D.C. Cook 2
1Q/2003 Plant Inspection Findings

Initiating Events

Significance: Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement a Corrective Action to Prevent Recurrence Associated with Reactor Control Instrumentation Power Supply Failures
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to take corrective action to preclude the repetition of reactor control instrumentation 24-volt direct current power supply failures. Specifically, the licensee failed to perform weekly verification of control group power supplies to ensure that the "power available" status lights were lit. This corrective action was identified by the licensee in response to the Unit 2 reactor trip on May 12, 2002, which was caused by the failure of redundant power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The licensee subsequently performed this check on November 22, 2002, and discovered a failed 24-volt direct current power supply in Unit 1 cabinet 1-PS-CGC-16. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue could be reasonably viewed as a precursor to a significant event (i.e., potentially result in a reactor trip similar to the Unit 2 trip on May 12, 2002), and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets would upset plant stability (cause a reactor trip) and challenge the function of critical safety equipment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding contributes to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, the inspectors determinedthat this finding required a Phase 2 SDP analysis. After a review of additional information, the inspectors determined that a Phase 3 analysis was required. The Phase 3 SDP analysis, performed with the assistance of the NRC probabilistic risk analysis staff, determined that the resultant Core Damage Frequency and Large Early Release Frequency associated with this finding were less than 1E-6 per year and 1E-7 per year, respectively. Based on these results, this issue was determined to be of very low safety significance.
Inspection Report# : 2002009*(pdf)*

Significance: Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Assure that Prompt Corrective Actions were taken to Address Age-Related Failures of Reactor Control Instrumentation Power Supplies to Prevent Repetition of Power Supply Failures
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures, a significant condition adverse to quality. This issue was self-revealed on May 12, 2002, when an automatic reactor trip of Unit 2 occurred due to the failure of redundant 24-volt direct current power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The failure of both power supplies caused the number 21 steam generator feedwater regulating valve to close. Unit 2 subsequently

tripped on low steam generator water level coincident with low feedwater flow. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with the likelihood of continued failures of reactor control instrumentation power supplies and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets would upset plant stability (cause a reactor trip) and challenge the function of critical safety equipment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding contributes to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, the inspectors determined that this finding required a Phase 2 SDP analysis. After a review of additional information, the inspectors determined that a Phase 3 analysis was required. The Phase 3 SDP analysis, performed with the assistance of the NRC probabilistic risk analysis staff, determined that the resultant Core Damage Frequency and Large Early Release Frequencyassociated with this finding were less than 1E-6 per year and 1E-7 per year, respectively. Based on these results, this issue was determined to be of very low safety significance.
Inspection Report# : 2002009*(pdf)*

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Provide Appropriate Instructions for a Planned Shutdown of Unit 2 which Resulted in Unnecessarily Challenging the Automatic Start Funtion of Unit 2 Turbine Auxiliary Feedwater Pump
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The licensee failed to provide appropriate instructions for conducting a planned shutdown of Unit 2 on January 19, 2002, which resulted in unnecessarily challenging the automatic start function of Unit 2 turbine driven auxiliary feedwater pump (TDAFWP). This issue was self-revealed when the TDAFWP unexpectedly started due to low steam generator levels following the manual reactor trip. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective and was therefore more than a minor concern. Specifically, the function of critical safety equipment was challenged and plant stability was upset during the performance of a normal plant shutdown by the automatic start of Unit 2 TDAFWP. The inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding did not cause or contribute to the likelihood of an initiating event, the inspectors concluded that this issue was of very low safety significance.
Inspection Report# : 2002009*(pdf)*

Significance: G Jun 19, 2002
Identified By: NRC
Item Type: FIN Finding
Failure to Perform Switchyard Current Transformer Preventive Maintenance in Accordance with Vendor Recommendations
The inspectors identified a finding of very low safety significance for the failure to perform preventive maintenance on 345 kV switchyard current transformers in accordance e with vendor recommended schedules without adequate justification for the deviations. Specifically, the licensee tested 345 kV current transformers less frequently than recommended by the vendor and did not perform several recommended tests. The inspectors determined that this finding did not constitute a violation of NRC requirements. The inspectors concluded that testing switchyard equipment less frequently than recommended by the vendor credibly affected the objective of the initiating events cornerstone of

reactor safety. Performance of preventive maintenance testing that was less conservative than vendor recommendations could result in the failure to detect and repair component degradation, which could increase the likelihood of component failures. Consequently, the inspectors concluded that this issue could increase the loss of offsite power events and limit the ability of the licensee to mitigate power grid instability events. However, due to the availability of redundant power supplies to safety-related equipment, the inspectors determined that this issue was of very low safety significance.
Inspection Report# : 2002007*(pdf)*

Mitigating Systems

Significance: Mar 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Promptly Evaluation Operability of a Letdown Isolation Valve for Degraded Conditions
The licensee failed to promptly evaluate operability of the Unit 1 normal Reactor Coolant System letdown isolation valve 1-QRV-112 on two occasions when its ability to satisfy inservice testing program requirements could not be demonstrated. This issue was of very low safety significance since the redundant letdown isolation valve, 1-QRV-111, was available during the period that 1-QRV-112 was inoperable and therefore no actual loss of safety function occurred. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified.
Inspection Report# : 2003002*(pdf)*

Significance: Mar 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Evaluate Inservice Testing Failures of a Steam Generator Power Operated Relief Valve
Licensee personnel failed to promptly evaluate operability of number 23 steam generator power operated relief valve (PORV) 2-MRV-233 following inservice testing failures on two occasions. This issue was of very low safety significance since the redundant steam generator PORVs were available and therefore no actual loss of safety function occurred. One Non-Cited Violation of Technical Specification 4.0.5.a was identified.
Inspection Report# : 2003002*(pdf)*

Significance: Mar 03, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Corrective Action to Ensure That Only Turbine Trip Throttle Valve Latch Hooks with the Correct Geometry Would be Installed in the Turbine-Driven Auxiliary Feedwater Pumps
The licensee failed to take corrective action to ensure that only turbine trip throttle valve latch hooks with the correct geometry would be installed in the turbine-driven auxiliary feedwater pumps after determining that the incorrect part had been supplied by the manufacturer. This finding was determined to be a Non-Cited Violation of 10 CFR 50 Appendix B Criterion XVI, "Corrective Action". This finding was of very low safety significance because failure to take corrective action did not result in parts of incorrect geometry being installed in the auxiliary feedwater system and therefore did not affect the operability or availability of the system.
Inspection Report# : 2003004*(pdf)*

Significance: Mar 03, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Adequate Corrective Action to Revise Procedure 12-MHP-5021-056-007
The licensee failed to take adequate corrective action to revise procedure 12-MHP-5021-056-007, "Turbine-driven Auxiliary Feedwater Pump Trip and Throttle Valve Linkage Adjustment" to include the manufacturer's recommendations regarding the set-up of the turbine trip throttle valve. This finding was determined to be a Non-Cited Violation of 10 CFR 50 Appendix B Criterion XVI, "Corrective Action". This finding was of very low safety significance because the inadequate corrective action in revising the procedure did not affect the operability or availability of the auxiliary feedwater system.
Inspection Report# : 2003004*(pdf)*

Significance: Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Assure that Corrective Actions were taken to Preclude Repetition of EDG Starting Air System Relay Failures
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to assure that corrective actions were taken to preclude repetition of emergency diesel generator (EDG) starting air system relay failures, a significant condition adverse to quality. This issue was self-revealed when the failure of a starting air system relay for the Unit 2 AB EDG occurred on October 16, 2002, causing the engine to roll without a valid start signal. The inspectors subsequently identified that appropriate corrective actions to prevent repetition had not been taken following two previous age-related EDG air start relay failures in January 1999 and September 2000. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the repetitive EDG air start relay failures affected the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; (4) did not represent an actual loss of safety function of one or more Non-Technical Specification trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2002009*(pdf)*

Significance: Jun 30, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Provide Work Instructions Appropriate to the Circumstances for Adjustment of Stem Lock Nut on 2-CS-369
A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for maintenance procedures inappropriate to the circumstances, was self-revealed following gas binding of the Unit 2 West centrifugal charging pump. On February 16, 2002, the running charging pump became gas bound following attempts to switch the suction source from the volume control tank to the refueling water storage tank. Follow-up investigation

revealed that valve 2-CS-369 (reactor coolant pump seal water heat exchanger to volume control tank shutoff valve) was partially open, allowing transfer of volume control tank cover gas directly to the suction of the Unit 2 charging pumps. The licensee later determined that the position of the 2-CS-369 stem stop nut prevented full closure of the valve. Approximately two weeks prior to this event, the licensee replaced the diaphragm in 2-CS-369 using instructions provided in maintenance procedure 12 MHP-5021-001-023. However, the instructions contained in 12 MHP-5021-001-023 were inconsistent with vendor recommendations for stem stop nut adjustment and contributed to the failure to maintain proper positioning of the stem stop nut. The inspectors determined that the failure to provide procedures appropriate to the circumstances for the adjustment of the 2-CS-369 stem stop nut was a violation of NRC requirements. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue had a credible impact on safety and was therefore more than a minor concern. In particular, the gas intrusion into the suction of the running Unit 2 West centrifugal charging pump while aligned to the refueling water storage tank, a potential common cause failure mechanism for both of the Unit 2 charging pumps, impacted the capability of the high head injection system to provide the inventory and reactivity control safety functions. Therefore, the inspectors determined that this issue was associated with the mitigating systems cornerstone. During the Phase 1 SDP review, the inspectors concluded that this issue degraded the licensee's ability to add inventory to the reactor coolant system and therefore a Phase 2 SDP analysis was required. The Phase 2 shutdown risk SDP analysis, performed with the assistance of the Region III Senior Reactor Analyst and headquarters probabilistic risk assessment staff, determined that the total change in Core Damage Frequency associated with this condition was estimated to be approximately 3E-7 per year. The risk analysts reviewed several shutdown accident scenarios and determined that drain down to mid-loop operation after refueling to support vacuum refill of the reactor coolant system was the most limiting scenario. Based on the overall change in Core Damage Frequency, this issue was determined to be of very low safety significance.
Inspection Report# : 2002003*(pdf)*

Significance: G May 17, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Human Performance Weakness During the Degraded Essential Service Water Event of August 29, 2001 Associated with Control Board Monitoring and Procedural Adherence
The inspectors identified a Non-Cited Violation of Technical Specification (TS) 6.8.1 associated with operator procedural adherence deficiencies during the degraded essential service water event of August 29, 2001. Specifically, the operators failed to (1) effectively monitor the control boards for changing indications, adverse trends, and abnormal indications, (2) effectively communicate receipt of an abnormal temperature alarm for the component cooling water (CCW) heat exchanger, and (3) enter the CCW abnormal operating procedure as directed by the abnormal temperature alarm response procedure. The inspectors determined that the failure to adequately implement procedures associated with control board monitoring, logkeeping, and annunciator response had a credible impact on safety and therefore were more than a minor concern. Specifically, these issues could reasonably result in the failure to identify and promptly correct degradation of safety related equipment and therefore impact the reliability and availability of a safety system. Because these performance deficiencies contributed to delays in identifying degradation of the ESW and CCW mitigating systems, the inspectors determined that these human performance weaknesses were associated with the mitigating systems cornerstone. Although this issue adversely impacted the licensee's response to the August 29, 2001 event, none of the performance deficiencies directly resulted in the actual loss of safety system function or the loss of a single safety system train for greater than its TS allowed outage time. Consequently, the inspectors concluded that this issue was of very low safety significance.
Inspection Report# : 2001017*(pdf)*

Significance: W May 17, 2002
Identified By: NRC

Item Type: VIO Violation
Essential Service Water Strainer Maintenance Instructions Not Appropriate to the Circumstances
Documented instructions for essential service water (ESW) pump discharge strainer maintenance did not contain adequate detail regarding critical parameters for basket installation. Consequently, faulty strainer basket installation practices contributed to the failure of an ESW pump discharge strainer basket and created the potential for debris to bypass the strainer and enter the ESW system. On August 29, 2001, the failed Unit 1 East ESW pump discharge strainer, in conjunction with the ESW system alignment with all normal and alternate diesel generator (D/G) ESW supply valves open, caused significant debris fouling of the D/G heat exchangers. While operator actions prevented the debris fouling from causing a complete loss of the D/Gs ability to perform their emergency AC power safety function, the potential for a complete loss of all emergency AC power during a loss of offsite power was determined to exist. This finding was assessed using the applicable SDP as a potentially safety significant finding that was preliminarily determined to be of substantial safety significance. Final Significance Determination for a White Finding and Notice of Violation Letter issued on October 3, 2002, EA-01-286. SUPPLEMENTAL INSPECTION SUMMARY - INSPECTION REPORT 2003-04 The NRC performed this supplemental inspection to assess the licensee's evaluation of two White findings in the Mitigating Systems Cornerstone. The first White finding involved the failure to take appropriate corrective action to prevent the repetitive failure of the Unit 2 turbine-driven auxiliary feedwater (TDAFW) pump. The second White finding involved a failed essential service water (ESW) strainer basket, caused by inadequate strainer basket installation instructions, which permitted debris to bypass the strainer and enter the essential service water system, resulting in the debris intrusion event experienced at the D.C. Cook Nuclear Power Plant on August 29, 2001. During this supplemental inspection, a significant weakness was identified with regard to the licensee's evaluation of the findings. The licensee's evaluation adequately assessed the root causes, and appropriate corrective actions were initially assigned. The inspectors identified that two corrective actions assigned to perform important extent of condition reviews were not adequately completed. These reviews were to determine the extent of condition of the adequacy of maintenance procedures and to determine the extent of condition of equipment-related condition reports that were inadequately evaluated or closed. The failure to adequately complete the extent of condition reviews was determined to be a significant weakness in the licensee's evaluation. As a result, the two White performance issues associated with the Degraded Cornerstone will not be closed at this time.
Inspection Report# : 2001017*(pdf)*

Significance: G Apr 23, 2002
Identified By: NRC
Item Type: FIN Finding
Failure to Consistently Identify a Reasonable Apparent Cause for Conditions Adverse to Quality
The inspectors identified a Green finding for the failure to consistently identify reasonable apparent causes for conditions adverse to quality. The inspectors determined that the failure to consistently identify reasonable apparent causes for conditions adverse to quality could have had a credible impact on safety by affecting the availability, reliability, operability or functionality of mitigating equipment. This inspector identified finding was determined to be of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not result in a loss of function of a single train of any mitigating systems for greater than its Technical Specification allowed outage time and did not represent an actual loss of the safety function for any mitigating system; (3) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours; (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event in that the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding or severe weather initiating event; and (5) did not involve the loss of a safety function that contributed to external event initiated core damage accident sequences.
Inspection Report# : 2002004*(pdf)*

Significance: G Apr 23, 2002

Identified By: NRC
Item Type: NCV NonCited Violation

Failure to Take Prompt Corrective Action to Address Abnormal Degradation of the Safety-Related 250 Vdc Battery 2AB

The inspectors identified a Green finding that is being treated as a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to take prompt action to address abnormal deterioration of the safety-related 250 Vdc Battery 2AB. This finding was determined to be of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not result in a loss of function of a single train of a mitigating system for greater than its Technical Specification allowed outage time and did not represent an actual loss of safety function because the cracked cell covers and subsequent replacement activities did not render the 2AB battery incapable of supporting emergency electrical loads; (3) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event in that the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding or severe weather initiating event; and (4) did not involve the loss of a safety function that contributed to external event initiated core damage accident sequences.
Inspection Report# : 2002004*(pdf)*

Significance: W Mar 31, 2002
Identified By: NRC
Item Type: VIO Violation

Failure to Take Prompt Corrective Action to Prevent Repetitive Failure of the Unit 2 Turbine Driven Auxiliary Feedwater Pump

A Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," was identified for the licensee's failure to take prompt corrective actions to prevent a repetitive failure of the Unit 2 turbine driven auxiliary feedwater pump (TDAFWP). Specifically, the Unit 2 TDAFWP failed to start on August 10, 2001, due to the failure of the trip throttle valve latch mechanism to remain engaged during pump start. On December 13, 2001, the licensee obtained information from the trip throttle valve vendor identifying critical parameters for the trip hook mechanism geometry and alignment and failed to promptly perform corrective actions to verify that the Unit 2 TDAFWP trip hook conformed to these critical parameters. Consequently, a second failure of the Unit 2 TDAFWP occurred on January 18, 2002, due to the failure of the trip throttle valve latch mechanism to remain engaged during pump start. The inspectors and Region III Senior Reactor Analysts assessed this finding using the Significance Determination Process (SDP). A Phase 3 SDP analysis was performed using insights from the licensee's updated Probabilistic Risk Assessment model. Based on the results of the Phase 3 SDP analysis, the NRC staff determined that this finding has a low to moderate safety significance because the resultant 80 day fault exposure time represented an actual loss of safety function for a single train of auxiliary feedwater for greater than its Technical Specification allowed outage time and the train would have been unavailable if called upon for actual mitigation purposes. Final Significance Determination for a White Finding and Notice of Violation Letter issued on May 6, 2002, EA-02-010. SUPPLEMENTAL INSPECTION SUMMARY - INSPECTION REPORT 2002005 This supplemental inspection was performed to assess the licensee's evaluation of a White inspection finding that resulted from the licensee's failure to take appropriate corrective actions to prevent the repetitive failure of the Unit 2 TDAFWP. The pump failures were due to the unlatching of the TDAFWP trip throttle valve caused by incorrect machining of the trip throttle valve trip hook. During this supplemental inspection, performed in accordance with NRC Inspection Procedure 95001, the inspector concluded that the licensee performed a thorough root cause evaluation of the pump failures and identified the root cause and contributing cause for the events. The licensee's corrective actions were reasonable and appropriately addressed the causes and the extent of condition of the pump failures. However, the inspector concluded that the licensee's apparent cause evaluation, which was supposed to address the cause for the corrective action violation itself, did not adequately address why the licensee failed to take appropriate corrective actions to prevent a repetitive failure of the Unit 2 TDAFWP in January 2002. In response to the inspector's questions, the licensee re-opened the evaluation and provided reasonable corrective actions. Given the licensee's acceptable performance in addressing the repetitive TDAFWP failures, the White finding associated with this issue will only be considered in assessing plant performance for a total of four quarters in accordance with the guidance in NRC Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." SUPPLEMENTAL

INSPECTION SUMMARY - INSPECTION REPORT 2003004 The NRC performed this supplemental inspection to assess the licensee's evaluation of two White findings in the Mitigating Systems Cornerstone. The first White finding involved the failure to take appropriate corrective action to prevent the repetitive failure of the Unit 2 TDAFWP. The second White finding involved a failed essential service water strainer basket, caused by inadequate strainer basket installation instructions, which permitted debris to bypass the strainer and enter the essential service water system, resulting in the debris intrusion event experienced at the D.C. Cook Nuclear Power Plant on August 29, 2001. During this supplemental inspection, a significant weakness was identified with regard to the licensee's evaluation of the findings. The licensee's evaluation adequately assessed the root causes, and appropriate corrective actions were initially assigned. The inspectors identified that two corrective actions assigned to perform important extent of condition reviews were not adequately completed. These reviews were to determine the extent of condition of the adequacy of maintenance procedures and to determine the extent of condition of equipment-related condition reports that were inadequately evaluated or closed. The failure to adequately complete the extent of condition reviews was determined to be a significant weakness in the licensee's evaluation. As a result, the two White performance issues associated with the Degraded Cornerstone will not be closed at this time.
Inspection Report# : 2002005*(pdf)*
Inspection Report# : 2002002*(pdf)*

Barrier Integrity

Significance: Jun 30, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Containment Isolation Valve Alignment Error During Local Leak Rate Testing Resulted in Inoperable Containment Penetration During Refueling and Violation of Technical Specification 3.9.4.c
A Non-Cited Violation of Unit 2 Technical Specification (TS) 3.9.4.c was self-revealed for the licensee's failure to have the nitrogen to pressurizer relief tank containment penetration isolated prior to commencing core alterations. An operator incorrectly opened the instrument root shutoff containment isolation valve and removed the "Do Not Operate" tag from the valve without verifying the required position of the valve for local leak rate testing. This resulted in an inoperable containment penetration during refueling and resulted in the plant being in a higher risk configuration than that planned by the licensee. The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs and the valve was not in the correct position to fulfill its design safety function. The inspectors utilized the event information in conjunction with Appendix G, "Shutdown Operations Significance Determination Process," of Manual Chapter 0609, Table T-1, "Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer." This self-revealed issue was determined to be of very low significance by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee's ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because the small diameter penetration would be a very small leakage path, this issue was of very low safety significance.
Inspection Report# : 2002003*(pdf)*

Significance: Jun 30, 2002
Identified By: Self Disclosing

Item Type: NCV NonCited Violation
Failure of Lower Containnment Airlock Door Interlock and Failure to Follow Instructions Resulted in Inadvertent Opening of Both Airlock Doors
A Non-Cited Violation of Unit 2 Technical Specification (TS) 3.6.1.3 was self-revealed for the licensee's failure to have at least one containment airlock door closed while the airlock was inoperable with Unit 2 in Mode 3. The mechanical interlock on the lower containment personnel airlock malfunctioned and personnel opening the inner airlock door challenged the interlock by not verifying the outer door was closed prior to opening the inner door. This created a direct access path from the containment atmosphere to the outside atmosphere. The inspectors assessed this event using the Significance Determination Process (SDP). The inspectors determined that this issue had a credible impact on safety because the licensee failed to have at least one airlock door closed while the containment airlock was inoperable as required by the TSs and the resultant rapid containment pressure change also affected the operability of the ice condenser. The inspectors reviewed the guidance in NRC Inspection Manual Chapter 0609, Appendix H, "Containment Integrity SDP," and determined the finding was a Type "B" finding. Type "B" findings have no impact on the determination of Core Damage Frequency (CDF) and therefore they are not processed through the CDF based SDP. These findings, however, are potentially important to Large Early Release Frequency (LERF) determinations. The initial screening of the finding determined that the issue was potentially risk significant based on containment and ice condenser integrity which can be affected by the finding. The issue was therefore referred to the regional Senior Reactor Analyst (SRA) for further review. The analyst evaluated the circumstances of the issue to determine the actual duration of the finding. It was determined that the T/2 approach for fault exposure was not appropriate as the containment airlock doors were not discovered in the open position. In addition, the T/2 approach is generally used to estimate when a condition first occurred. The analyst therefore used the 5 second duration of time that the doors were actually opened, as each entry through the containment airlock is a deliberate, monitored activity (rather than a random event) and the licensee would be expected to identify the problem (both containment airlock doors opened simultaneously) as soon as it occurs. In determining the actual risk significance the SRA with the assistance of the headquarters containment risk analyst, utilized the LERF methodology identified in Appendix H for Type "B" findings. Utilizing this approach with actual plant specific probabilistic risk assessment values, the issue was determined to be of very low safety significance.
Inspection Report# : 2002003*(pdf)*

Significance: G Jun 30, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Technical Specification 3.9.4.c was Violated During Core Alterations When Containment Isolation Valve (2-XCR-101) was Stroked Open for Testing
A Non-Cited Violation of Unit 2 Technical Specification (TS) 3.9.4.c was self-revealed for the licensee's failure to maintain refueling integrity configuration control of containment penetration CPN-74 during core alterations when containment isolation valve 2-XCR-101 was stroked open for testing. Opening this valve created a direct access path from the containment atmosphere to the outside atmosphere. The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs. The inspectors utilized the event information in conjunction with Appendix G, "Shutdown Operations Significance Determination Process," of Manual Chapter 0609, Table T-1, "Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer." This issue was determined to be of very low significance by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee's ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because 2-XCR-101 was open for a short period of time and the small diameter penetration would be a very small leakage path, this issue was of very low safety significance.

Inspection Report# : 2002003*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Significance: N/A Apr 23, 2002
Identified By: NRC
Item Type: FIN Finding
Corrective Action Cross-cutting Finding for the Failure to Promptly Implement Effective Corrective Actions for Conditons Adverse to Quality Impacting the Mitigating Systems and Public Radiation Safety
Several findings associated with the implementation of the corrective action program were identified within the mitigating systems and public radiation cornerstone areas. The inspectors determined that the 6 findings identified in the past 12 months indicated an adverse performance trend and had a common causal factor associated with the failure to promptly and effectively resolve conditions adverse to quality. Although the individual findings highlighted were of very low safety significance the number of findings were determined to be a substantive cross-cutting issue indicative of an adverse performance trend pertaining to implementation of the corrective action program.
Inspection Report# : 2002004*(pdf)*

Significance: N/A Apr 23, 2002
Identified By: NRC
Item Type: FIN Finding
Summary Conclusion PI&R Inspection
The inspectors concluded that the licensee's corrective action program attributes enabled timely problem identification commensurate with the significance level and that the threshold for problem identification was low. Significance level of identified problems was appropriately characterized and the backlog items that were reviewed revealed that resolution of problems were prioritized based on safety significance. Based on information obtained during interviews, there was no evidence that a safety conscious work environment did not exist. Root cause evaluations were thorough and appropriate corrective actions for significant conditions adverse to quality were identified. However, while implementation of corrective actions to prevent recurrence of significant conditions adverse to quality was considered adequate, a recurring issue was identified regarding the failure to implement some corrective actions as prescribed in root cause evaluations. Four of the eight apparent cause evaluations reviewed by the inspectors failed to identify a

reasonable apparent cause of the problem. Therefore, the licensee's ability to consistently identify reasonable causes for conditions adverse to quality was considered inadequate which could adversely impact implementation of prompt and effective corrective actions to resolve the problem. Also, a review of previously documented findings revealed that an adverse performance trend exists regarding the ability to promptly and effectively resolve conditions adverse to quality which was considered a substantive cross-cutting issue.
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