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

Initiating Events

Significance: G Jun 29, 2006
Identified By: NRC
Item Type: FIN Finding
Inadequate Preventive/Corrective Maintenance on Turbine Building Sump Overflow Check Valve 12-DR-129
The inspectors identified a finding of very low safety significance. The licensee failed to perform adequate preventive and corrective maintenance on Turbine Building sump overflow check valve 12-DR-129. As a result, the valve was found in a significantly degraded condition such that it would not function to mitigate the consequences of a design basis seiche event. No violation of regulatory requirements was identified. Immediate corrective actions to address this finding included replacing the check valve and implementing a preventive maintenance activity to ensure that it would function.

This finding was of more than minor significance because it was associated with the Protection Against External Factors attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since inadequate preventive and corrective maintenance led to the significantly degraded condition of 12-DR-129. Although this issue affected the ability of the check valve to mitigate the consequences of a design basis seiche event, the Regional Senior Reactor Analyst determined that this finding was of very low safety significance during a Phase 3 Significance Determination Process evaluation because considering the seiche initiating event frequency, the change in core damage frequency for this finding was calculated to be well below 1.0E-6. This finding affected the cross-cutting area of problem identification and resolution because the licensee failed to identify and correct the degraded valve condition. Corrective actions that were taken were not timely, were not commensurate with the significance of the issue, and early corrective actions were ineffective.
Inspection Report# : 2006004 *(pdf)*

Mitigating Systems

Significance: G 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)*

Significance: Jun 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Potential External and Internal Flooding Impact on Safe Shutdown Equipment in the Lake Screen House
The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criteria III, "Design Control." The licensee failed to correctly translate the design basis into specifications for

the essential service water (ESW) system by ensuring that ESW system components in the Lake Screen House would be protected to the 595' elevation as described in Section 10.6 of the Updated Final Safety Analysis Report, in the event of flooding due to a design basis seiche event. The licensee was evaluating corrective actions for this issue at the end of the inspection period. No immediate actions were necessary due to the present low lake level.

This finding was of more than minor significance because it was associated with the Design Control 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 since the failure to maintain adequate design control for the affected ESW system components in the Lake Screen House could possibly have resulted in damage to safe shutdown plant equipment during a design basis seiche event. The finding was of very low safety significance because it was a design or qualification deficiency confirmed not to result in loss of operability.
Inspection Report# : 2006004 *(pdf)*

Significance: G Jun 29, 2006
Identified By: NRC
Item Type: FIN Finding
Inadequate Functionality Evaluation for Degraded Check Valve Condition
The inspectors identified a finding of very low safety significance. The licensee did not adequately evaluate the functionality of Turbine Building sump overflow check valve 12-DR-129, while the valve was in a significantly degraded condition such that it would not function to mitigate the consequences of a design basis seiche event. No violation of regulatory requirements was identified. Immediate corrective actions to address this finding included a detailed calculation to determine the potential for flooding in the emergency diesel generator (EDG) rooms to support a past operability evaluation for the EDGs.

This finding was of more than minor significance because if left uncorrected, the failure to properly evaluate the functionality of equipment important to safety could result in incorrectly concluding that the equipment was functional. The inspectors determined that this finding was related to the Protection Against External Factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Consistent with the Phase 3 Significance Determination Process evaluation performed in Section 1R06.b.2, this finding was determined to be of very low safety significance. This finding affected the cross-cutting area of problem identification and resolution because the licensee did not apply appropriate rigor and detail to its evaluation of the non-functional check valve; and as a result, the potential impact on safe shutdown equipment was not evaluated and timely corrective actions were not taken.
Inspection Report# : 2006004 *(pdf)*

Significance: SL-IV Jun 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Inappropriate Deletion of Technical Requirements Manual Sections
The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee’s failure to perform a safety evaluation for the deletion of four sections of the Technical Requirements Manual (TRM). Specifically, the licensee deleted Sections 8.4.7, Tavg Lower Limit, 8.6.1, Ice Bed Temperature Monitoring System, and 8.6.2, Inlet Door Position Monitoring System, and 8.3.7, Post Accident Monitoring (PAM) Instrumentation, Table 8.3.7-1 without evaluating these changes per the requirements of 10 CFR 50.59.

Because the issue potentially impacted the NRC’s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the Updated Final Safety Analysis Report change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC’s inspection manual chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and answered “no” to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, even though these TRM sections along with their associated surveillance requirements were deleted, the licensee was able to show that all deleted surveillance requirements had been performed satisfactorily and within their prescribed frequency in spite of the deletion. This issue was entered into the licensee’s corrective action program.
Inspection Report# : 2006009 *(pdf)*

Significance: SL-IV Jun 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform 10 CFR 50.59 Evaluation for Modification to the 2-East Centrifugal Charging Pump
The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for the modification of the 2-East Centrifugal Charging Pump (CCP). Specifically, the licensee performed modifications to the 2-East Centrifugal Charging Pump that required more restrictive frequency requirements to be established than were already in the Technical Specifications. Had a 10 CFR 50.59 evaluation been performed, as required, the evaluation should have shown that a change to the Technical Specifications (TS) was required so that the new required frequency value could be incorporated into the applicable TS Surveillance Requirements. This issue was entered into the licensee's corrective action program.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the modification of the 2-East CCP would not have ultimately required NRC approval. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening for the mitigating systems cornerstone and determined that the finding was of very low safety significance because they were able to answer "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the 10 CFR 50.59 evaluation, and ultimately the required license amendment, were not performed as required, administrative controls were put into place after the modification was performed such that the CCP would always be able to perform its function.
Inspection Report# : 2006009 *(pdf)*

Significance: G Jun 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Non-Conservative Verification of Containment Average Air Temperature
The inspections identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance. Specifically, verification of containment lower compartment average temperature per Surveillance Requirement 3.6.5.2 was being performed using temperature readings that were not representative (and non-conservative) of the true average temperature in the lower containment. The issue was entered into the licensee's corrective action program.

The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the methodology for determining lower containment average temperature was non-conservative and did not account for the heightened temperatures that were experienced in the Steam Generator (SG) Enclosure Rooms. Had average temperature been above the TS limits, temperatures during a Design Basis Accident could have exceeded the ratings of safety related mitigating equipment thereby challenging the functionality of the equipment. This finding was of very low safety significance, because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after performing a calculation that included the SG Enclosure Rooms, the licensee determined that under worst case historical conditions, average air temperature was 119.5 degrees which was still less than the TS requirement of 120 degrees F.
Inspection Report# : 2006009 *(pdf)*

Barrier Integrity

Significance: G 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: G 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# : 2006502 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Aug 18, 2006
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution
The team identified that the licensee was effective at identifying problems and incorporating them into the corrective action program. The licensee's effectiveness at problem identification was evidenced by the relatively few deficiencies identified by the team that had not been previously identified by the licensee during the review period. In general, the licensee was effectively prioritizing, evaluating and resolving problems. However, the inspectors found several examples where the documentation of an issue did not clearly indicate whether it had been properly evaluated, what the status of the corrective actions were, or whether it had been effectively resolved.

Operating experience usage was also effective, but the team found several examples where operating experience, primarily issued by the NRC, was not screened by the station or was not properly evaluated by the assigned department.

Licensee audit and self-assessments were generally through, probing, and made good use of outside resources to maintain independence. On the basis of interviews conducted during this inspection, workers at the site felt free to input safety findings into the corrective action program.
Inspection Report# : 2006008 *(pdf)*

Last modified : June 01, 2007