Crystal River 3
2Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Manual Reactor Trip Due to Group 7 Control Rods Insertion Caused by Inadequately Protected Test Jumper (Section 4OA3.3)
A self-revealing NCV of Improved Technical Specification (ITS) 5.6.1.1.a was identified for the failure to follow the provisions of preventative maintenance procedure PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train. Failure to follow PM-126 caused the failure of the Group 7 control rod programmer during maintenance and resulted in the unexpected insertion of the Group 7 control rods fully into the core. This unexpected insertion of these control rods into the core caused control room operations personnel to manually trip the reactor from 100 percent power. The licensee entered this issue into the corrective action program as NCR 351705.

This finding was determined to be more than minor because it was associated with the initiating events cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting area of Human Performance with a work practices aspect (H.4 (b)). Specifically, the workers failed to follow the preventative maintenance procedure. (Section 4OA3.3)

Inspection Report# : 2009005 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Compensatory Actions When a MCR to CSR Floor/Ceiling Interface Access Hatch Was Open
The inspectors identified a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C.(9), for failure to take compensatory actions when a main control room (MCR) and cable spreading room (CSR) floor/ceiling interface access hatch was open rendering the CSR Halon fire extinguishing system inoperable. Once identified, the licensee initiated nuclear condition report (NCR) 266356 in the corrective action program to address this issue.

The finding is more than minor because it is associated with the protection against external factors attribute, i.e., fire, and degraded the Mitigating Systems cornerstone objective to ensure the availability of systems that respond to initiating events. Specifically, the finding adversely affected the suppression fire extinguishing system capability defense-in-depth element. The inspectors evaluated this finding under NRC
Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process (SDP). The inspectors determined that a Phase 2 SDP was required for this finding because the CSR Halon concentration was highly degraded; a fire could occur due to non-qualified cables or transient combustibles while the hatch between the MCR and CSR was open; a duration factor (exposure time) was between 3 and 30 days; and control room operators evacuated the MCR in response to the fire. However, Phase 2 SDP of IMC 0609 Appendix F does not currently include explicit treatment of fires leading to MCR abandonment, either due to fire in the MCR or due to fires in other

fire areas. Therefore, a Phase 3 SDP evaluation for this type of finding was needed. A Regional Senior Reactor Analyst performed a Phase 3 SDP for this finding and concluded that the finding was of very low safety significance (Green). The major assumptions and the dominant accident sequence were discussed in the 4OA5 analysis section of this report. The inspectors did not identify a cross-cutting aspect associated with this finding because it does not reflect current licensee performance. (Section 4OA5)
Inspection Report# : 2010002 *(pdf)*

Significance: Dec 31, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Follow a Plant Procedure Resulted in an Inoperable HPI System
A self-revealing Non-Cited Violation (NCV) of Improved Technical Specification (ITS) 5.6.1.1.a was identified for the failure to follow a plant procedure which resulted in a loss of a 480 volt engineered safeguards motor control center (ES MCC)-3B1. Concurrent with pre-existing conditions, the high pressure injection (HPI) system was declared inoperable and ITS 3.0.3 was entered for a period of one hour and 24 minutes. The licensee entered this issue into the corrective action program as nuclear condition report (NCR) 333515.

The finding was more than minor since it affected the equipment availability attribute of the mitigating system cornerstone and resulted in ITS 3.0.3 entry for the HPI system being inoperable. The finding was evaluated against NRC Phase 1 Significance Determination Process (SDP) and Phase 2 SDP was required due to a loss safety function of the HPI system. A Regional Senior Reactor Analyst performed a Phase 3 SDP evaluation and concluded this finding was of very low safety significance (Green). The major assumptions of the evaluation were that the HPI function was out of service for exposure period (1 .5 hours) and there would be no recovery of the de-energized motor control center. The dominant accident sequence involved a support system failure of the Emergency Feedwater (EF) Indication and Control System rendering Main Feedwater and automatic control of EF unavailable, operators were unable to manually control EF flow causing its failure and with the HPI function lost due to the performance deficiency, core damage ensued. The inspectors determined the cause of the finding is related to the cross-cutting area of Human performance with a work practices aspect H.4 (c)). Specifically, work scope changes involving safety-related equipment did not receive the appropriate level management oversight resulted in a plant procedural violation. (Section 4OA3.2)

Inspection Report# : 2009005 *(pdf)*

Significance: Sep 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Risk Assessments When Performing Surveillance Testing
The inspectors identified a non-cited violation (NCV) of 10 CFR 50.65(a)(4) for the failure to perform adequate risk assessments associated with a number of surveillance tests. Specifically, it was determined that risk assessments were not being properly performed for equipment that became unavailable as a result of surveillance testing. This condition has existed since implementation of the Equipment out of Service (EOOS) risk assessment software more than 10 years ago. Short term corrective actions include performance of additional peer reviews of upcoming performance and surveillance tests (PTs and SPs) to ensure they are included in the plant risk assessment and a similar independent review by the corporate probabilistic risk assessment staff. Long term corrective actions include: screen all SPs and PTs to evaluate for risk impact; develop a methodology to include risk significant SPs and PTs in the plant risk assessment, either automatically from the work schedule or a manual process; incorporate risk assessment process changes in licensee procedures; and provide additional EOOS training to the plant staff.

Utilizing IMC 0612, Appendix B, Issue Screening, the finding was determined to be more than minor since licensee risk assessments failed to consider risk significant systems and support systems that were unavailable during maintenance. In order to determine the risk significance of this finding, the inspectors selected two recently performed surveillance procedures for two high risk systems that were not included in the licensee’s risk assessment. The SPs selected were decay heat system (DHR) SP-340B, DHP-1A, BSP-1A and Valve Surveillance and emergency feedwater (EFW) system SP-146A, EFIC Monthly Functional Test (During Modes 1, 2, 3). The risk deficit for SP-340B was determined to be less than 1E-6 incremental core damage probability deficit (ICDPD). The risk associated with SP-146A was not quantified since it was determined that the system did not lose its functionality during the SP. Utilizing IMC 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process (SDP), Flow Chart 1, the finding was determined to be of very low safety significance. This finding was not assigned a cross cutting aspect since the issue existed for greater than 10 years and is not indicative of current licensee performance.

Inspection Report# : 2009004 *(pdf)*

Barrier Integrity

Emergency Preparedness

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

Significance: N/A Apr 23, 2010
Identified By: NRC
Item Type: FIN Finding
Problem identification and Resolution (PI&R)
The team concluded that, in general, problems were properly identified, evaluated, prioritized, and corrected. The licensee was effective at identifying problems and entering them into the corrective action program (CAP) for resolution, as evidenced by the relatively few deficiencies identified by external organizations (including the NRC) that had not been previously identified by the licensee, during the review period. Generally, prioritization and evaluation of issues were adequate, formal root cause evaluations for significant problems were adequate, and corrective actions specified for problems were acceptable. Overall, corrective actions developed and implemented for issues were generally effective and implemented in a timely manner.
The team determined that overall, audits and self-assessments were adequate in identifying deficiencies and areas for improvement in the CAP, and appropriate corrective actions were developed to address the issues identified. Operating experience usage was found to be generally acceptable and integrated into the licensee’s processes for performing and managing work, and plant operations. However, the team found examples where operating experience was not adequately evaluated.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors determined that personnel at the site felt free to raise safety concerns to management and use the CAP to resolve those concerns.
Inspection Report# : 2010006 *(pdf)*

Last modified : September 02, 2010