Perry 1
1Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
FAILURE TO ADHERE TO MAINTENANCE INSTRUCTIONS RESULTED IN LOSS OF RECIRCULATION PUMP 'A'
A finding of very low significance was self-revealed on October 15, 2009, when one of two reactor recirculation pumps failed to transfer to slow speed while operators were attempting to downshift both pumps. The finding involved the licensee's failure to adhere to maintenance instructions when personnel incorrectly assembled a relay contactor during maintenance activities on an 'A' recirculation pump low frequency motor generator relay panel. The improperly assembled contactor led to the failure of the 2A breaker to close and re-energize recirculation pump 'A' in slow speed, which caused the loss of the pump and a subsequent unplanned drop in power. No violation of regulatory requirements occurred, and the issue was entered into the licensee's corrective action program.

The failure to adhere to the maintenance instructions resulted in the loss of recirculation pump 'A,' which caused an actual upset in plant stability, and directly affected the objective for the Initiating Events cornerstone. The finding was more than minor because the reactor recirculation pump failure to downshift affected the equipment performance attributes of availability and reliability of the Initiating Events Cornerstone of Reactor Safety. The issue was of very low safety significance because the finding did not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and did not affect other mitigation systems; the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available; and the finding did not increase the likelihood of a fire or internal/external flood. The primary cause of this finding was related to the cross-cutting area of human performance, per IMC 0305 H.4.a., work practices, human error prevention techniques, because the licensee did not ensure that appropriate human error prevention techniques were used.

Inspection Report# : 2009005 *(pdf)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
UNEXPECTED HALF SCRAM DUE TO FAULTY TROUBLESHOOTING PLAN
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 self-revealed for the licensee's failure to have an appropriate troubleshooting plan for repairing Average Power Range Monitor (APRM) 'A.' Specifically, the troubleshooting plan for inoperable APRM 'A' did not provide proper guidance to the technicians resulting in an unexpected half scram on the reactor protection system and subsequent required operator actions. The licensee entered the error into their corrective action program as CR 09-63991. As part of its corrective actions, the licensee planned to place placards in the APRM cabinets warning of the special instructions to remove and replace the cards.

The finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 4.b, and resulted in operator intervention to change reactor power to maintain reactor power at a stable value. Therefore, the performance deficiency impacted the Initiating Events cornerstone objective to limit 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. While the finding increased the likelihood of a reactor trip, it did not increase the likelihood that mitigation equipment would not be available, and therefore, the inspectors determined the finding to be of very low safety significance. The finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a), because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site condition, or the need for planned contingencies, compensatory actions and abort criteria. Specifically, licensee personnel did not adequately research the impact of a circuit card's removal and reinsertion into the control circuitry for APRM 'A,' on other related systems contributing directly to an unplanned power transient on the reactor.

Inspection Report# : 2009004 *(pdf)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
MOISTURE SEPARATOR REHEATER LEVEL SWITCH MAINTENACE CAUSED UNIT TRIP
A finding of very low safety significance was self-revealed on June 21, 2009, for the failure to adequately implement the requirements of Nuclear Operating Procedure (NOP)-WM-4300, Order Execute Process. Specifically, a supervisor authorized work order steps to be performed out of sequence on level switches for the moisture separator reheaters (MSR). The failure to perform steps in order led to some steps being missed and ultimately to a main turbine trip and associated reactor scram. The licensee entered this item into their corrective action program as CR 09-60855. The licensee's immediate actions included response to the reactor scram and formation of a troubleshooting team to conduct a root cause investigation of the failure of the MSR level indicators.

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. Specifically, inadequate adjustment and calibration of the level switches following replacement resulted in a main turbine trip and reactor scram from full power. 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. While the finding resulted in a reactor trip, it did not contribute to the likelihood that mitigation equipment would not be available, therefore, the inspectors determined the finding to be of very low safety significance. This finding has a cross cutting aspect in the area human performance, resources per IMC 0305 H.2(c), because the licensee did not ensure that procedures were adequate to assure nuclear safety. Specifically, the generic instrumentation and control instruction and the work order for conducting maintenance on the moisture separator reheater level switches did not contain critical vendor information or guidance to reflect the significance of taking as found data to support calibration of the replacement switches.

Inspection Report# : 2009004 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inability to Operate RHR common suction Line Valve
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 failure to implement the requirements of licensee (normal operating procedure) NOP OP-1014, "Plant Status Control," Revision 00. Specifically, operations personnel used a mechanical advantage device to operate valve 1E12F0010 without evaluating the affect on the valve. Damage to residual heat removal (RHR) valve 1E12F0010 prevented the plant from entering shutdown cooling. It was determined that the valve operator stem sheared due to excessive torque used to operate the valve. As part of their immediate corrective actions, licensee personnel repaired the valve stem operator to restore shutdown cooling and entered the issue into their corrective action program.
The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the failure of the RHR shutdown cooling common suction isolation valve caused both trains of shutdown cooling to be unavailable during shutdown operations. Using IMC 0609, Appendix G, "Shutdown Operation Significance Determination Process," Checklist 7, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative

assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee did not ensure adequate supervisory and management oversight of work activities to ensure nuclear safety. Specifically, supervisors were aware of the use of mechanical advantage devices on the RHR shutdown cooling common suction manual isolation valve and did not ensure an appropriate evaluation was conducted.

Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Near Loss of Shutdown Cooling due to Maintenance Activity
. 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 when technicians performed maintenance on protected equipment without implementing risk management requirements specified in station procedures. This resulted in a loss of shutdown cooling flow to the reactor coolant system. Specifically, the licensee established procedure NOP-OP-1005, “Shutdown Defense in Depth,” Revision 10 as the implementing procedure to manage risk during shutdown conditions. The licensee failed to implement the significant risk management actions prescribed in procedure NOP-OP-1005 for maintenance on protected equipment. This resulted in a blown fuse in the reactor protection system causing a loss of shutdown cooling flow to the reactor coolant system. The licensee replaced the fuse and restored shutdown cooling. This issue was entered into the corrective action program as CR 09-58110.
The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the finding resulted in a loss of reactor decay heat removal event while the reactor was shutdown. Using IMC 0609, Appendix G, “Shutdown Operation Significance Determination Process,” Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a), because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site conditions, or the need for planned contingencies, compensatory actions, and abort criteria.

Inspection Report# : 2010002 *(pdf)*
Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Work on Wrong Relay Potential Impact on Shutdown Cooling
A finding of very low safety significance was self-revealed on April 28, 2009, for the failure to follow maintenance procedure PTI-N41-P0002, “Generator Switchgear Protective Relay Trip Test,” when electricians performed maintenance on an incorrect relay associated with bus L11. The licensee posted bus L11 as a protected train and repaired the 1R22-Q103A and 86B circuitry. The licensee entered the issue into their corrective action program as CR-09-58187.

The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, had the 1R22-Q103A relay circuitry functioned as designed, a loss of decay heat removal event would have occurred. Using IMC 0609, Appendix G, “Shutdown Operation Significance Determination Process,” Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(a), because the licensee did not use error prevention techniques commensurate with the risk of the maintenance activity.

Inspection Report# : 2009003 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO CORRECTLY ASSESS RISK DURING POST-MAINTENANCE ACTIVITIES
. A finding of very low safety significance and associated NCV of 10 CFR 50.65(a)(4) was identified by the inspectors for the licensee's failure to accurately assess plant risk during maintenance activities. The inspectors determined that the licensee failed to correctly identify the plant risk condition when the Unit 1 Division 1 Emergency Diesel Generator (EDG) was out of service for maintenance. Specifically, there was a 5 hour period of time that the licensee restored plant risk to GREEN status while the EDG remained unavailable and plant risk was actually YELLOW. The licensee entered the issue associated with their failure to correctly assess the plant risk condition into their corrective action program (CAP).

The performance deficiency was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 7.e, and resulted in actual plant risk being in a higher licensee-established risk category than declared. The finding was of very low safety significance because the risk deficit, or incremental core damage probability deficit (ICDPD) was < 1E-6. This finding had a cross-cutting aspect in the area of Human Performance, Decision-Making per IMC 0310 (H.1(b)) because the licensee did not use conservative assumptions in decision making nor adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, the licensee chose to minimize system unavailability time and as a result did not perform a complete post-maintenance test which would have verified the EDG system was fully functional and available to perform its mission at the end of the maintenance period.

Inspection Report# : 2010002 *(pdf)*

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: FIN Finding
FAILURE TO MAKE AN ACCURATE IMMEDIATE OPERABILITY DETERMINATION
A finding of very low safety significance was identified by the inspectors for the licensee’s failure to make an accurate immediate operability determination (IOD) based on the actual plant conditions and the available information to provide reasonable assurance of operability. Specifically, on February 15, 2010, through wall leakage was identified coming from a welded elbow connection of an instrument line associated with the 'B' Emergency Closed Cooling (ECC) system supply to the 'B' control complex chiller heat exchanger. This instrument line is an American Society of Mechanical Engineers (ASME) Section III, Class 3 piping system, and the licensee’s IOD declared the 'B' ECC system operable without the degradation mechanism being discernable from visual examination (such as external corrosion or wear) or having substantial operating experience (site specific) with the identified degradation mechanism in the affected system. No violation of regulatory requirements occurred, and the issue was entered into the licensee’s CAP.

The performance deficiency was determined to be more than minor because it is associated with the Mitigating Systems cornerstone attribute of “Equipment Performance-Availability, Reliability,” and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance because a loss of system safety function, or the actual loss of safety function of a single train for greater than its TS-allowed outage time did not occur, and the finding does not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding had a safety culture cross-cutting aspect in the area of Problem Identification and Resolution, related to the Operating Experience component for not implementing and institutionalizing operating experience through changes to station processes, procedures, equipment, and training programs per IMC 0310 (P.2 (b)). Specifically, the requirement for the degradation mechanism of through wall leakage on ASME Section III, Class 2 and 3 piping, to be readily apparent from visual examination in order to support an operable IOD, was not completely understood by operations personnel. This finding did not involve a violation of regulatory requirements.

Inspection Report# : 2010002 *(pdf)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO PERFORM AN ADEUQATE POST-MAINTENANCE TEST FOLLOWING INSTALLATION OF NEW EMERGENCY DIESEL GENERATOR CARBON DIOXIDE SYSTEM CONTROL PANELS
A finding of very low safety significance (Green) and associated non-cited violation of license condition 2.C.(6), Fire Protection, was self-revealed for the licensee's failure implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR). Specifically, the licensee failed to ensure that, "…the main floor [of the Diesel Generator Building] is protected by a total flooding carbon dioxide system for fire suppression." The licensee had installed a permanent modification to the carbon dioxide system for the diesel generator room, but had chosen not to conduct complete post modification testing. The failure to conduct a complete test resulted in a wiring error to go undetected. Testing after the system was placed in service identified that the system did not function as designed. Troubleshooting identified that Division 2 and 3 emergency diesel generators (EDG) pneumatic electric relays in the new control panel were cross-wired to Division 3 and 2 EDG fan relays, respectively, in the control box. As part of their corrective actions, the licensee re-labeled the wires correctly in the CO2 panels and landed them on their appropriate terminals. The licensee entered this issue into their corrective action program as CR 09 60866. The licensee's immediate corrective actions included placing the system in lockout and notifying all fire team personnel of the manual actions required to initiate CO2 flow into the emergency diesel generator rooms.

The finding was determined to be more than minor because, if left uncorrected, the inability of the EDG automatic fire suppression system to perform its function would become a more significant safety concern. Specifically, a fire in the Division 2 EDG room would not have been protected by adequate automatic fire suppression and it would render the Division 3 EDG inoperable. Similarly, a fire in the Division 3 EDG room would not have been protected by adequate automatic fire suppression and it would render the Division 2 EDG inoperable. The inspectors concluded this finding was associated with the Mitigating Systems cornerstone. In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 Initial Screening and Characterization of Findings," Table 3b, the inspectors determined that the finding degraded the fire protection defense in depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," was required. Using Part 1 of the Fire Protection SDP Phase 1 Worksheet in Manual Chapter 0609, Significance Determination Process, the performance issue was determined to be in the fixed fire protection systems category based on the fixed fire suppression systems being degraded. This finding did not screen as very low safety significance (Green) in the Phase 1 analysis and a Phase 2 analysis under IMC 0609 Appendix F was required.

A regional senior reactor analyst evaluated this finding and assumed the fire frequency to be 3.0E-2 for the EDG rooms based on the licensee's IPEEE (Individual Plant Examination – External Events). Considering the fire frequency and remaining mitigating capability in the event of a plant transient, the senior reactor analyst determined that the risk associated with this finding was less than 1.0E-06. Therefore, this finding was determined to be best characterized as very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance, decision making, per IMC 0609 H.1(b), because the licensee's decisions did not demonstrate that nuclear safety was an overriding priority. Specifically, the licensee chose to minimize system unavailability time over performing a full and complete post-maintenance test on a newly installed EDG CO2 control system, a test that would have identified the wiring error.

Inspection Report# : 2009004 *(pdf)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
MMAINTENANCE ERRORS CAUSE LOSS OF DIVISION 1 ECCS ELECTRICAL POWER
A finding of very low safety significance (Green) and associated non-cited violation of Technical Specification 5.4.1 was self-revealed when the licensee failed to follow Nuclear Operating Procedure (NOP)-WM-3001; Work Management PM Processes. Specifically, step 4.5.5 of NOP-WM-3001 states, “If a General Nuclear Preventative Maintenance (GNPM) Order cannot be completed as planned due to ... replacement of a failed or degraded component, then the MWC Supervisor shall take appropriate actions in accordance with the flow diagram in Attachment 6 ...” During the performance of work order 200297036, for safety related 480-V breaker EF1A03, the supervisor directed a 4-point switch be replaced as part of the work order; however, no evaluation of the change in scope was completed and a CR was not written as required by Attachment 6 of NOP-WM-3001. The failure to evaluate the replacement lead to the loss of power to a number of safety related components. The licensee entered this item into their corrective action program as CR 09-63681. The licensee’s immediate action included entry into the appropriate technical specifications, restoration of the lost electrical power bus and restoration of emergency core cooling systems which were made inoperable as a result of the power loss.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of human performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the inadequate work planning caused a loss of electrical power to bus EF-1-A, the safety-related 480 V power supply to Division 1 components placing the plant in an orange probabilistic safety analysis risk condition. 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 2, the inspectors determined that core decay heat removal was degraded. Using Table 4a, “Characterization Worksheet for IE, MS, and BI Cornerstones,” the inspectors assessed the finding as having very low safety significance (Green) because no loss of safety system function occurred and no loss of function of a single train occurred for greater than its TS-allowed outage time. This finding has a cross-cutting aspect in the area of human performance, work control per IMC 0609 H.3(b) because the licensee did not plan and coordinate work activities consistent with nuclear safety. Specifically, licensee personnel failed to plan and coordinate the replacement of an auxiliary switch in breaker EF1A03 thereby not incorporating the impact of the changes to the work scope or activity on the plant and human performance.

Inspection Report# : 2009004 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Prevent Contact of Energized Components Renders RCIC System Inoperable
A finding of very low safety significance (Green) and associated NCV of Technical Specification 5.4.1 was self-revealed when technicians failed to implement actions to prevent contact of energized electrical components during maintenance. Specifically, the reactor core isolation cooling (RCIC) Division 2 logic tripped while attempting to lift leads and a test lug. The technicians suspended their surveillance procedure and operators restored the RCIC system in accordance with licensee procedures. Operators also verified high pressure core spray (HPCS) was operable. The licensee visually inspected the RCIC system and found no apparent damage. The licensee conducted additional training on the use of error prevention tools and entered the issue into the corrective action program as CR 09-59356. 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 to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the short circuit resulted in the RCIC system being inoperable. The finding was determined to have very low safety significance because it did not represent a loss of system safety function, a loss of safety function of a non-TS train designated as risk significant for greater than 24 hours, an actual loss of safety function of a single train for greater than its TS- allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance per IMC 0305 H.4 (a) because the technician failed to use error prevention techniques, such as self-checking, that are commensurate with

the risk of the assigned task. Specifically not using 'STAR' (Stop, Think, Act, Review) during an activity that could render the RCIC system inoperable.

Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
RHR System Over-Pressurization Due to Failure to Implement Corrective Actions for a Condition Adverse to Quality
A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was self-revealed for the failure to implement corrective actions to ensure residual heat removal (RHR) check valve 1E12F0050A seated during plant pressurizations. Specifically, the licensee failed to establish and maintain corrective actions for check valve 1E12F0050A inability to seat under low differential pressure conditions, resulting in the over-pressurization of a section of RHR system piping. As part of the licensee's corrective action, the operators depressurized the RHR system below operating pressure and were revising procedures to ensure the check valve 1E12F0050A seats fully during system pressurization. This issue was entered into the licensee corrective action program by CR 09-58808 and CR 09-58995 and an appropriate permanent corrective action was being evaluated.
The inspectors determined that the finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, removal of the RHR venting evolution from station procedures resulted in an unexpected over-pressurization which could have resulted in system damage. Using IMC 0609, Appendix G, “Shutdown Operations Significant Determination Process,” Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of problem identification and resolution per IMC 0305 P.1(c), because the organization failed to thoroughly evaluate the impact of modifying a corrective action. Specifically, the licensee failed to thoroughly evaluate the consequences of removing the venting section of a procedure that was a corrective action for the check valve’s inability to seat under low differential pressure conditions.
Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 26, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Procedures for Post Fire Operation of Control Room HVAC Fans and Control of Remote Shutdown Room Toolbox Inventory.
The inspectors identified a finding of very low safety significance and associated NCV of Technical Specification (TS) 5.4.1a for failure to maintain procedures for post-fire operation of control room heating, ventilation, and air conditioning (HVAC) Train A fans and for control of the Division 1 remote shutdown room toolbox inventory. Specifically, Procedure ARI H13 P904 0001 B6, "Control Room HVAC Train A Tripped," stated that if a fire has occurred and the Train A fans have tripped, then restart the Train A fans in emergency recirculation mode. The correct action was to restore Train B fans in emergency recirculation mode. In addition, Procedure IOI 11 "Control Room Isolation," Attachment 20, contained a list of equipment operators were to obtain from the toolbox, located by the alternate remote shutdown panel, following a control room fire. The list included nine items; one of the items consisting of three FRS R 4 Amp fuses was missing. The procedure should have been revised to remove the requirement to obtain the fuses. The licensee entered this finding into their corrective action program as CR 09 60317 and CR 09 60373.
The finding was determined to be more than minor because the finding was associated with the mitigating system cornerstone attribute of procedure quality and affected the cornerstone’s objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to maintain the procedures could have complicated plant safe shutdown in the event of a fire. The inspectors determined that the finding was of very low safety-significance since the procedure deficiencies did not substantially impact performance in the event of a fire.

Inspection Report# : 2009007 *(pdf)*

Significance: Jun 26, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Assure That Systems Structures and Components Necessary to Achieve And Maintain Hot Shutdown Conditions Were Free of Fire Damage Without Repair Actions.
The inspectors identified a finding of very low safety significance and associated NCV of the Perry Nuclear Power Plant, Unit 1, Facility Operating License Condition 2.C.(6) and 10 CFR Part 50, Appendix R, Section III.G.2, involving the licensee's failure to ensure, in the event of a fire, that one redundant train of systems necessary to achieve and maintain hot shutdown conditions was free of fire damage. Specifically, the licensee failed to ensure, in the event of a control room fire, certain Systems, Structures,
and Components (SSCs) necessary to achieve and maintain hot shutdown conditions (e.g., MOVs 1P45 F0014A and 1P45 F0068A and emergency service water system (ESW) equipment) were free of fire damage. The licensee entered this finding into their corrective action program for resolution as CR 09 60977. The control room area, which is susceptible to this condition, is continuously manned; therefore making a compensatory action of a fire watch un-necessary.

The finding was determined to be more than minor because the finding was associated with the mitigating systems cornerstone attribute of protection against external factors (Fire) and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The Phase 2 screening determined this finding was of very low safety-significance because no potentially challenging fire scenarios were developed.
Inspection Report# : 2009007 *(pdf)*

Significance: Jun 26, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Provide Required Electrical Isolation for Post Fire Safe Shutdown Electrical Circuits
The inspectors identified a finding of very low safety significance and associated NCV of the Perry Nuclear Power Plant, Unit 1, Facility Operating License Condition 2.C.(6) and 10 CFR Part 50, Appendix R, Section III.G.3, for failure to provide the required electrical isolation in the design of the post-fire safe shutdown control logic circuitry. Specifically, the control logic for the Division 1 diesel generator building ventilation fan 1M43 C001A did not have the required physical isolation to isolate control room fire-induced electrical faults when transferring control to the remote shutdown station. This is required to ensure that postulated fire-induced electrical faults would not result in the loss of post-fire alternative safe shutdown equipment. The licensee's immediate corrective action was to perform a preliminary evaluation of the electrical circuitry and cables. The licensee entered this finding into their corrective action program as CR 09 60873. The control room area, which is susceptible to this condition, is continuously manned; therefore making a compensatory action of a fire watch unnecessary.

The finding was determined to be more than minor because the finding was associated with the mitigating systems cornerstone attribute of protection against external factors (Fire) and affected the cornerstone's objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. The violation is associated with degradation of a fire protection feature. In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 Initial Screening and Characterization of Findings," Table 3b, the inspectors determined the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," was required. Using Part 1 of the Fire Protection SDP Phase 1 Worksheet in Manual Chapter 0609, "Significance Determination Process [SDP]," the performance issue was determined to be in the post-fire safe shutdown category. The degradation rating was low based on FirstEnergy Nuclear Operating Company's (FENOC's) engineering evaluation that concluded that there were no fire induced electrical faults resulting from a control room fire that would prevent the plant from achieving and maintaining a safe shutdown in the event of a control room fire. Therefore, the finding screened as Green or of very low safety significance in the Phase 1 Worksheet. This violation is being treated as a NCV consistent with Section VI.A of the Enforcement Policy. The cause of the finding related to the cross-cutting aspect of problem identification and resolution (Section 1R05.6b).

Inspection Report# : 2009007 *(pdf)*

Barrier Integrity

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
RCS Temp Below Minimum Allowed by TS due to Inadequate Station Procedures
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 when operators failed to maintain reactor coolant system (RCS) temperature greater than the Technical Specification minimum allowable temperature because Integrated Operations Instruction (IOI)-9, "Refueling," was inadequate. The licensee did not properly control an outage activity in that they failed to ensure the water sprayed into the reactor pressure vessel met temperature requirements. As part of their immediate corrective actions, licensee personnel stopped the activity and restored the RCS system above the TS 3.4.11 temperature requirements of 70 °F. The licensee entered this issue into the corrective action program as CR 09-55397.

This finding was determined to be more than minor because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, failure to maintain RCS temperature greater than the minimum allowed by TS affected the functionality of this barrier. Using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a) because the organization failed to appropriately plan work and coordinate work activities consistent with nuclear safety. Specifically, job site conditions, including environmental conditions which may impact plant structures, systems, and components; were not considered to ensure water sprayed into the RCS would maintain temperature above 70 °F.

Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Follow Industrial Safety Manual Results in Damage to Fuel Handling building Roll-up Door
A finding of very low safety significance was self-revealed when contract personnel failed to follow the FENOC Industrial Safety Manual to control vehicle movement inside the fuel handling building (FHB). Specifically, a Sea-land truck backed into the FHB roll-up door, dislodging the door in the open position. The licensee suspended fuel movements and implemented compensatory measures for containment integrity. The licensee repaired the roll-up door and conducted training with contract and oversight personnel, and entered the issue into the corrective action program as CR 09 56062.

The finding was determined to be more than minor because the finding was associated with Barrier Integrity cornerstone attribute of SSC and barrier performance 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 finding resulted in an event that challenged FHB integrity, which is a functional barrier to fission product release. Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," was used since IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," does not address the potential risk significance of FHB operations. Regional management determined that the finding was of very low safety significance because there was no fuel handling accident during this period. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported.

Inspection Report# : 2009003 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
UNPOSTED HIGH RADIATION AREA AT THE TIP MACHINES
The inspectors identified a finding of very low safety significance and an associated non-cited violation of Technical Specification 5.7.1 for the failure to conspicuously post a high radiation area on the 599' elevation of the containment building. Corrective actions included instituting high radiation area controls when the traverse in-core probe system is operated. The licensee entered the issue into its corrective action program as Condition Reports 09-59344 and 09-67987.
The finding was more than minor because it impacted the Occupational Radiation Safety Cornerstone objective for ensuring adequate protection of worker health and safety from exposure to radiation in the attribute of program and process for as-low-as-is-reasonably-achievable (ALARA) planning, in that, not conspicuously posting high radiation areas may result in unnecessary and unplanned radiation exposures to workers. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The primary cause of this finding was related to the cross-cutting area of human performance in work practices, per IMC 0305 H.4.a., in that, personnel work practices and human performance error reduction techniques were not used commensurate with the risk of the assigned task.

Inspection Report# : 2009005 *(pdf)*

Significance: G Dec 31, 2009
Identified By: NRC
Item Type: FIN Finding
EXCESS DOSE INCURRED FOR THE ALTERNATE DECAY HEAT REMOVAL PROJECT
The inspectors identified a finding of very low safety significance for inadequate job planning and ineffective work controls which adversely impacted the licensee's ability to minimize dose for the alternate decay heat removal (ADHR) project during Refuel Outage 12. Specifically, controls were not effectively implemented to reduce ambient radiation levels, and minimize in-field work hours for craft personnel. The issue resulted in an actual dose outcome that was not consistent with the planned, intended dose for work associated with modifications to the ADHR. Corrective actions were implemented to address the organization and programmatic deficiencies in managing the installation of major plant modifications.
The finding was more than minor because it impacted the Occupational Radiation Safety Cornerstone objective for ensuring adequate protection of worker health and safety from exposure to radiation in the attribute of program and process for ALARA planning, in that, ineffective ALARA planning and work control deficiencies contributed to an actual increase in worker doses in excess of five person-rem and exceeded the licensee's initial intended dose estimates by more than 50 percent. The finding did not involve: (1) an overexposure; (2) a substantial potential for an overexposure; or (3) an impaired ability to assess dose. While the finding involved ALARA planning and controls, the 3 year rolling average dose for the Perry Plant was less than the SDP threshold of 240-person-rem for boiling water reactors at the time the performance deficiency occurred. Consequently, the inspectors concluded through the SDP

assessment that this is a finding of very low safety-significance. The finding was determined to be associated with a cross-cutting aspect in the area of human performance in work controls, per IMC 0305 H.3.a., in that, the licensee did not appropriately plan work activities by incorporating radiological safety.

Inspection Report# : 2009005 *(pdf)*

Significance: G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Appropriate Water Sheilding Between Irradiated Fuel and in-vessel 360 degree platform
The inspectors identified a finding of very low safety significance and an associated NCV of TS 5.4.1.a for the failure to establish, implement, and maintain adequate written procedures regarding the radiation safety program. The licensee failed to implement procedurally required compensatory measures associated with moving irradiated fuel assemblies. Specifically, workers on the 360 platform were within close proximity of the refueling mast while fuel moves were in progress. As corrective actions, the licensee posted information signs to control access to specific areas of the 360 platform and planned to incorporate more rigorous radiological controls into the governing procedure. The licensee entered the issue into its corrective action program as CR 09-54697.

The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that, not implementing adequate radiological control may potentially result in unplanned exposures to radioactive material. The finding was determined to be of very low safety significance because it was not an as-low-as-is-reasonably-achievable (ALARA) planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The finding was determined to have a cross-cutting aspect in the decision making component of the human performance area in accordance with IMC 0305 H.1(b), because the licensee did not adequately use conservative assumptions in decision making.

Inspection Report# : 2009003 *(pdf)*

Significance: G Jun 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Barricade and Conspicuously Post a High Radiation Area
The inspectors reviewed a self-revealing finding of very low safety significance and an associated NCV of Technical Specification 5.4.7.a for the failure to barricade and conspicuously post a high radiation area on the 599' elevation of the auxiliary building. As corrective actions, the licensee barricaded and conspicuously posted the affected area as a high radiation area and performed gravity flushes of piping with clean water to reduce the ambient dose rates. The licensee entered the issue into its corrective action program as CR 09-55453.
The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that, not barricading and conspicuously posting high radiation areas may result in unnecessary and unplanned radiation exposures to workers. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The finding was determined to have a cross-cutting aspect in the work control component of the human performance area in accordance with IMC 0305 .H.3(b), because the licensee did not adequately coordinate work activities.

Inspection Report# : 2009003 *(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 : July 02, 2010