D.C. Cook 2

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: 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: G 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: G 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: 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.

Inspection Report# : 2001017*(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: 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: 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: 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 200202-05 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."

Inspection Report# : 2002005*(pdf)*

Significance: G Mar 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Use Valid Acceptance Criteria for Stroke Time Testing the Unit 2 Pressurizer Power Operated Relief Valves
A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," was identified for the licensee's failure to utilize valid acceptance criteria for stroke time testing the Unit 2 pressurizer power operated relief valves (PORVs). Specifically, the licensee failed to assure that the correct acceptance criteria contained in the applicable design document were incorporated into the surveillance test procedure used for testing the PORVs. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors determined that this issue could become a more significant safety concern if left uncorrected and was therefore more than a minor concern. Specifically, the failure to adequately perform surveillance testing with valid acceptance criteria could reasonably result in the failure to identify degraded or inoperable safety related components. The inspectors also concluded that this issue could credibly affect the operability of the pressurizer PORVs, which are mitigating system components under the SDP. The inspectors determined that, because the as-found stroke times were found within the correct acceptance criteria, this issue was of very low safety significance.
Inspection Report# : 2002002*(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 200202-05 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."
Inspection Report# : 2002002*(pdf)*

Significance: G Feb 28, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Assess the Risk Associated with Maintenance Activities Affecting both Unit 2 Safety Injection System Pumps
A Non-Cited Violation of 10 CFR 50.65(a)(4) was identified for the licensee's failure to assess the risk associated with maintenance activities affecting both Unit 2 safety injection (SI) system pumps. Operators deviated from the licensee's outage schedule and prematurely vented and drained both Unit 2 SI system pumps without assessing the increase in shutdown risk during a period of reduced reactor coolant system (RCS) inventory. This resulted in the inadvertent entry into a higher shutdown risk configuration, for which the licensee had not implemented additional risk management actions to protect available equipment and to maintain an adequate level of defense as required by the licensee's plant shutdown safety and risk management procedure. The inspectors assessed this finding using the Significance Determination Process. The inspectors concluded that this issue had a credible impact on safety because the SI pumps were made unavailable for core cooling in the event of a loss of RCS inventory. At the time, Unit 2 was in Mode 5 (Cold Shutdown) with the RCS loops not filled and vented, and only one of the two Unit 2 centrifugal charging pumps was available. The inspectors reviewed the guidance in Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," including the checklist for "Pressurized Water Reactor Cold Shutdown and

Refueling Operation - Reactor Coolant System Closed and No Inventory in Pressurizer." Although having both SI pumps unavailable degraded the licensee's ability to add inventory to the RCS, the inspectors determined that sufficient plant equipment existed to keep the core covered because the capability existed for operators to cross-tie the Unit 1 and Unit 2 charging systems to make an additional centrifugal charging pump available. The inspectors concluded that this issue was of very low safety significance because there was no challenge to RCS inventory control during the time that the SI pumps were unavailable.
Inspection Report# : 2001020*(pdf)*

Barrier Integrity

Significance: G 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)*

Significance: G 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: G Feb 28, 2002

Identified By: NRC

Item Type: NCV NonCited Violation

Violation of 10 CFR Part 50.55a(g)(5)iii for Failure to Obtain NRC Concurrence Associated With Incomplete Nondestructive Weld Examinatons

A Non-Cited Violation of 10 CFR 50.55a(g)(5)(iii) was identified for failure to obtain NRC concurrence (Code relief) associated with incomplete weld examinations. This finding had the potential to affect the barrier integrity and initiating events cornerstones and was more than minor because, the reduced examination of welds was left uncorrected, which could result in operation with undetected flaws affecting the reactor coolant system pressure boundary. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low as determined by the Reactor Safety Significance Determination Process.

Inspection Report# : 2001020*(pdf)*

Significance: G Feb 28, 2002

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Failure to Properly Evaluate and Correct the Cause for an Inservice Stroke Time Test Failure in April 2001

A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the licensee's failure to adequately correct a failure of containment isolation valve 2-CCR-440 during routine inservice testing on April 11, 2001. Specifically, the licensee adjusted the 2-CCR-440 position indication limit switch mechanism to obtain indication of valve closure without verifying that 2-CCR-440 was capable of fully closing. Subsequently, on January 20, 2002, 2-CCR-440 failed a 10 CFR 50, Appendix J, leak rate test due to the valve not being fully closed. The inspectors assessed this finding using the Significance Determination Process. The inspectors concluded that this issue represented an actual degradation in the redundancy of a containment penetration barrier and had a credible impact on safety and was more than a minor concern. The inspectors determined that the failure of a containment isolation valve was associated with the containment barrier and was within the barrier integrity cornerstone. As described in Updated Final Safety Analysis Report, Table 5.4-1, "Unit 2 Containment Penetration Isolation Barriers," 2-CCR-440 was a barrier for containment penetration CPN-25. The second barrier for CPN-25 was composed of the closed component cooling water (CCW) system piping loop inside containment. Based on satisfactory Appendix J, Type C, leak rate test results for CPN-25 obtained on January 29, 2002, the inspectors determined that the CCW piping inside containment was intact and that failure of 2-CCR-440 to fully close did not represent an actual open pathway in the physical integrity of the reactor containment. Consequently, the inspectors determined that the failure to properly evaluate and correct the cause for the inservice stroke time test in April 2001, did not result in an open leak path from the Unit 2 containment.

Inspection Report# : 2001020*(pdf)*

Significance: G Feb 28, 2002

Identified By: NRC

Item Type: NCV NonCited Violation

Violation of TS 4.0.5.a for Application of Incorrect Acceptance Criteria Applied To Flaws in the Pressurizer Welds

A Non-Cited Violation of Technical Specification 4.0.5.a was identified for application of incorrect acceptance criteria to flaws in the pressurizer vessel welds. This finding had the potential to affect the initiating events and barrier integrity cornerstones and was more than minor because these types of errors, if left uncorrected, could result in acceptance of a flaw size greater than that allowed by the American Society of Mechanical Engineers Code. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low as determined by the Reactor Safety Significance Determination Process.

Inspection Report# : 2001020*(pdf)*

Significance: Feb 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Impement the Requirements of Surveillance Procedure 12-MHP 4030.010.003
A Non-Cited Violation of Technical Specification (TS) 6.8.1 was identified for the licensee's failure to adequately implement the requirements of 12-MHP 4030.010.003, "Ice Condenser Lower Inlet Door Surveillance." Specifically, the licensee failed to adequately perform the following: (1) install a protective end tip on the spring scale to protect the lower ice condenser doors from damage as required by step 4.2.6, (2) ensure that installation of the TE-132 test fixture met the moment arm and degree of opening requirements in accordance with steps 4.2.3 and 4.2.5, and (3) accurately record surveillance test data for lower inlet door limit switch checks as required by steps 4.1.8.d and 4.1.9.d. The inspectors assessed this finding using the Significance Determination Process. The inspectors determined that the failure to correct these procedural implementation inadequacies could become a more significant safety concern if left uncorrected and was therefore more than a minor concern. Specifically, the failure to adequately perform surveillance testing could result in the failure to identify degraded or inoperable safety related equipment. Because the ice condenser was primarily associated with containment heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the barrier integrity cornerstone. Because the Unit 2 ice condenser was not required to be capable of performing a safety related function immediately following the inadequate surveillance testing on January 24, 2002, the inspectors concluded that this issue did not result in an actual loss or degradation of the heat removal function performed by the ice condenser.
Inspection Report# : 2001020*(pdf)*

Significance: Feb 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Measure the Ice Condenser Lower Inlet Door Opening Torque and Closing Torque in Accordance with Technical Specification Requirements
The inspectors identified a Non-Cited Violation of Unit 2 Technical Specification (TS) 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 requirements associated with testing of the ice condenser lower inlet doors. Contrary to the TS requirements, previous TS 4.6.5.3.1.b surveillance testing performed in Unit 2 on April 21, April 22, and May 4, 2000 failed to adequately measure the door opening torque and the door closing torque in accordance with the TS requirements. Specifically, the methodology used by the licensee to perform TS 4.6.5.3.1.b.3 and 4.6.5.3.1.b.4 testing resulted in door closing torques that were greater in magnitude than the door opening torques, contrary to the TS description of these torque values. The inspectors identified that the measured opening torque values for all the Unit 2 lower inlet doors were less than the associated door closing torque values. Because calculation of the door frictional torque required accurate measurement of the door opening and closing torques, the licensee was unable to demonstrate compliance with the requirements of TS 4.6.5.3.1.b.5. The inspectors assessed this finding using the Significance Determination Process. The inspectors determined that the failure to adequately implement TS 4.5.6.3.b requirements for testing of the Unit 2 lower inlet doors had a credible impact on safety and was more than a minor concern. As stated in the TS 3.6.5 bases, operability of the ice condenser doors ensures that reactor coolant fluid released during a loss of coolant accident (LOCA) will be diverted through the ice condenser bays for heat removal. The ice condenser also augmented the containment recirculation sump water inventory in the event of certain small break LOCAs and limited ice maldistributions within the ice condenser. Because the proper functioning of the ice condenser lower inlet doors was primarily associated with the heat removal function of the ice condenser, the inspectors determined that this issue was associated with the barrier integrity cornerstone. Based on a review of additional testing results for the Unit 2 lower inlet doors performed on February 3 and 4, 2002, the inspectors concluded that there was no actual reduction in the atmospheric pressure control function of the reactor containment nor a loss of capability to provide additional recirculation sump inventory during certain small break LOCAs.
Inspection Report# : 2001020*(pdf)*

Significance: Feb 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of 10 CFR Part 50 Appendix B, Criterion IX for Failure to Correctly Calibrate an Ultrasonic Transducer
A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion IX was identified for an inadequate calibration of an ultrasonic transducer used to size flaws in the pressurizer girth weld. This finding had the potential to affect the initiating events and barrier integrity cornerstones and was more than minor because it had a credible impact on safety, in that, errors in the ultrasonic testing calibration invalidated the flaw sizes recorded. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low as determined by the Reactor Safety Significance Determination Process.
Inspection Report# : 2001020*(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.
Inspection Report# : 2002004*(pdf)*

Last modified : March 25, 2003