D.C. Cook 1

Initiating Events

Significance: G Dec 28, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Provide an Appropriate Procedure for Testing the Unit 1 Pressurizer Power Operated Relief Valves Causing an Uncontrolled Release of Reactor Coolant System Inventory
A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed. The licensee failed to provide an appropriate procedure for testing the Unit 1 pressurizer power operated relief valves (PORVs), causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank. This issue was self-revealed on June 5, 2002, when pressurizer PORV 1-NRV-153 inadvertently opened while testing actuation logic circuitry for pressurizer PORV 1-NRV-151. The surveillance test procedure failed to provide adequate control of 1-NRV-151 and 1-NRV-153, which have a common automatic opening signal. The release rate exceeded the 25 gallons-per-minute limit established for declaring an Unusual Event in accordance with the licensee's Emergency Plan. 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 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 uncontrolled release of reactor coolant system inventory upset plant stability and challenged the inventory control safety function. Because Unit 1 was in a shutdown mode during this period, the inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process." Based on the plant conditions at the time, the inspectors concluded that the most appropriate Appendix G checklist to use for this issue was the checklist for "Pressurized Water Reactor Hot Shutdown Operation - Time to core boiling less than 2 hours." Because, operator intervention was required to manually close the affected PORV block valve, the inspectors concluded that the unit was in a configuration where a single active failure or personnel error could have resulted in a rapid loss of reactor coolant system inventory as described in Section II.B.(2) of the checklist. Consequently, the inspectors concluded that this issue increased the likelihood of a loss of reactor coolant system inventory and therefore required a Phase 2 SDP analysis. The inspectors discussed the safety significance of this issue with the Regional Senior Reactor Analyst (SRA). The SRA reviewed the finding and determined that the drain path could be easily isolated, accurate reactor coolant system level indication was available, all steam generators were available for cooling, and all trains of standby injection were available and not impacted by the finding. Based on these factors the finding was determined to be 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 Implement Adequate Foreign Material Exculsion Controls Resulted in Degradation of Unit 1 West ESW Pump
A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed following the identification of foreign material in the Unit 1 West essential service water (ESW) pump. On June 24, 2002, the licensee identified a rapid degradation in the performance of the Unit 1 West ESW pump. Subsequent investigation identified that plastic barrier tape, a foreign material, had been ingested by the pump and had become wound tightly around the pump's impeller. The inspectors concluded that the licensee failed to establish appropriate work controls to control foreign material in areas adjacent to the Unit 1 West ESW pump in accordance with the requirements of PMI-2220, "Foreign Material Exclusion." The inspectors evaluated this failure to establish appropriate foreign material controls in the vicinity of the Unit 1 West ESW pump using the Significance Determination Process. The inspectors determined that this issue had a credible impact on safety and was more than a minor concern. Specifically, ingestion of foreign material by the Unit 1 West ESW pump degraded pump performance and rendered the pump inoperable, which affected the reliability and capability of the ESW system. The safety function of the ESW system is to provide sufficient cooling capacity for continued operation of safety-related equipment during normal and accident conditions. Consequently, the inspectors determined that this issue affected the objectives of the mitigating systems cornerstone. The inspectors concluded that this issue did not result in an actual loss of the safety function of a single train of ESW for greater than the TS allowed outage time. Additionally, because of the continued availability of ESW capability from both of the Unit 2 ESW trains and the Unit 1 East ESW train, the inspectors concluded that the foreign material ingestion did not result in an actual loss of the ESW system safety function. Consequently, the inspectors concluded that this issue was 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: 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 Resolve a Degraded Condition of the Unit 1 East Motor Driven Auxiliary Feedwater Pump Room Cooler
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 corrective action to resolve a degraded condition of the control circuitry on the Unit 1 East Motor Driven Auxiliary Feedwater Pump room cooler. 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 the auxiliary feedwater mitigating system for greater than its Technical Specification allowed outage time in that necessary repairs for the room cooler following the August 21, 2001, failure were completed within the allowed outage time. Also, the auxiliary feedwater pump room temperatures were maintained within the required temperature bands during the February 2001 failure; (3) did not represent an actual loss of the auxiliary feedwater system safety function; (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)*

Barrier Integrity

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify and take Appropriate Corrective Actions to Preclude the Failure of Reactor Coolant System Pressure Boundary Charging Line Check Valves which were at Risk of Common Cause Failure
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to identify and take appropriate corrective actions to preclude the failure of four Unit 1 reactor coolant system pressure boundary charging line check valves (Velan Model B10-3114B-13M), which were at risk of common cause failure due to industry identified design and manufacturing

defects, a significant condition adverse to quality. This issue was self-revealed when the check valves were all found to be stuck in either the full or partially open position during radiographic nonintrusive testing in May 2002. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this finding was associated with the barrier integrity cornerstone and adversely affected the cornerstoneobjective, and as such it was more than a minor concern. Specifically, the charging line check valves perform a safety-related function of limiting the release of reactor coolant inventory should a charging line failure occur. The failure of the valves in the open position would prohibit the performance of this function and therefore affects the objective of the barrier integrity cornerstone. 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 involved the integrity of the reactor coolant system barrier, the inspectors determined that this finding required a Phase 2 SDP analysis. After consulting with the Regional Senior Reactor Analyst, the inspectors determined that this issue was of very low safety significance because no actual loss of safety function occurred. The inspectors concluded that no actual loss of safety function occurred based on the reported minimal force required to shut the valves (indicating they would have shut given the differential pressure applied during accident conditions) and the redundancy provided by a third check valve (1-CS-321) in the charging line. In accordance with IMC 0609, Appendix A, Attachment 1, Step 2.6, the SDP results were not evaluated for potential risk contribution due to Large Early Release Frequency because the accident sequence result was less than 1E-7 per year.
Inspection Report# : 2002009*(pdf)*

Significance: G Jun 30, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Measure Unit 1 Lower Ice Condenser Inlet Door Opening Torque and Closing Torque in Accordance TS Requirements
The inspectors identified a Non-Cited Violation of Unit 1 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 1 on November 21, 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 36 of 48 Unit 1 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 1 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 augments the containment recirculation sump water inventory in the event of certain small break LOCAs and limits 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 1 lower inlet doors performed in May 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. Therefore, this 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
Pressurizer Power Operated Relief Valves Inoperable Due to Mis-Positioned Control Switches
A Non-Cited Violation of Unit 1 Technical Specification 3.4.11.c was self-revealed. An operator incorrectly positioned the control switches for pressurizer power operated relief valves (PORVs) 1-NRV-152 and 1-NRV-153, rendering the valves unavailable for automatic pressure control. With Unit 1 in Mode 1 and two PORVs inoperable due to causes other than excessive seat leakage, the licensee failed to restore at least one of the inoperable PORVs to operable status within the following 72 hours or be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours. 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 two PORVs were not capable of automatically controlling reactor coolant system (RCS) pressure below the setting of the pressurizer code safety valves, thereby reducing challenges to these valves. At the time of this event, the third pressurizer PORV (1-NRV-151) was already unavailable (automatic function only) with its manual isolation valve closed due to excessive seat leakage. Therefore the automatic function of all three PORVs was disabled. Although all three PORVs were not capable of automatic operation, the valves were still capable of manual operation to mitigate a steam generator tube rupture accident or as an alternate means of decay heat removal during plant shutdown. The inspectors concluded that this issue affected the operability of the pressurizer PORVs, which are barrier integrity components under the SDP designed to maintain the integrity of the RCS. The inspectors performed a Phase 2 SDP analysis for this finding using the following assumptions: (1) manual operation of the PORVs for primary heat removal using the feed and bleed safety function was not affected; therefore, the inspectors only evaluated the Anticipated Transients Without Scram (ATWS) initiator which considered the primary relief safety function; (2) the duration of the performance deficiency was 13 days; and (3) operator action to manually actuate the failed automatic function of the PORVs was credited. Results of the Phase 2 ATWS worksheet determined that only one accident sequence was affected and resulted in this issue being characterized as having very low safety significance. In accordance

with NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, Step 2.6, the SDP results were not evaluated for potential risk contribution due to Large Early Release Frequency because the accident sequence result was less than 1E-7 per year.
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)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Jun 30, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Implement All Intended Radiological Engineering Controls During Steam Generator Eddy Current Testing, as Required by 10 CFR 20.1701
A Non-Cited Violation of 10 CFR 20.1701 was identified for the licensee's failure to utilize all intended radiological engineering controls to limit the concentration of radioactive material in air during steam generator eddy current testing, resulting in intakes to four workers. This finding was determined to be of very low safety significance since radiation exposures to involved workers were low relative to regulatory limits, and because radiological conditions were not of a magnitude sufficient to create a substantial potential for an overexposure.
Inspection Report# : 2002003*(pdf)*

Public Radiation Safety

Physical Protection

Miscellaneous

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)*

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)*

Last modified : March 25, 2003