Crystal River 3
2Q/2006 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: FIN Finding
Inadequate procedure guidance resulted a Loss of Condensate flow and a Manual Reactor Trip
A self-revealing finding was identified for failure to provide adequate condensate system procedural guidance to preclude condensate pump operation at critical speed. As a result, prolonged operation at critical speed caused the condensate pump to fail and subsequently, the reactor was manually tripped in anticipation of a loss of the normal heat sink. The licensee entered this issue into the licensee's corrective action program as nuclear condition reports (NCRs) 174440 and 174442.

This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. This finding also affected the equipment reliability attribute of the Mitigating Systems Cornerstone objective and resulted in a loss of the normal heat sink. Because two Cornerstones were affected, a Phase 2 analysis was required. The consequences of the finding were assessed through the Significance Determination Process (SDP) Phase 2, and although the likelihood of a unit trip was increased and resulted in a loss of the normal heat sink, the exposure time for this condition was less than 3 days and all other mitigation capabilities described on the Phase 2, SDP worksheet for transient (reactor trip) core damage sequences were maintained. Consequently, the finding was determined to be of very low safety significance (Green). The finding was associated with non-safety related equipment and therefore, no violation of regulatory requirements occurred (Section 4OA3.2).

Inspection Report# : 2005005*(pdf)*

Mitigating Systems

Significance: G Jun 23, 2006
Identified By: NRC
Item Type: FIN Finding
Failure to Conduct an Extent of Condition Review after Three Motor Operated Valves Were Found with Their Pinion Gears Installed Incorrectly
A Green finding was identified by the inspectors for failure to conduct an extent of condition evaluation when three motor operated valves (MOVs) which were thought to not be susceptible to incorrect pinion gear installation were found with their pinion gears installed backwards.

This finding is more than minor because it affected the equipment performance attribute of the mitigating system cornerstone and affected the cornerstone objective of ensuring reliability of a mitigating system. Using NRC Manual Chapter 0609, “Significance Determination Process, “ Appendix A, Phase 1, this finding was determined to be of very low significance (Green), because the finding has not resulted in a loss of safety function and was not screened as potentially risk significant due to external events. The primary cause of the finding was related to the cross cutting area of Problem Identification and Resolution, in that station personnel failed to determine the need for additional MOV inspections when three MOVs which were initially thought to not be susceptible to incorrect pinion gear installation were found with reversed pinion gears, one of which was also discovered with an improperly staked pinion key.

Inspection Report# : 2006009*(pdf)*

Significance: G Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to properly assess and correct condition of water in the 1A diesel fuel tank
An NRC identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly assess and correct a long standing issue associated with minor amounts of water intrusion into the 1A Emergency Diesel Generator (EGDG) diesel fuel tank (DFT). As a result, the DFT remained susceptible to water intrusion during a postulated peak high tide associated with a probable maximum hurricane which could have affected the operability of the 1A EGDG. The licensee entered the issue into the corrective action program and is performing a root cause evaluation to determine short and long term corrective actions.

This finding is more than minor because it affected the protection against external factors attribute of the mitigating system cornerstone and affected the cornerstone objective of ensuring availability of a mitigating system. During a design basis flood event, enough water could have entered the DFT through a loose cap adapter connection to render the 1A EGDG inoperable. Using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1, this finding was determined to require a Phase 3 analysis since the finding screened as potentially risk significant due to a flooding initiating event. A Regional Senior Reactor Analyst performed the Phase 3 evaluation and determined that the finding was of very low safety significance. This finding also involved a cross cutting aspect of Problem Identification and Resolution, because station personnel missed several opportunities to properly assess and correct this degraded condition.(Section 1R06)

Inspection Report# : 2005004*(pdf)*

Significance: W Sep 07, 2005
Identified By: NRC
Item Type: VIO Violation
Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action
10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G, Fire Protection of Safe Shutdown Capability.

Section III.G.2 states that, except as provided for in Section III.G.3, where cables or equipment (including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground) of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating (Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier.);

b. separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards (In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.); or

c. enclosure of cable, equipment, and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. (In addition, fire detectors and an automatic fire suppression system
shall be installed in the fire area.)

Contrary to the above, on January 26, 2005, the licensee failed to ensure that one of the redundant trains of systems necessary to achieve and maintain hot shutdown conditions would be free of fire damage via one of the three means specified in 10 CFR Part 50, Appendix R, Section III.G.2. Specifically, cables for the electrical protection and metering circuit located in the 3A 4160-V engineered safeguards (ES) switchgear room were vulnerable to fire damage that could disable both the 3A 4160-V ES switchgear and the redundant train 3B 4160-V ES switchgear resulting in a loss of all safety-related alternating current power.

This violation is associated with a White Significance Determination Process finding for Unit 3 in the mitigating systems cornerstone.

Inspection Report# : 2005011*(pdf)*

Barrier Integrity

Significance: G Sep 26, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Conduct Adequate Corrective Actions During Review of Steam Generator Inspection Results During 12R Refueling Outage Inspections
The inspectors identified a Non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for inadequate corrective action during review of the results of the refueling outage 12 steam generator Tube End Crack inspections. As a result, Crystal River 3 operated with the calculated leakage exceeding the Technical Specification leakage limit. The licensee entered this condition into their corrective action program.

The finding is more than minor because it was associated with steam generator tube integrity and affected the barrier integrity cornerstone, and if left uncorrected, a more significant safety concern could occur if appropriate corrective actions were not applied to unexpected results found during steam generator inspection activities. This finding represents a cross cutting aspect of problem identification and resolution. The finding was of very low safety significance because it did not result in loss of structural integrity of the steam generators, the small increase in estimated leak rate under main steam line break accident scenarios would not have any significant effect on core damage frequency or large early release frequency, and the contained location of flaws in the tubes makes it impossible to cause spontaneous tube ruptures.

Inspection Report# : 2005009*(pdf)*

Significance: SL-IV Sep 26, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Completeness and Accuracy of Information Provided to the NRC Concerning Steam Generator Inspection Results
The inspectors identified a Non-cited violation (NCV) of 10 CFR 50.9, Completeness and Accuracy of Information, for several examples of inaccuracies and incomplete information in required reports and correspondence. The licensee entered this condition into their corrective action program.

This violation was assessed using traditional enforcement because it impacted the regulatory process. The issue is more than minor because the NRC relies on complete and accurate information to reach conclusions concerning the allowable time between steam generator inspections. It was determined to be a Severity Level IV violation because it was not willful, the technical issue associated with the incomplete and inaccurate information was of very low safety significance, and the NRC had not yet made a regulatory decision based on the information.

Inspection Report# : 2005009*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jun 23, 2006
Identified By: NRC
Item Type: FIN Finding
Identification and Resolution of Problems
The team concluded that in general, problems were properly identified, evaluated, prioritized, and corrected within the licensee’s problem identification and resolution program. Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluations for issues classified as significant conditions adverse to quality were comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. One exception was noted concerning corrective action for identified deficiencies with three motor-operated valves.

The processes and procedures of the licensee’s corrective action program (CAP) were generally adequate; thresholds for identifying issues were appropriately low, and in most cases, corrective actions were adequate to address conditions adverse to quality. Nuclear Assessment Section audits and departmental self-assessments were effective in identifying issues and directing attention to areas that needed improvement. Licensee identified weaknesses and issues in self-assessments were appropriately entered into the CAP and addressed. However, the inspectors observed that several lower threshold issues had not been entered into the CAP.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors did not identify any reluctance to report safety concerns.

Inspection Report# : 2006009*(pdf)*

Last modified : August 25, 2006