Crystal River 3
3Q/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 Sep 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Flood Calculations did not Reflect Plant Configuration
The inspectors identified a non-cited violation (NCV) of 10 CFR 50 Appendix B, Criterion III, “Design Control,” regarding the licensee’s failure to ensure that the design bases of two components were correctly translated into specifications, drawings, procedures, and instructions. Specifically, licensee personnel failed to ensure that two floor penetration flood barriers (metal sleeves) were of the proper height to prevent water from entering the A train decay heat removal (DHR)/building spray (BS) vault during a design basis internal flooding event. The design basis did not assume any leakage to the vault. The licensee initiated nuclear condition report (NCR) 409263 in the corrective action program to address the issue.

This finding is more than minor because it affects the design control attribute of the mitigating system cornerstone, and affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609, Phase 1 screening worksheet, the inspectors determined that the finding has very low safety significance because it did not result in a loss of any system safety function. The inspectors found that the cause of the finding is not reflective of current performance and therefore, a cross-cutting aspect will not be assigned. (Section 1R06)

Inspection Report# : 2010004 *(pdf)*

Significance: G Sep 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Inoperable Fire Barrier Penetration Seals
The inspectors identified an NCV, with five examples, of Crystal River Unit 3 Operating License Condition 2.C (9), fire protection program. The NCV was associated with one inoperable fire penetration seal in the ceiling of the B train decay heat and building spray pump vault and four inoperable fire penetration seals associated with the main steam piping in the wall between the intermediate building and the turbine building. Once identified, the licensee initiated an hourly watch and entered the issue in the corrective action program as nuclear condition reports 369096, 406215, and 418755.

The finding is more than minor because if left uncorrected, the fire seals could experience further degradation and potentially lead to a more significant safety concern.
Using NRC IMC 0609, Appendix F, Fire Protection Significance Determination Process, the inspectors assessed the defense-in-depth (DID) element of each fire barrier degradation in the fire confinement category. One penetration was determined to have a low degradation rating and was determined to be of very low safety significance. The other four degraded penetrations were determined to have moderate degradation and were screened to be very low safety significance due to having non-degraded automatic full area water-based fire suppression system available in the exposing fire area. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution with an evaluation aspect (P.1.(c)). Specifically, the licensee had the opportunity to evaluate the need to change the frequency of main steam line fire penetration inspections after finding degradation of main steam piping penetrations in 2007. (Section 4OA5)

Inspection Report# : 2010004 *(pdf)*

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: G 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)*

Barrier Integrity

Significance: G Sep 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Submit Production Splices of Swaged Mechanical Splices for the Testing
The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” for the licensee’s failure to establish measures to assure that testing of rebar splices would adhere to the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Specifically, licensee procedures for containment building repairs did not accommodate rebar production splice testing, which was required by Code. As part of their immediate corrective actions, the licensee revised their procedures to include production splice testing and also entered the issue into their corrective action program.

The inspectors determined that the finding was more than minor because it was associated with the human performance attribute of the barrier systems cornerstone and affected the cornerstone objective of ensuring the reliability of containment wall barrier system. Failure to adhere to ASME Code testing requirements can adversely affect assurance that the rebar splices would meet strength requirements as part of the containment barrier. The inspectors completed a Phase 1 screening of the finding using Inspection Manual Chapter 0609, “Significance Determination Process,” Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings” and determined that the performance deficiency represented a finding of very low safety significance (Green). Specifically, the finding did not result in the actual loss of function of the Unit 3 Containment Wall. This finding has a cross-cutting aspect in the area of Human Performance under the “Effectiveness Reviews” aspect of the “Decision-Making” component because the licensee failed to validate assumptions used as a basis for their decision to pursue an alternative testing plan. [H.1(b)] (Section 4OA5)

Inspection Report# : 2010004 *(pdf)*

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)*