D.C. Cook 1
3Q/2007 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: Jun 29, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Use of Incorrectly Configured Test Leads Rendered Two EDGs Inoperable
A finding of very low safety significance with an associated NCV of TS 5.4.1.a was self-revealed. On two separate occasions, a maintenance craftsman performed procedure steps to connect a multi-meter to an emergency diesel generator (EDG) kilowatt meter using incorrectly configured test leads, which caused a short-circuit and subsequent failure of a fuse in the EDG metering circuit when the engine was started during surveillance testing. This adversely affected the operability and availability of both the Unit 1 AB and CD EDGs. Corrective actions included replacing the fuses, coaching the maintenance craftsman involved with the incidents, and temporary suspension of his qualifications.

This finding was of more than minor significance because it is related to the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the use of incorrectly configured test leads rendered the EDGs inoperable and unavailable to perform their safety function. The finding was of very low safety significance because it did not represent a design or qualification deficiency, loss of safety function for a single train for greater than its TS allowed outage time, and was not risk-significant due to external event initiators. The primary cause of this finding was related to the cross-cutting area of human performance because the licensee's human error prevention techniques were not used commensurate with the risk of the task being performed. Specifically, the maintenance craftsman failed to appropriately control the test leads and to use self-verification techniques to ensure that correctly configured test leads were used during EDG testing. (IMC 0305 H.4(a))
Inspection Report# : 2007004 *(pdf)*

Significance: Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Demonstrate Performance or Condition of Nuclear Instruments Were Effectively Controlled Through Performance of Appropriate Preventive Maintenance
The inspectors identified a finding of very low safety significance and an Non-Cited Violation of 10 CFR 50.65(a)(2). The licensee failed to demonstrate that the performance or condition of the Unit 1 and Unit 2 power range and intermediate range nuclear instruments was effectively controlled through appropriate preventive maintenance. As a result, the licensee failed to establish goals or monitor the performance of these instruments in accordance with paragraph (a)(1) of the Maintenance Rule to ensure that appropriate corrective actions were taken. The licensee was further evaluating corrective actions, including training, for this issue at the end of the inspection period and had placed the system into 10 CFR 50.65(a)(1) status.

This finding was of more than minor significance because violations of 10 CFR 50.65(a)(2), such as failure to demonstrate effective control of performance or condition and failure to classify the affected structure, system, or components (SSC) in (a)(1) status, involve degraded SSC performance or condition. The finding was of very low safety significance because the finding was associated with the Mitigating Systems Cornerstone and did not represent a design or qualification deficiency, loss of safety function for a train or system, and was not risk-significant due to

external event initiators. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because the licensee failed to thoroughly evaluate multiple nuclear instrumentation component failures by appropriately completing the Maintenance Rule Evaluations. (IMC 0305, P.1(c))
Inspection Report# : 2007003 *(pdf)*

Significance: Mar 02, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify and Correct a Condition Adverse to Quality
The inspectors identified a finding having very low safety significance and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” for the licensee’s failure to promptly identify that the Unit 1 Train A (1-CD) emergency diesel generator (EDG) would exceed its capacity rating. Specifically, the 1-CD EDG’s capacity rating would have been exceeded if the 1-CD EDG was allowed to run at the upper frequency band of 61.2 Hz as allowed by Technical Specifications (TS). As a result, the licensee performed corrective action calculations to assess the finding and on March 1, 2007, imposed an operational upper frequency limit of =60.5Hz on the station’s Unit 1 EDGs. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not take appropriate corrective action to address the safety issue in a timely manner commensurate with its safety significance and complexity.

This finding was more than minor because the 1-CD EDG would have exceeded its design load rating at the maximum TS allowed frequency of 61.2Hz. Without the evaluation and imposing an administrative limit, the licensee could not ensure that the 1-CD EDG would reliably perform its safety related-function. The finding was of very low safety significance based on a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations.”
Inspection Report# : 2007002 *(pdf)*

Significance: Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Accept for Continued Service, by Correction, or Evaluation or Test, a Degraded Chemical Volume Control System Support
The inspectors identified a Non-Citied Violation of 10 CFR Part 50.55(a)(g)(4), for failure to accept for continued service, by correction, or evaluation or test, a chemical and volume control system (CVCS) support (2-ACS-R-913) whose examination detected a condition (loose anchor plate nut) unacceptable for continued service in accordance with American Society of Mechanical Engineers (ASME) Section XI Code. The licensee, having instead dispositioned the condition in accordance with operability screening procedure ES-PIPE-1002-QCN, subsequently completed an analysis to confirm that the support was operable with this configuration and entered this issue into their corrective action program.

This finding was of more than minor significance because the licensee neither corrected this condition (e.g., tighten the loose nut) nor completed an engineering evaluation or test to confirm the ability of this support to carry design loads as required by ASME Code prior to returning it to service. The failure to repair or to perform an engineering evaluation that demonstrated this degraded CVCS support would carry design loads, increased the likelihood of a component failure and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance because the licensee subsequently completed an evaluation which confirmed that the support was operable in this configuration. In particular, it did not affect the availability or function of the mitigating system.

This has a cross-cutting aspect in the area of human performance because the licensee’s screening procedure was not adequate and directed the licensee to perform actions contrary to the requirements of the ASME Code.
Inspection Report# : 2006007 *(pdf)*

Barrier Integrity

Significance: Mar 02, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correct Inadequate Safety Analysis Dose Calculations
The inspectors identified a finding having very low safety significance and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” for failure to promptly identify and correct a condition adverse to quality regarding inadequate safety analysis dose calculations. Specifically, the licensee failed to address the aggregate effect of various nonconforming conditions on containment leakage rates for offsite dose and control room calculations to ensure that accurate and adequate margin remained available for offsite dose analyses and control room habitability. The finding was entered into the licensee’s corrective action program and an operability determination evaluation was initiated during the inspection. The primary cause of this violation was related to the cross-cutting area of problem identification and resolution because the licensee did not thoroughly evaluate known discrepant conditions.

This finding was more than minor because the licensee did not verify the capability of containment to maintain the offsite and control room dose within required limits under post-accident conditions to the values assumed in the analyses. The finding was of very low safety significance based on a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.”
Inspection Report# : 2007002 *(pdf)*

Significance: Mar 02, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Previously Imposed Compensatory Measures
The inspectors identified a finding having very low safety significance and an associated Non-Citied Violation of 10 CFR Part 50.36, “Technical Specifications.” Specifically, the licensee failed to maintain previously imposed administrative limits (i.e., compensatory measures) required by non-conforming Updated Final Safety Analysis Report offsite and control room dose analyses. The station operated from April 25, 2003, through February 28, 2007, based on analyses that included assumed containment leakage values that were not bounded by the licensee’s TS 5.5.14, “Containment Leakage Rate Testing Program.” Once the finding was identified by the inspectors, the licensee re-imposed the required compensatory measures during the inspection. The primary cause of this violation was related to the cross-cutting area of human performance because the licensee failed to communicate decisions with respect to containment leakage and the basis for those decisions to personnel.

The finding was more than minor in accordance with Inpection Manual Chapter (IMC) 0612, Appendix B because the finding was associated with the configuration control (containment design parameters maintained) attribute of the Barrier Integrity Cornerstone and affected the cornerstone’s objective of maintaining the functionality of containment. Specifically, the licensee did not re-impose compensatory measures to limit the maximum allowable containment leakage rate to the values assumed in the analyses. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.”
Inspection Report# : 2007002 *(pdf)*

Significance: SL-IV Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform 10 CFR 50.59 Evaluation to Remove Design Basis Requirement from Updated Final Safety Analysis Report.
The inspectors identified a Non-Citied Violation of 10 CFR 50.59 because the licensee failed to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the Updated Final Safety Analysis Report (UFSAR). In 10 CFR 50.59 Screen No. 2006-0041, "Replace Unit 1 Reactor Vessel Closure Head (1-OME-1)," Revision 0, the licensee evaluated an UFSAR change that removed the emergency load condition specified in UFSAR Tables 2.9-1 and 2.9-2. Within the 10 CFR 50.59 screen, the licensee failed to identify that the proposed activity involved revising or replacing an UFSAR described evaluation methodology that is used in establishing the design bases. As a result, a 10 CFR 50.59 evaluation for the UFSAR change was not performed. The licensee entered the

issue into its corrective action program (AR 00803398 and AR 00803828).

The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. However, where possible, the underlying technical issue is evaluated under the Significance Determination Process to determine the severity of the violation. In this case, the finding screened as having very low safety significance (Green) using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the inspectors answered "no" to LOCA initiators question 1 under the Initiating Events Cornerstone column of the Phase 1 worksheet. Specifically, since the licensee had evaluated the faulted loading condition as part of the design basis for the replacement reactor vessel closure head, the finding was not a design or qualification deficiency that was confirmed to result in a loss of operability or functionality per Part 9900 Technical Guidance, "Operability Determination Process for Operability and Functional Assessment." Therefore, the finding would not have likely affected other mitigating systems resulting is a total loss of their safety function.
Inspection Report# : 2006007 *(pdf)*

Emergency Preparedness

Significance: SL-III Oct 25, 2006
Identified By: NRC
Item Type: VIO Violation
Failure to Provide Complete and Accurate Information to the NRC Which Impacted a Licensing Decision
The inspectors identified an apparent violation of 10 CFR 50.54(q) involving 10 CFR 50.47(b)(4). Title 10, Part 50, Section 54(q) of the Code of Federal Regulations states in-part, “the nuclear power reactor licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E to this part." Title10, Part 50, Section 47(b)(4) of the Code of Federal Regulations states in part, "a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee.” The licensee made and implemented a change to its emergency plan emergency action level (EAL) scheme on April 16, 2003, which appeared to decreased the effectiveness of the emergency plan without prior NRC approval.

Specifically, the licensee changed the EAL to remove the condition, “release of secondary coolant from the associated steam generator to the environment is occurring,” from the Fission Product Barrier Matrix EAL for a loss of containment barrier due to a steam generator secondary side release. The revised emergency action level, “secondary line break outside containment results in release (greater than 30 minutes) to the environment,” added a non-conservative 30 minutes before meeting this emergency action level. There is a potential that a release condition could have existed which would not have been declared, resulting in either no action or delayed action by off-site authorities when measures to protect the health and safety of the public were warranted. In a previous 1995 correspondence between the NRC and the licensee concerning a proposal to revise the licensee’s EALs, the licensee proposed to implement a similar change to its EALs; however, the NRC specifically provided a written response to the licensee which indicated that a revision to the EAL which included a 30 minute criteria was unacceptable.

The apparent violation was considered to be more than minor because the licensee made changes to the emergency plan and procedures that decreased the effectiveness of the plan without prior approval of the NRC. Because this apparent violation affected the NRC’s ability to perform its regulatory function, it was evaluated using the traditional enforcement process. There were no actual emergency events associated with this EAL during the time the change was in effect; however, the failure of the licensee to meet an emergency planning standard involving assessment does have regulatory significance.

Notice of Violation Issued October 6, 2006, ML062790406.

The VIO was opened in NRC Inspection Report 05000315/316/2006502. Apparent violation AV 0500315/316/2006501-01 is updated to VIO 05000315/316/2006502-01 (Failure to Provide Complete and Accurate

Information to the NRC Which Impacted a Licensing Decision).
Inspection Report# : 2007502 *(pdf)*
Inspection Report# : 2006502 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : December 07, 2007