Identification and Resolution (e.g., corrective action program) because the licensee failed to thoroughly evaluate the problem addressed in NCV 05000237/2008008 02; 05000249/2008008 02, “Failure to Develop a Pre fire Plan for Fire Zone 18.6,” such that appropriate corrective actions to address safety issues and adverse trends were not taken in a timely manner, commensurate with their safety significance and complexity.
Inspection Report# : 2009002 *(pdf)*

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Implement and Maintain in Effect All Provision of the Approved Fire Protection Program as described in the UFSAR
A self revealed NCV of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance was identified for the licensee’s failure to implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee failed to ensure that the floor penetrations to Fire Zone 2.0 were sealed as described in the Fire Hazards Analysis. Licensee corrective actions included revising the Fire Hazard Analysis and sealing the floor penetrations.

The finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., flood hazard, fire) and impacted the Mitigating Systems objective to ensure availability, reliability, and capability of systems that respond to initiating events (i.e., flood hazard, fire) to prevent undesirable consequences. The inspectors performed a Phase 1 qualitative screening and the finding screened to very low safety significance. The inspectors determined that because the modifications took place in the 1985 to 1986 timeframe, the performance deficiency is not reflective of current licensee performance and therefore no cross cutting area was affected.
Inspection Report# : 2009002 *(pdf)*

Significance: SL-IV Jan 15, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Provide Complete and Accurate Information to the NRC Associated with Verifying No Operating Test Item Duplication with the Audit Test
The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 55.40, “Implementation,” 10 CFR 50.9, “Completeness and accuracy of information,” and 10 CFR 55.49, “Integrity of examinations and tests.” For the Dresden Station March 2009 NRC Initial Operator License Examination, the inspectors identified that the examination author and the facility reviewer had initialed Step 2.b and Step 3.a.(3) of Form ES-201-2, “Examination Outline Quality Checklist,” on August 15, 2008, and August 19, 2008, respectively, and Step 1.c of Form ES-301-3 “Operating Test Quality Checklist,” on January 15, 2009, and January 20, 2009, respectively, which indicated that the operating test did not duplicate items from the applicants’ audit test, when, upon NRC review, it was determined that six of the 23 dynamic simulator scenario events, and one of the 15 Job Performance Measures (JPMs) for the Reactor Operator (RO) candidates were duplicated from the applicants’ audit test.

The finding was determined to be more than minor, because the integrity of the NRC initial operator licensing examination could have been compromised if, but for detection by the NRC examiners, the NRC examination had been administered with the duplication of the operating test items from the applicants’ audit test. The finding was determined to be of very low safety significance because the duplication of operating test items was discovered by the NRC examiners prior to administration of the NRC examination, the duplicate test items were either removed from the audit test or the NRC exam changed to remove the duplication, and the facility implemented examination security requirements for the audit test similar to that which was required for the NRC examination. The inspectors concluded that this finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because the licensee did not define and effectively communicate expectations regarding procedural compliance and for personnel to follow procedures (i.e., in the development of the NRC initial operator license examination).
Inspection Report# : 2009301 *(pdf)*

Significance: G Dec 31, 2008

Identified By: NRC
Item Type: NCV NonCited Violation
LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving the low pressure coolant injection (LPCI) heat exchangers cooling capability during a design basis loss of coolant accident (LOCA). Specifically, the licensee failed to evaluate the effects of higher containment pressure post power up-rate on the LPCI heat exchangers' differential pressure set-point calculation. In response to the issue, the licensee implemented compensatory actions including updating various calculations and performing several operability evaluations.
This finding was more than minor because there was reasonable doubt on the operability of the LPCI heat exchangers and if left uncorrected, these heat exchangers had the potential to be inoperable during the summer months. This finding was of very low safety significance because the inspectors determined that the LPCI heat exchangers were in a non-conforming but operable condition and the issue screened as Green using the SDP Phase 1 screening worksheet.

Inspection Report# : 2008005 *(pdf)*

Barrier Integrity

Significance: G Sep 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Technical Specification 5.5.2 Implementing Procedures
The inspectors identified several examples of failure to follow the procedures that implemented Technical Specification (TS) 5.5.2, "Primary Coolant Sources Outside Containment." These failures were determined to represent a Green finding and a non cited violation. Planned corrective actions associated with this violation included, but were not limited to: a revision to DTP 09, "Leak Detection and Reduction Program," to restore commitments made to the NRC; changes to the work control program to ensure that leaks identified by the Leakage Reduction Program are given a high priority; assignment of a program owner; revising operating surveillances to ensure they meet the requirements of TS 5.5.2; initiating a training program for operations and engineering personnel on TS 5.5.2; and developing an administrative limit on emergency core cooling system leakage outside the primary containment.
The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, the failure to track, trend, and repair leakage outside primary containment could lead to exceeding radiation exposure limits in the event of an accident. This finding was determined to have very low safety significance because the actual emergency core cooling system leakage outside the primary containment was low. This finding had a cross cutting aspect in the area of Human Performance, Work Practices because the licensee did not effectively communicate expectations regarding procedural compliance with regard to TS 5.5.2, "Primary Coolant Sources Outside Containment." Specifically, licensee personnel failed to follow several procedural requirements because they were unaware of the requirements.
Inspection Report# : 2009004 *(pdf)*

Significance: G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Have a Procedure to Sample and Establish Administrative Controls for pH in the Torus
The inspectors identified a finding of very low safety significance involving a Non Cited Violation of Technical Specification 5.4.1 for the failure to include essential information in procedures CY AB 120 310, "Suppression Pool/Torus Chemistry," and CY DR 120 31, "Suppression Pool/Torus Chemistry," to ensure torus pH values were above 5.6 in support of the radiological consequence dose analyses as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." As corrective actions, the licensee changed procedures CY-AB-120-310 and CY-DR-120-31 to include essential information for sampling the torus and revised the methodology for calculating torus pH.

Using IMC 0612, Appendix E, "Examples of Minor Issues," issued on September 20, 2007, and Appendix B, "Issue

Screening," issued on December 4, 2008, the inspectors determined that this finding was more than minor because there was reasonable doubt on the operability of the standby liquid control system and its ability to maintain torus pH above 7 following a loss of coolant accident and because of significant programmatic deficiencies in the licensee's corrective action program. The inspectors also determined that this finding impacted the Barrier Integrity objective to provide reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events. The failure to maintain adequate procedures addressing torus pH sampling resulted in a condition where there was reasonable doubt of the operability of the standby liquid control system. The inspectors completed a Phase 1 significance determination on this issue using IMC 0609, "Significance Determination Process," Attachment 4, Table 4a, dated January 10, 2008. The inspectors determined that this finding only represented a degradation of a radiological barrier function and therefore screened as Green. This finding was related to the cross cutting issue of problem identification and resolution (corrective action program) because the licensee did not take appropriate corrective actions to address safety issues in a timely manner.
Inspection Report# : 2009003 *(pdf)*

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Declare Primary Containment Isolation Valve Inoperable and Take Required Actions
A self-revealed NCV of Dresden Station Improved Technical Specification (TS) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," of very low safety significance was identified for the failure to declare primary containment isolation valve 3 3702 inoperable and take actions in accordance with the requirements of TS 3.6.1.3 required action A. The licensee generated IR 837675 and IR 839009 to address this issue. Corrective actions included: the initiation of a training request to re enforce with Operations personnel the potential operability issues when light indications are not functioning properly, and the revision of Operations procedures to include guidance to alert users that a failed or flickering indication light associated with a motor operated valve may indicate problems that could affect valve operation, and that valve operability must be verified.

The finding was more than minor because it impacted the Barrier Integrity objective to provide reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," The inspectors answered NO to all questions in the Containment Barrier column of Table 4a, therefore the finding screened as Green (i.e., very low safety significance). The inspectors determined that this finding also affected the cross cutting area of Human Performance, resources aspect (H.2(c)) because the licensee failed to provide complete, accurate and up to date procedures.

Inspection Report# : 2009002 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

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 : December 10, 2009