North Anna 2
4Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2010
Identified By: NRC
Item Type: FIN Finding
Failure to Maintain PM Procedures for Circuit Breakers Current with Industry Information and OE
A Green, self-revealing finding was identified for the failure to maintain a preventative maintenance (PM) procedure for circuit breakers current with industry information and operating experience (OE), as required by procedure, DNAP-2001, "Equipment Reliability Process," Revision 0. The licensee entered this problem into their corrective action program as condition report 331819.

The failure to maintain an adequate preventive maintenance (PM) procedure led to an age related failure of a motor starter (main contactor) causing a fire in safetyrelated breaker cubicle J1 of motor control center (MCC) 1J1-2S which supplied power to the D control rod drive mechanism cooling fan, 01-HV-F-37D. The failure to establish an adequate PM task for testing the main contactor of a circuit breaker to ensure that it is in good operating condition and will operate reliably until the next scheduled maintenance was determined to be a performance deficiency. Significance Determination Process (SDP) phase 1 screening of the finding was performed and the finding was determined to increase the likelihood of a fire external event and required a phase 3 SDP evaluation. A phase 3 SDP analysis was performed by a regional SRA in accordance with Inspection Manual Chapter 0609 Appendix F, NUREG /CR -6850 as amended by NUREG/CR -6850 supplement 1, with the NRC North Anna SPAR risk model used to determine the conditional core damage probability (CCDP) for the fire scenarios. The dominant sequence was a fire in MCC1J1-2S damaging MSIV cables resulting in a reactor trip transient with failure of high pressure recirculation and residual heat removal due to fire effects leading to core damage. The evaluation concluded that the core damage frequency (CDF) increase of the potential fire scenarios was characterized as of very low safety significance (Green). This finding involved the cross-cutting area of problem identification and resolution, the component of OE, and the aspect of implementation and institutionalization of OE through changes to station processes and procedures (P.2(b)), because the licensee failed to incorporate existing industry OE to ensure procedural guidance was adequate for testing of the main contactor.
Inspection Report# : 2010005 *(pdf)*

Significance: G Sep 30, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to conduct adequate review of calculation results in main turbine/reactor trip
A self-revealing finding was identified for the licensee's failure to conduct an adequate review of calculations for the operation of the Unit 2 main generator automatic voltage regulator (AVR), as required by licensee procedure CM-AA-CLC-301, "Engineering Calculations", Rev. 3, which resulted in the actuation of a main generator protective lockout relay and subsequent main turbine/reactor trip. The licensee entered this problem into their corrective action program as condition report 378800.

The inspectors determined that the failure to conduct an adequate owner's review of calculation EE-0826, as required by licensee procedure CM-AA-CLC-301, "Engineering Calculations", Rev. 3, was a performance deficiency (PD). The inspectors reviewed IMC 0612, Appendix E and determined the PD was more than minor, because it was similar to example 4.b in that the procedural error resulted in a reactor trip or other transient. In addition, the inspectors determined that it adversely impacted 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 power operations, specifically the attribute of Design Control in that the AVR design change was not properly controlled and Human Performance in that licensee personnel conducting the owner's review failed to follow the requirements of CM-AA-

CLC-301 and conduct an owner's review of calculation EE-0826. The inspectors reviewed IMC 0609 Attachment 4 and determined that the finding was of very low safety significance, or Green, because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The cause of this finding involved the cross-cutting area of human performance, the component of decision making, and the aspect of conservative assumptions and safe actions, H.1(b), because the licensee failed to use conservative assumptions and demonstrate that the proposed action was safe in making the decision that the incorrect inputs for the five-point curve would not be used by the MEL tuning software. (Section 4OA3.1.1)

Inspection Report# : 2010004 *(pdf)*

Significance: G Sep 30, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate set point for balbance-of-plant bus undervoltage relay
A self-revealing finding was identified for the failure to establish an adequate set point for a balance-of-plant 4160 V bus undervoltage protection relay. The inadequate set point caused a reactor trip upon automatic start of a steam generator feedwater pump. The event was reported to the NRC in Licensee Event Report (LER) 0500339/2010-002-00. Corrective action has been taken to reduce the probability of recurrence of the problem. The licensee has placed this issue in their corrective action program as Root Cause Evaluation (RCE) 001012.

The fact that the motor starting voltage dip of the twin 4500 horsepower motor feedwater pump was below the set point of the bus undervoltage protection relays was a performance deficiency. The typical industry standard practice for bus undervoltage is that the set point be below the motor starting voltage dip to preclude spurious actuation of the undervoltage relays for expected voltage transients such as motor stating. This industry standard practice is documented in Institute of Electrical and Electronics Engineers Standard 666-1991, "IEEE Design Guide for Electric Power Service Systems for Generating Stations." Table 7.2, "Motor Protection Devices," states that the suggested setting for undervoltage relay is that it be set to override voltage drop due to motor starting. The potential for spurious tripping of the undervoltage relays has nuclear safety ramifications, in that it can contribute to a reactor trip, as it did on May 28, 2010. The performance deficiency is more than minor because it was associated with the attribute of design control and adversely affected the objective of the initiating event cornerstone. The inappropriate undervoltage relay set point contributed to a reactor trip which is an event that upset plant stability and challenged critical safety functions. The finding was evaluated for significance using Inspection Manual Chapter 0609, Appendix E. The finding was determined to be very low safety significance, Green, because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation functions will not be available. The cause of the finding was evaluated in the licensee's corrective action program as RCE001012. According to the LER and RCE001012, the cause of the finding was determined to be lack of a design basis for the undervoltage protection relay. Since the set point was established well outside the two-year window of current performance and there was no prior event that provided an opportunity to identify this problem, this issue did not represent current licensee performance. Therefore, no associated cross-cutting aspect was identified. (Section 4OA3.3)
Inspection Report# : 2010004 *(pdf)*

Significance: G Sep 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to follow procedure to trip "A" reactor coolant pump on high bearing temperature
A non-cited violation of Technical Specifications 5.4.1a was identified by the inspectors for the failure to adequately implement procedural requirements which resulted in operation of the 'A' reactor coolant system (RCS) pump (RCP) beyond the motor high bearing temperature limit of 195 degF for approximately 10 minutes. The licensee entered this problem into their corrective action program as corrective action 170278 associated with condition report 382725.

The inspectors determined that the failure to implement an alarm response procedure to trip the 'A' RCP in a timely manner was a performance deficiency (PD). The PD was more than minor, because it could be reasonably viewed as a precursor to a significant event due to RCP motor operation in an unknown condition of bearing performance in which the melting of Babbitt material can lead to excessive shaft vibrations and consequent adverse impact on RCP seal performance leading to a seal loss of coolant accident. Significance determination process (SDP) phase 1 screening

determined the finding to be a primary system loss of coolant accident initiator contributor as RCP operation without motor bearing cooling could lead to motor bearing failure, RCP vibration and potential vibration induced RCP seal damage. The finding was determined to fit under the Initiating Events cornerstone in that assuming worst case degradation the potential seal leakage could exceed the technical specification limit for RCS leakage and required phase 2 analysis. Since the North Anna SDP pre-solved worksheet did not specifically address loss of cooling to the RCP motor bearings, a phase 3 analysis was performed by a regional SRA using the NRC's North Anna SPAR model. The sequence was a reactor trip transient caused by a lightning strike in the switchyard, loss of the 1H emergency bus, RCP motor bearing damage due to loss of bearing cooling, failure to trip the RCP, RCP seal failure, failure of high pressure injection, successful depressurization and failure of low pressure injection leading to core damage. A diagnosis and action human error probability for RCP trip was developed for the event conditions. The risk of the event was mitigated by the availability of seal cooling, seal injection and the time and cues available to the operator to trip the RCP prior to vibration induced seal failure. The phase 3 risk evaluation determined that the risk increase of the finding was <1E-6 for core damage frequency and <1E-7 for Large Early Release Frequency, a finding of very low risk significance (Green). This finding involved the cross-cutting area of human performance, the component of decision making and the aspect of decision communications, H.1(c), because a reactor operator failed to communicate the loss of component cooling to the RCP motors to the senior reactor operator which led to the failure to trip the 'A' RCP on exceeding the motor bearing high temperature limit. (Section 4OA5.3)
Inspection Report# : 2010004 *(pdf)*

Significance: G Mar 31, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Follow Procedures Results in Loss of Offsite Power to 1H and 2J Emergency Buses
A Green, self-revealing finding was identified for the licensee's failure to follow or adhere to licensee procedures for switchyard relay maintenance, human performance and management oversight, which resulted in the loss of the Technical Specifications (TS) required offsite circuit for the '1H' and '2J' emergency buses and the consequent auto-start of the respective emergency diesel generators (EDGs). The licensee entered this problem into their corrective action program as condition report 361280.

The inspectors determined that the failure to follow procedures to successfully accomplish nuclear switchyard relay maintenance was a performance deficiency (PD). The PD had a credible impact on safety due to the loss of a TS required offsite power supply and the start of the respective EDGs to restore power to the affected emergency buses. The inspectors determined the PD was more than minor because it impacted 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 power operations, and the related attribute of human performance due to human error in the implementation of a non-safety nuclear switchyard related procedures. In accordance with NRC Inspection Manual Chapter 0609, "Significant Determination Process," the inspectors performed a Phase 1 risk analysis and determined the finding was of very low safety significance (Green) because the finding did contribute to a reactor trip but did not contribute to the likelihood that mitigation equipment or functions would not be available. This finding involved the cross-cutting area of human performance, the component of the work practices, and the aspect of personnel use human error prevention techniques commensurate with risk for the assigned task, H.4(a), because a licensee technician's failure to use proper human error prevention techniques resulted in a partial loss of offsite power on both units.
Inspection Report# : 2010002 *(pdf)*

Significance: G Mar 31, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Establish a Procedure for Undervoltage Timers Results in Main Turbine/Reactor Trip
A Green, self-revealing finding was identified for the licensee's failure to establish an adequate procedure for calibration of under voltage timers which resulted in the failure of the Unit 2 'G' bus to fast transfer from 'C' reserve station service transformer (RSST) to the 'B' RSST and consequent loss of main turbine condenser vacuum causing a main turbine/reactor trip. The licensee entered this problem into their corrective action program as condition report 361280.

A self-revealing performance deficiency (PD) involving a Unit 2 turbine trip on loss of condenser vacuum was identified and resulted from the failure to establish an adequate procedure for the calibration of the time delay relays associated with the “G” bus cross-tie fast transfer circuit. This PD was the result of the failure to establish an adequate procedure for calibration of fast transfer relays. Specifically, licensee did not have mandated documentation in place to require technicians to use a proven method for timer calibration. The cause of PD was reasonably within the licensee’s ability to foresee and correct. Specifically, the NRC previously issued NCV 05000338/2008002-03, “Inoperability of '1H' EDG Due to Failure to Adequately Establish Procedural Requirements for Protective Relay Testing,” which involved a lack of procedural guidance and reliance of worker skill of the craft to successfully complete the activity. The PD adversely impacted 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 power operations and was related to the attribute of procedure quality because the correct timing relay calibration methodology was not documented in a procedure. In accordance with NRC Inspection Manual Chapter 0609, “Significant Determination Process,” a phase 1 significance determination process (SDP) screening determined that a phase 2 evaluation was required as the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available. A phase 3 SDP evaluation was performed by a regional SRA since the North Anna Risk Informed Inspection Notebook did not have the level of detail to accurately assess the finding. The NRC’s SPAR model was utilized to assess the risk significance of the finding modeling the impact of a loss of power to the 2G Bus without the fast transfer circuit available resulting in a reactor trip due to low condenser vacuum. The dominant sequence was a reactor trip without the condenser heat sink, caused by loss of power to the 2G bus with a failure of the cross-tie fast transfer circuit, with subsequent failures of main feedwater, auxiliary feedwater, and failure of feed and bleed cooling leading to core damage. The evaluation determined that the risk increase in core damage frequency was <1E-6 per year, a finding of very low safety significance, Green. This finding involved the cross-cutting area of human performance, the component of the resources, and the aspect of complete, accurate and up-to-date procedures, H.2(c), because the licensee failed to establish an accurate procedure to ensure correct calibration of under voltage timers.
Inspection Report# : 2010002 *(pdf)*

Mitigating Systems

Significance: G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Corrective Action for Fatigued Fuse Clips in Safety-Related Breakers
A Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the NRC for failure to promptly identify and correct a condition adverse to quality regarding fatigued fuse clips associated with safety-related breakers. The licensee entered this problem into their corrective action program as condition report 400128.

The inspectors determined that the failure to promptly initiate corrective actions for fatigued fuse clips was a performance deficiency (PD) which resulted in two safetyrelated breaker failures. The inspectors reviewed IMC 0612, Appendix B, and determined the PD was more than minor because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and the related attribute of design control for the initial structure, system, component design. In accordance with NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” the inspectors performed a Phase 1 analysis and determined that the finding was of very low significance because the finding was not a design deficiency, did not represent a loss of safety function and did not screen as potentially risk significant due to a seismic, flooding or severe weather initiating event. This finding involved the cross-cutting area of problem identification and resolution, the component of the corrective action program, and the aspect of thorough evaluation of problems such that resolutions address extent of condition, P.1(c), because the licensee failed to initiate adequate corrective actions to address extent of condition for fatigued fuse clips.
Inspection Report# : 2010005 *(pdf)*

Significance: Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Design Control Measures for Field Changes Affecting Station Battery Cables
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the failure to ensure that design control measures for field changes impacting the support of station battery cables were commensurate with those applied to the original design requirements. The licensee entered this problem into their corrective action program as condition report 358461.
The inspectors determined that the failure to adhere to the requirements of Criterion III for field changes involving the support of station battery cables was a performance deficiency (PD). This PD had a credible impact on safety due to an increase in battery post loading not analyzed by the vendor for a seismic event impacting the unsupported cables. The PD was more than minor, because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences and the related attribute of design controls due to changes made to battery cable supports which created a condition adverse to quality. In accordance with NRC Inspection Manual Chapter (IMC) 0609, “Significant Determination Process,” the inspectors performed a Phase 1 analysis and determined that the finding was of very low significance (Green) because the design deficiency did not result in the loss of functionality. The finding had no cross-cutting aspects because it is not indicative of current licensee performance.
Inspection Report# : 2010005 *(pdf)*

Significance: Sep 30, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to correctly translate turbine driven auxiliary feedwater pump lube oil subsystem vent design basis into specifications or drawings
A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified for the failure to correctly translate the design basis of the Unit 2 turbine driven auxiliary feedwater pump (TDAFWP) lube oil subsystem vent into specifications or drawings. The licensee entered this problem into their corrective action program as condition report 378798.

The inspectors determined that the licensee’s failure to correctly translate the Unit 2 TDAFWP lube oil subsystem vent into specifications or drawings as required by Criterion III was a performance deficiency (PD). The inspectors reviewed IMC 0612, Appendix E and determined the PD was more than minor, because it was similar to examples 3b and 3k in that the failure to correctly translate the design into drawings adversely impacted the operation of the system and resulted in reasonable doubt about the operability of the system. The inspectors reviewed IMC 0609 Attachment 4 and determined that the finding was of very low safety significance, or Green, because the finding was a design or qualification deficiency confirmed not to result in loss of operability or functionality. The cause of this finding did not involve a cross-cutting aspect because it is not indicative of current licensee performance. (Section 4OA3.1.2)

Inspection Report# : 2010004 *(pdf)*

Significance: Jun 30, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Promptly Correct a Condition Adverse to Quality For 2-RH-MOV-2700 Breaker
Green: A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct a condition adverse to quality for the breaker associated with 2-RH-MOV-2700, Loop ‘A’ Hot Leg to RH Pump Isolation Valve. The licensee entered this problem into their corrective action program as condition report 372940.

The inspectors determined that the licensee’s failure to promptly correct a known condition adverse to quality, as required by 10 CFR 50, Appendix B, Criterion XVI, was a performance deficiency. The inspectors reviewed IMC 0612, Appendix E and determined the finding was more than minor because it was similar to examples 4d and 4f. The phase 1 screening resulted in a need to perform phase 2 and phase 3 evaluations due to the finding resulting in the loss of mitigating function, specifically the ability to perform decay heat removal. A phase 3 analysis was performed by a regional senior risk analyst in accordance with the guidance of NRC Inspection Manual Chapter 0609 Appendix A. The significance determination process phase 3 risk evaluation resulted in a risk increase for the finding <1E-6 for core damage frequency (CDF) and <1E-7 for large early release frequency (LERF). The dominant sequence involved a steam generator tube rupture, followed by failure of the RHR system, and failure of the operators to refill the emergency condensate storage tank to continue secondary side cooling. The analysis assumed the operators, given the additional time while cooling the core using the secondary side, would be able to manually open 2-RH-MOV-2700. The finding was characterized as of very low safety significance (Green). The cause of this finding involved the cross-cutting area of problem identification and resolution, the component of corrective action program, and the aspect of implementation of corrective action (P.1(d)) because the licensee failed to correct the safety issue that existed with 2-RH-MOV-2700 in a timely manner, commensurate with its safety significance and complexity.
Inspection Report# : 2010003 *(pdf)*

Significance: Mar 31, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Procedure Implementation Results in Inoperability of a Fire Suppression System
A self-revealing, non-cited violation of Technical Specifications 5.4.1d was identified for the failure to adequately implement procedural requirements to reset a high pressure carbon dioxide (CO2) fire suppression system for the EDG pump room #1. The licensee entered this problem into their corrective action program as condition report 371298.

A self-revealing performance deficiency (PD) was identified for the failure to adequately implement fire protection procedure requirements of 0-PT-104.2 to ensure zone 3 was adequately reset. This PD had a credible impact on safety due to an inoperable fire suppression system for safety-related components. The PD was more than minor because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and the respective attributes of external events regarding fire due to the adverse impact on the capability of the fire suppression system and human performance due to the failure to properly implement
a test procedure. In accordance with NRC Inspection Manual Chapter 0609, “Significant Determination Process,” the inspectors performed a Phase 1 analysis and determined the finding was of very low safety significance (Green) because core damage frequencies related to the fuel oil pump room #1 were less than 1E-6 and
the duration of the system inoperability was less than three days. This finding involved the cross-cutting area of human performance, the component of work practices and the aspect of personnel do not proceed in the face of unexpected circumstances, H.4(a), because licensee personnel encountered problems with a CO2 system reset and failed to stop for proper guidance from supervision.
Inspection Report# : 2010002 *(pdf)*

Significance: Jan 25, 2010
Identified By: NRC
Item Type: FIN Finding
Failure to Ensure RSST ‘A’ LTC Controller Settings Were Correctly Implemented
Green: The inspectors identified a finding having very low safety significance (Green) involving the failure of the licensee to ensure that the control settings for the non-safety related reserve station service transformer (RSST) ‘A’ replacement load tap changer (LTC) controller installed through design change package (DCP) 05-108 were correctly implemented such that the LTC could respond as expected and credited across the range of design conditions. The licensee declared the RSST inoperable and implemented a change to the controller settings in compliance with design, and is tracking further actions under CR 358215.

The inspectors concluded that the finding was more than minor in that it is associated with the reactor safety

mitigating systems cornerstone attribute of equipment performance and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failure of the LTC to operate, as credited, due to incorrect LTC controller set points or inadequate control voltages, would have caused the 4kV safety related buses to prematurely disconnect from offsite power during a design basis event. The finding is of very low safety significance as it did not result in an actual loss of safety function. Further, this finding did not constitute a violation of NRC requirements as the RSST ‘A’ is a non-safety related component. The team also evaluated the finding for cross-cutting aspects and determined it to involve a failure to ensure adequately trained resources were available to design, check, and review complex digital controllers and their settings, and so involved the human performance (H) resources component cross cutting aspect (H.2.(c)).
Inspection Report# : 2009007 *(pdf)*

Significance: G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Ensure the Adequacy of Control Voltage to the 4160 and 480 VAC Equipment
Green: The team identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to ensure the adequacy of control voltage to the 4160 and 480 VAC equipment in support of mitigating system loads; specifically, a lack of voltage drop analysis for 125 VDC control power to breaker open/close coil, spring charging motors, and other miscellaneous DC loads. The licensee entered this issue into the corrective action program as CR361181.

The inspectors concluded that the finding is more than minor in that it involves the mitigating systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.
The inspectors determined the failure to assure and verify that adequate control voltage was available to close and open the 4160 VAC and 480 VAC breakers could have affected the capability of safety-related equipment to respond to initiating events. The finding is of very low safety significance as it did not result in an actual loss of safety function. The team also evaluated the finding for cross-cutting aspects and none were identified as this was determined to not be indicative of current licensee performance.
Inspection Report# : 2009007 *(pdf)*

Barrier Integrity

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

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