Crystal River 3

Initiating Events

Significance: G Dec 29, 2001
Identified By: NRC
Item Type: FIN Finding
Worker Fatigue
Green. The inspectors identified that the licensee did not consider worker fatigue in the licensee investigation of a potential loss of 4160 volt bus that involved worker performance issues. This finding is more than a minor because it was viewed as a precursor to a significant event (loss of decay heat removal). The finding is considered to be of very low safety significance because no actual loss of equipment occurred. (Section 1R20)
Inspection Report# : 2001004*(pdf)*

Mitigating Systems

Significance: G Dec 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to Follow Procedures During Electrical Maintenance on the Engineered Safeguards Bus
10 CFR Part 50, Appendix B, Criterion V, states that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to this requirement, on October 9, 2001, during maintenance on 4160 volt engineered safeguards (ES) bus 3A using Work Request 365187: 1) Although the work request stated "Check Bus for Voltage before starting work", dead bus checks were not done; 2) Although licensee Administrative Instruction AI-610, Electrical Safety, required a maintenance risk assessment be performed on all work on energized equipment with the work assessed as medium or high risk, the work request had not been risk assessed and was classified low risk; and 3) Although licensee Administrative Instruction AI-504, Guidelines for Cold Shutdown and Refueling, stated "Power supplies (for operating safety equipment shall be) controlled by physical barriers with signs" an energized power supply for the operating decay heat removal equipment accessed by a worker was not controlled by a physical barrier with a sign. This was identified in the licensee's corrective action program as CR-42306 (Green).
Inspection Report# : 2001004*(pdf)*

Significance: G Dec 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Main Steam Safety Valve Setpoints Outside Required Tolerance Longer than Allowed by Technical Specifications
Technical Specification 3.7.1.A requires that all main steam line code safety valves shall be operable with lift settings as specified in Table 3.7.1-1. If one or more main steam line code safety valves are inoperable, actions must be taken in accordance with Action A of Technical Specification 3.7.1. Contrary to the above, as of September 26, 2001 and September 29, 1999, main steam line safety valves did not have lift settings in accordance with Table 3.7.1-1, and the requirements of Action A of Technical Specification 3.7.1 were not met. This condition was identified by the licensee and documented in Nuclear Condition Report NCR 48648. This condition was reported in LER 50-332/01-002 (Green).
Inspection Report# : 2001004*(pdf)*

Significance: G Sep 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Installation Error Results in Containment Isolation Valve Inoperable Longer than Allowed by Technical Specifications
Technical Specification 3.6.3 Condition C requires that a containment penetration flow path be isolated within 4 hours, if the associated isolation valve is not operable. Contrary to this from May 13 to 14, a feedwater check valve (FWV-46) was not operable, and the penetration was not isolated. This was identified in the licensee's corrective action program as CR-42306. This finding is only of very low significance because it only affects the barrier integrity cornerstone and all other mitigating systems were functional.

(Green)
Inspection Report# : 2001003*(pdf)*

Significance: Mar 31, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to Implement Fire Protection Requirements for Inoperable Fire Damper.
Crystal River 3 Operating License Requirement 2.C.(9) requires that FPC shall implement and maintain in effect all provisions of the approved fire protection program. Table 6.7a of the Fire Protection Plan, requires that when a fire barrier penetration is not functional, either establish a continuous fire watch on at least one side of the barrier, or verify the operability of fire detectors on one side of the barrier and establish an hourly fire watch patrol. Contrary to the above, for various times prior to January 3, 2001, the air return fire barrier damper for the B engineered safeguards 4160 volt switchgear room was not functional and neither fire watch provision was met. This issue was described in the licensee corrective action program as PC 01-0012 and is being treated as a Non-cited violation.
Inspection Report# : 2000005*(pdf)*

Significance: Dec 30, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to Implement Fire Protection Plan Requirements When Two Cable Spreading Room Fire Dampers Were Not Operable.
Crystal River 3 Operating License Requirement 2.C.(9) requires that all provisions of the approved fire protection program be implemented. Table 6.7a of the Fire Protection Plan requires that when a fire barrier penetration is not functional, the licensee shall either establish a continuous fire watch on at least one side of the barrier, or verify the operability of fire detectors on one side of the barrier and establish an hourly fire watch patrol. For various periods of time from February 1999 to October 10, 2000, both exhaust fire barrier dampers (AHFD-47 and 83) for the cable spreading room were not functional and the fire watch provisions of the Fire Protection Plan were not met. The violation is in the licensee's corrective action program as Precursor Card 00-2918. (Section 4OA7)
Inspection Report# : 2000004*(pdf)*

Significance: Sep 30, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to Meet Technical Specification Requirements for Pressurizer Heaters
Green. A Non-Cited Violation of Technical Specification 3.4.8, Condition B was identified because requirements for electrical power supplies to the pressurizer heaters were not met in one instance. Breakers supplying one train of emergency power to the pressurizer heaters were removed from service for greater than the allowed period. The finding was determined to be of very low safety significance. Florida Power Corporation recently identified that procedures did not incorporate the power supply technical specification requirements and subsequently identified this single instance of noncompliance through a detailed review. The intent of the requirements is to ensure that the reactor coolant system is maintained subcooled with natural circulation flow under specific plant conditions. Although the pressurizer heaters support pressure control during natural circulation, there were other alternative methods available to maintain pressure. These methods are proceduralized and addressed in operator training. (Section 4OA3)
Inspection Report# : 2000003*(pdf)*

Barrier Integrity

Significance: N/A Aug 11, 2001
Identified By: NRC
Item Type: FIN Finding
Reactor Coolant System Leakage
This supplemental inspection was performed by the NRC to assess Florida Power Corporation's activities associated with increased reactor coolant system leakage. The leakage exceeded the NRC's Reactor Oversight Process White performance indicator threshold in May, 2001. The White indicator threshold is set at 50 percent of the leakage allowed by Technical Specifications and corresponds to a performance level that may result in increased NRC oversight. Using Inspection Procedure 95001, the inspector determined that the licensee's root cause evaluation for the reactor coolant leakage was acceptable and that the licensee had taken appropriate corrective actions. The source of the leakage was seal ring leakage from a valve located between the reactor and the decay heat removal system. The licensee stated that the most likely cause of the valve leakage, initially identified in April 2000, was uneven seating forces around the pressure retaining valve seal ring. The uneven forces had likely resulted from either a 1999 valve

rebuild or reactor coolant system pressure cycles associated with plant shutdown and subsequent heatup. A plant shutdown and restart in May 2001 was associated with a distinct increase in the leakage that caused the performance indicator to cross the White threshold. The leakage exceeded the White performance indicator threshold for about 24 hours during which time, the reactor remained shutdown. Prior to restarting the reactor following the leakage increase, the licensee installed a canopy over the top of the valve, moving the pressure retaining surface from the seal ring to the canopy. The modification stopped the leakage and the performance indicator returned to the Green performance level.
Inspection Report# : 2001007*(pdf)*

Significance: N/A Nov 01, 2000
Identified By: NRC
Item Type: FIN Finding
Reactor Coolant System Leakage
This supplemental inspection was performed to assess Florida Power Corporation's activities associated with increased reactor coolant system leakage. The leakage exceeded the NRC's Reactor Oversight Program White performance indicator threshold in August, 2000. The leakage remained above the threshold until September 9, 2000, when the plant was shutdown for repairs. This White indicator threshold is set at 50 percent of the leakage allowed by Technical Specifications and corresponds to a performance level that may result in increased NRC oversight. Using Inspection Procedure 95001, the inspector found that the licensee's root cause analysis for the leakage was acceptable and that the licensee had taken or planned appropriate corrective actions. The primary contributor to the leakage was identified as seat leakage thorough a pressurizer safety valve. Leakage from the pressure seal on a decay heat system valve also contributed to the overall reactor coolant system leakage. The licensee replaced the safety valve and conducted leak sealant repairs on the decay heat valve. These actions significantly reduced the reactor coolant system leakage and returned the performance indicator to the Green performance level. The licensee subsequently identified the root cause of the safety valve seat leakage and developed appropriate corrective actions. The licensee plans to complete permanent repairs to the decay heat valve at the next available opportunity.
Inspection Report# : 2000007*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Jul 01, 2000
Identified By: NRC
Item Type: FIN Finding
Radiological Collective Dose Expenditure Estimates Not Revised.
Collective dose expenditures for three high dose rate/dose evolutions conducted during the October 1999 Refueling Outage exceeded their original dose expenditure estimates by more than 50 percent. For steam generator tube maintenance activities, actual dose expenditures exceeded both the original and revised dose projections by more than 50 percent. For eddy current testing and scaffolding activities, revisions to the dose estimates were not conducted and documented until after the original dose expenditure estimates were exceeded. Differences between the original and revised estimates resulted from elevated dose rates, expanded job scope, and/or worker performance. Since the tasks did not result in any individual doses exceeding 10 CFR Part 20, Subpart C, Occupational Dose Limits, this finding was determined to be of very low safety significance. (Section 2OS2).
Inspection Report# : 2000002*(pdf)*

Significance: G Jul 01, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
High Radiation Area Controls Not Fully Effective.
On October 23, 1999, Health Physics (HP) technicians providing high radiation area job coverage failed to provide positive controls in accordance with Improved Technical Specification 5.8.1.c, for two contract workers performing leadscrew cleaning and inspection activities under Radiation Work Permit 99-0146. The two workers received cumulative doses of 330 and 550 millirem which exceeded the 250 millirem (mrem) cumulative dose expected for the task. Since there was no substantial potential for overexposure to occur based on the expected job duration (1 to 2 hours), and the maximum general area dose rates (300 mrem per hour), this finding was determined to be of very low safety significance. This finding was identified as a Non-Cited Violation (NCV) for failure to provide continuous health physics coverage required by Improved Technical Specification 5.8.1.c for work conducted in a High Radiation Area (Section 2OS1.2).
Inspection Report# : 2000002*(pdf)*

Public Radiation Safety

Physical Protection

Significance: G Mar 31, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to ensure that individuals provide identification prior to being granted unescorted access.
Green. A Non-cited violation of a license condition and procedural requirements was identified when two NRC inspectors were granted unescorted access to Crystal River 3 without being required to produce a valid picture identification. Provisions in the Crystal River Physical Security Plan and the requirements of Security Procedure SEC-NGGC-2101 for obtaining identification information prior to granting access were not met. The finding was of very low safety significance because, although the identification information was not verified as required prior to access, the individuals granted access met all requirements for authorization for unescorted access. (Section 3PP2)
Inspection Report# : 2000005*(pdf)*

Miscellaneous

Significance: N/A Dec 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to Report a Condition Prohibited by Technical Specifications
10 CFR 50.73 (a)(2)(B), requires that any condition prohibited by plant technical specifications shall be reported by the licensee in a Licensee Event Report within 60 days after discovery of the event. Contrary to the above, the licensee determined that two main steam safety valves had setpoints outside of the Technical Specification Table 3.7.1-1 required tolerance, the actions of the technical specification were not taken, and the condition was not reported within 60 days after discovery (September 1999). This condition was identified by the licensee and documented in Nuclear Condition Report NCR 51139 (No Color).
Inspection Report# : 2001004*(pdf)*

Significance: N/A Oct 06, 2000
Identified By: NRC
Item Type: FIN Finding
Corrective Action Program
Overall, the licensee's corrective action program was effective at identifying, evaluating, and correcting problems. The threshold for entering problems into the corrective action program was sufficiently low. Reviews of operating experience information were comprehensive. The priority grading system ensured timely resolution and corrective actions commensurate with safety significance. Corrective action backlog and precursor card evaluation timeliness were well managed. Root cause analyses were thorough. However, issues addressed in NRC inspection findings were not specifically reviewed to ensure adequate corrective actions. Licensee self-assessments and audits were effective in identifying deficiencies in the corrective action program. These deficiencies were entered into the corrective action program and, for the most part, resulted in the implementation of corrective actions. However, numerous Health Physics peer assessment recommendations, although entered in the corrective action program, were closed with inadequate documentation of disposition and corrective actions. A safety conscious work environment was present where employees felt free to raise safety concerns.
Inspection Report# : 2000006*(pdf)*

Significance: G Oct 06, 2000
Identified By: NRC
Item Type: FIN Finding
Cross Cutting Issue - Corrective Action Effectiveness
Green. A finding was identified associated with the depth and effectiveness of the licensee's evaluation and corrective actions for precursor card 99-4142. This precursor card addressed deficiencies involved with rigging of the reactor vessel plenum during reactor assembly. NRC Non-cited Violation 50-302/99-07-01, Reactor Plenum Rigged Improperly, also addressed this issue. The licensee did not fully assess the nature and extent of the issue. Consequently, important causal factors were not identified and corrective

actions to prevent recurrence were not thorough. This issue was determined to have very low safety significance because the licensee adequately addressed the potential adverse impact on equipment prior to reactor startup. The licensee's examination did not identify any damage to the reactor vessel or plenum. This instance of ineffective corrective action was an isolated example and is not considered indicative of the licensee's overall corrective action program. (Section 4OA2.2).
Inspection Report# : 2000006*(pdf)*

Last modified : July 22, 2002