Dresden 3
1Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Electro Hydraulic Control (EHC) Fluid Leaking From Stop Valve 3-5699-MSV4-FA Resulting in Forced Outage D3F49
The failure of the Unit 3 Main Turbine Stop Valve (MSV) # 4 fast acting solenoid valve on November 6, 2009, resulted in a self revealed finding of very low safety significance. The licensee failed to use the correct o rings and bolts when replacing the Unit 3 MSV #4 fast acting solenoid valve during the Unit 3 refueling outage in 2008 which led to the failure. The equipment was not safety related. Therefore, this finding did not result in a violation of regulatory requirements. The licensee's corrective actions included revising maintenance procedure DEP 5600 01, "Main Turbine Valve Solenoid and Servo Maintenance," to incorporate the actions described in GE Technical Information Letter 1594. The bolts on the U3 and U2 solenoid valves were replaced. The licensee did not determine that the o rings were defective until after both this Unit 3 forced outage and the Unit 2 November 2009 refueling outage were complete. Therefore, one corrective action was to write a work order to change the o rings on the solenoids for both units. In addition, corrective actions were put in place to address weaknesses in the evaluation of Operating Experience. The licensee addressed this issue in the corrective action program under Issue Reports 899829 and 989733.
The inspectors determined that the use of o rings, GE part number U472X000B906, in U3 turbine control valve solenoids, was contrary to Vendor Technical Information Program Binder D1180, General Electric Steam Turbine Generator (GEK5551), Tab 8, GE drawing 115D2402 (Revision 12), and GE Technical Information Letter (TIL) 1594, dated November 30, 2007, which required the use of o rings, GE part number U472X000BS906, and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of procedure quality and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability. 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 Events Cornerstone. The electro hydraulic control leakage caused by one or more failed o rings could have resulted in a turbine trip and reactor scram. However, the failure would not affect mitigating equipment or functions so the finding screened as having very low safety significance. This finding had a cross cutting aspect in the area of Problem Identification and Resolution, Operating Experience because the licensee did not implement and institutionalize Operating Experience through changes to station processes, procedures, equipment, and training programs.
Inspection Report# : 2010002 *(pdf)*

Significance: G Dec 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Procedural Deficiency Causing a Pressure Pulse Resulting in a Reactor Water Level Low-Low Group 1 Isolcation Signal and Unit 3 Reactor Scram
A self-revealed finding involving a non-cited violation (NCV) of Technical Specification 5.4.1 was identified on October 3, 2009, due to the licensee's failure to include essential information in DOP 1200-03, "RWCU System Operation with the Reactor at Pressure," Revision 51, regarding startup of the reactor water cleanup system with the reactor at pressure. This procedural deficiency caused a pressure pulse that resulted in a reactor water level Low-Low Group 1 Isolation Signal and Unit 3 reactor scram. This event was entered into the licensee's corrective action program (CAP) as Issue Report (IR) 974426. Corrective actions by the licensee included revising procedure DOP 1200-03.
This finding was considered more than minor because it affected 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 at power operations. The finding was determined to be of very low safety significance because it did not contribute to both the likelihood of a reactor trip AND the likelihood that mitigating equipment or functions will not be available. This finding has a cross-cutting aspect in the area of Human Performance (Resources) because the licensee did not provide complete, accurate and up-to-date procedures to plant personnel. H.2(c) (Section 4OA3.2)
Inspection Report# : 2009005 *(pdf)*

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)*

Mitigating Systems

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Significance of Potentially Submerged Safety and Non safety-related Low Voltage Power and Control Power Cables
The inspectors identified a finding of very low safety significance with an associated Non Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, licensee personnel failed to maintain safety related cables in underground manholes from becoming repeatedly submerged, which resulted in subjecting the cables to an environment for which they were not qualified. As corrective action, the licensee generated work order (WO) 01271108 on September 24, 2009, to remove the seals on the conduit which contained the cables and which kept water from draining out of the conduit. This issue was entered into the licensee’s corrective action program as Issue Report (IR) 975308.
The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because it was a qualification deficiency that did not result in a loss of operability. The inspectors concluded that there was not a cross cutting issue associated with this violation.
Inspection Report# : 2010002 *(pdf)*

Significance: G Dec 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Ensure a Safety-related Plus was Ordered and installed in the 2/3 Emergency Diesel Generator Turbo Lube Oil "Y" Strainer
A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion IV, “Procurement Document Control,” was self-revealed for the licensee's failure to ensure a safety-related plug was ordered and installed where required in the 2/3 EDG turbo lube oil “Y” strainer. Instead, a non-conforming part was installed, which resulted in a one-half gallon per minute oil leak and removal of the diesel generator from service. The issue was entered into the licensee’s CAP as IR 926605. Corrective actions included inspection of all other diesel generators to ensure the non-conforming condition did not exist on another machine, revising the procurement documents to ensure that future parts include a pressure retaining pipe plug with approved material, and adding a requirement for a quality inspection to be performed to “inspect the strainer for metallic pipe plug in blow down port.” Individual procedure compliance issues were addressed through the station’s performance improvement initiatives.
The finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 5 c because an incorrect and inadequate part was installed and the system was returned to service. This performance deficiency impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A Phase 3 SDP risk evaluation was performed by the regional Senior Risk Analyst who determined the risk significance of the finding to be less than 1.0E-6/yr delta core damage frequency (CDF) and less than 1.0E-7/yr delta LERF, which represents a finding of very low safety significance. Failure of plant personnel to question the plastic shipping plug before the equipment was installed and returned to service was not in compliance with MA-AA-716 008, “Foreign Material Exclusion Program,” and, therefore, inspectors determined that this event was cross-cutting in Human Performance, Work Practices, Procedural Compliance for failure of personnel to follow the procedure. H.4(b) (Section 4OA3.3)
Inspection Report# : 2009005 *(pdf)*

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)*

Barrier Integrity

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Technical Specification 5.5.4 Implementing Procedure
• The inspectors identified a finding of very low safety significance and associated Non Cited Violation of Technical Specification 5.5.4 for the licensee failing to follow Step I.2.a and b of Procedure DOS 1500 08, "Discharge of Containment Cooling Service Water (CCSW) From Low Pressure Coolant Injection (LPCI) Heat Exchanger (Hx) During CCSW Pump Operations," Revision 16. Specifically, the licensee failed to perform a tube leak test as required by DOS 1500 08 when activity exceeded 1.5E 6 microcuries/milliliter. The licensee's corrective actions included a change to DOS 1500 08 to ensure personnel do not waive performance of the test procedure until tube leak checks are considered during non routine samples of CCSW and revising the chemistry sampling procedure CY DR 110 220, "LPCI Service Water (CCSW) and Torus Water Sampling," to notify operations to evaluate performance of a tube leak check if activity exceeds 1.5E 6 microcuries/milliliter.
The inspectors determined that the failure to perform a tube leak test or perform Calculated CCSW Sample Activity Limit and Canal Activity Calculations was contrary to DOS 1500 08, and was a performance deficiency. The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, had there been an actual LPCI Hx tube leak radioactivity could have been released. 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 Containment Barrier Cornerstone. All four questions on this table were answered "no." There was no actual degradation of the containment barrier. Therefore, the issue screened as having very low safety significance. This finding had a cross cutting aspect in the area of Human Performance, Decision Making because the licensee did not demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that it was unsafe in order to disapprove the action. Specifically, the licensee assumed the activity in the sample was coming from the floor drain system with no valid proof that was the case.
Inspection Report# : 2010002 *(pdf)*

Significance: G Dec 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Mispositioning of a Unit 3 Control Rod at Power
A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed for the mispositioning of a Unit 3 control rod at power. Control rod G-11 was withdrawn one notch contrary to TS SR 3.1.3.3 requirements to insert each withdrawn control rod at least one notch. This was a performance deficiency. The violation was entered into the licensee's CAP as IR 993634. Corrective actions included inserting control rod G-11 one notch back to the original position and suspending control rod movement while all rods were verified to be in their correct position. The operator was removed from shift duties and the oncoming shift was briefed of the event.
The finding was determined to be more than minor because the finding was associated with the Barrier Integrity Cornerstone attributes of human performance and configuration control of a control rod, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases

caused by accidents or events. Specifically, the operator withdrew a control rod contrary to expected operation. This added positive reactivity and caused an unanticipated power increase. The inspectors evaluated the finding 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 Fuel Barrier Cornerstone. Per Table 4a, any issue that involves the fuel barrier is screened as Green. This finding had no cross-cutting aspect. (Section 1R22)
Inspection Report# : 2009005 *(pdf)*

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)*

Emergency Preparedness

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: FIN Finding
Failure to Meet Regulatory Commitment to Maintain Contingency Plans for Post-Accident Sampling
The inspectors identified a finding of very low safety significance for the failure to meet a regulatory commitment to maintain a contingency plan for obtaining highly radioactive samples of reactor coolant, the suppression pool, and drywell atmosphere for post accident plant recovery planning. Specifically, the licensee's contingency plan was not adequately maintained to ensure the High Radiation Sampling System (HRSS) functioned adequately or otherwise was demonstrated to be in a state of readiness to allow samples to be obtained within a two week window. No violations of regulatory requirements were identified related to this finding. Corrective actions were being developed to ensure the licensee's contingency plan commitments would be met. Those actions included a means to improve system ownership and establishment of an effective process for HRSS equipment maintenance and repair at a priority consistent with its intended use.
The finding was more than minor because it impacted the facilities and equipment attribute of the Emergency Preparedness Cornerstone and adversely affected the cornerstone objective of ensuring capability to implement adequate measures to protect health and safety of the public in the event of a radiological emergency. Specifically, equipment intended to obtain highly radioactive samples that are used to assess reactor core condition as part of post accident recovery activities was not demonstrated to be in a readiness condition consistent with the licensee's contingency plan. The finding was determined to be of very low safety significance because it involved equipment, which supplements the licensee's emergency plan for reentry and recovery activities as provided in the planning standard of 10 CFR 50.47(b)(8), and represented a planning standard problem associated with demonstrating functional readiness of that equipment. The finding was determined to be associated with a cross cutting aspect in the area of human performance in the resources component, in that, the licensee failed to ensure that equipment to support its emergency plan was functional or otherwise was demonstrated to meet a defined status of operational readiness.
Inspection Report# : 2010002 *(pdf)*

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

Significance: SL-IV Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Record the Identity of Personnel Performing Post Maintenance Tests

A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVII, “Quality Assurance Records,” was identified by the inspectors for the licensee’s failure to record the identity of various personnel who performed seven post-maintenance tests (PMTs) related to Unit 3 EDG maintenance. Despite the PMTs being related to work on safety related components, an activity affecting quality, neither the licensee’s procedure MA AA 716 012, “Post-Maintenance Testing,” nor DAP 15 10, “Post-Maintenance Testing Program,” required the identity of the inspector or tester to be recorded. Completed corrective actions included adding PMT documentation requirements to DAP 15 10 and briefing individuals who perform PMTs.
This finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E examples 1b since a portion of required records were irretrievably lost, and 2h since multiple examples were identified as failures to properly implement the same regulatory requirement. Following IMC 0612, Appendix B, it was apparent that this issue did not fall directly under a cornerstone and that incomplete information was recorded in the seven PMTs. Therefore, the Enforcement Policy was used to screen the severity in conjunction with the IMC 0612, Appendix E, Examples 1b and 2h. Since MA AA 716 012, “Post-Maintenance Testing,” did not properly implement regulatory requirements, this finding has a cross-cutting aspect in the area of Human Performance, Resources because the licensee did not provide complete, accurate, and up-to-date procedures to plant personnel.
Inspection Report# : 2010002 *(pdf)*

Last modified : May 26, 2010