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
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 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 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: G 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: G 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: G 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: 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)*

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)*

Last modified : April 01, 2002

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
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 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 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: 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)*

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)*

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: 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)*

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)*

Last modified : April 01, 2002

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 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)*

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 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 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: G 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: G 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)*

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 : March 29, 2002

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 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: G 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)*

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: G 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)*

Barrier Integrity

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)*

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)*

Emergency Preparedness

Occupational Radiation Safety

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)*

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)*

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 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)*

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)*

Last modified : March 28, 2002

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 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: G 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: G 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)*

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)*

Barrier Integrity

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)*

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)*

Emergency Preparedness

Occupational Radiation Safety

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)*

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)*

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: 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)*

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: 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)*

Last modified : March 28, 2002

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 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: G 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: G 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)*

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 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 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)*

Barrier Integrity

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)*

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)*

Emergency Preparedness

Occupational Radiation Safety

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)*

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)*

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 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)*

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)*

Last modified : March 27, 2002

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 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: G 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: G 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: 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 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 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)*

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)*

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
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 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 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: G 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: G 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: G 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: 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)*

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)*

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: G 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: G 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: G 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

Crystal River 3

Initiating Events

Significance: G Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Corrective actions to address a feedwater transient which occurred on December 15, 2001, had not been implemented.
The inspectors identified that corrective actions to address a feedwater transient which occurred on December 15, 2001, had not been implemented. This issue was more than minor because the feedwater transient required operator intervention in order to stabilize the plant and resulted in cavitation of a feedwater booster pump, which if it had tripped or become damaged, could have resulted in more severe consequences. Therefore, it was important that corrective actions should have been implemented. This finding was determined to be of very low safety significance (Green) by the significance determination process because the impact was limited to a slightly increased likelihood of a plant transient. (Section 4OA2.c)
Inspection Report# : 2002006*(pdf)*

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 Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to properly position a component during clearance (tagging) activities
Technical Specification 5.6.1, Procedures, states that written procedures shall be implemented covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. The regulatory guide list in Appendix A, includes procedures for equipment control (tagging). Licensee procedure OPS-NGGC-1301, Equipment Clearance implements this requirement and states, in step 4.10, that the "Tag Hanger positions components as specified on the Clearance Checklist." Contrary to the above, the tag hanger for Clearance Checklist 35216, on April 4, 2002, failed to position the emergency feedwater pump (EFP-3) fuel rack, in the "tripped" position prior to placing a "Diesel Engine Tripped" tag on the fuel rack. The rack was in the Normal, not tripped position and this was not identified by

the independent second-checker. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report 58819.
Inspection Report# : 2002002*(pdf)*

Significance: Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to complete an accurate risk assessment per 10 CFR 50.65(a)(4)
10 CFR 50.65 (a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, the licensee failed to assess the maintenance risk for all plant maintenance to be performed during the week of February 11, 2002. Specifically, the completed assessment failed to account for raw water pumps within licensee-established risk assessment scope that were concurrently out of service. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report 58911.
Inspection Report# : 2002002*(pdf)*

Significance: 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: 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: 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: 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)*

Significance: 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)*

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 Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Identification and Resolution of Problems Based on the results of the inspection, one finding and several negative observations were identified. The licensee was effective at identifying problems
Based on the results of the inspection, one finding and several negative observations were identified. The licensee was effective at identifying problems at a low threshold and putting them into the corrective action program. Although two issues were identified that the licensee had not entered into the corrective action program, these were considered isolated instances and not indicative of a weakness in this area. Generally, the licensee properly evaluated issues and implemented effective and timely corrective action. Formal root causes for issues classified as significant conditions adverse to quality were especially thorough and detailed. The inspectors identified several examples in which condition reporting evaluations lacked thoroughness or were too narrowly focused, and some corrective actions were not comprehensive or were not implemented as intended. One finding of very low safety significance was identified. The inspectors identified that corrective actions to address a feedwater transient had not been implemented. Licensee audits and self-assessments were effective in identifying deficiencies in the corrective action programs. In addition, audit and assessment findings were consistent with the inspectors' observations. Based on interviews of plant personnel from various departments, personnel indicated that they felt free to input safety issues and conditions adverse to quality into the corrective action and employee concerns programs. A safety conscious work environment was evident at Crystal River.
Inspection Report# : 2002006*(pdf)*

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 : August 29, 2002

Crystal River 3

Initiating Events

Significance: Sep 28, 2002
Identified By: Self Disclosing
Item Type: FIN Finding
Corrective Actions
Green. The licensee's corrective actions for a failed power cable were insufficient to prevent recurrence of a partial loss of offsite power event. The finding was more than minor because it increased the likelihood of a loss of offsite power. The finding was determined to be of very low safety significance by the safety determination process because it did not involve a total loss of offsite power and power remained available for safety equipment. (Section 4OA2)
Inspection Report# : 2002003*(pdf)*

Significance: Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Corrective Actions to Address a Feedwater Transient
Green. The inspectors identified that corrective actions to address a feedwater transient which occurred on December 15, 2001, had not been implemented. This issue was more than minor because the feedwater transient required operator intervention in order to stabilize the plant and resulted in cavitation of a feedwater booster pump, which if it had tripped or become damaged, could have resulted in more severe consequences. Therefore, it was important that corrective actions should have been implemented. This finding was determined to be of very low safety significance (Green) by the significance determination process because the impact was limited to a slightly increased likelihood of a plant transient. (Section 4OA2.c)
Inspection Report# : 2002006*(pdf)*

Significance: 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: Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to complete an accurate risk assessment per 10 CFR 50.65(a)(4)
10 CFR 50.65 (a)(4) requires, in part, that before performing maintenance activities (including but not limited to

surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, the licensee failed to assess the maintenance risk for all plant maintenance to be performed during the week of February 11, 2002. Specifically, the completed assessment failed to account for raw water pumps within licensee-established risk assessment scope that were concurrently out of service. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report 58911.
Inspection Report# : 2002002*(pdf)*

Significance: Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to properly position a component during clearance (tagging) activities
Technical Specification 5.6.1, Procedures, states that written procedures shall be implemented covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. The regulatory guide list in Appendix A, includes procedures for equipment control (tagging). Licensee procedure OPS-NGGC-1301, Equipment Clearance implements this requirement and states, in step 4.10, that the "Tag Hanger positions components as specified on the Clearance Checklist." Contrary to the above, the tag hanger for Clearance Checklist 35216, on April 4, 2002, failed to position the emergency feedwater pump (EFP-3) fuel rack, in the "tripped" position prior to placing a "Diesel Engine Tripped" tag on the fuel rack. The rack was in the Normal, not tripped position and this was not identified by the independent second-checker. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report 58819.
Inspection Report# : 2002002*(pdf)*

Significance: 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: 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: G 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: G 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: G 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 Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Identification and Resolution of Problems
Based on the results of the inspection, one finding and several negative observations were identified. The licensee was effective at identifying problems at a low threshold and putting them into the corrective action program. Although two issues were identified that the licensee had not entered into the corrective action program, these were considered isolated instances and not indicative of a weakness in this area. Generally, the licensee properly evaluated issues and implemented effective and timely corrective action. Formal root causes for issues classified as significant conditions adverse to quality were especially thorough and detailed. The inspectors identified several examples in which condition reporting evaluations lacked thoroughness or were too narrowly focused, and some corrective actions were not comprehensive or were not implemented as intended. One finding of very low safety significance was identified. The inspectors identified that corrective actions to address a feedwater transient had not been implemented. Licensee audits and self-assessments were effective in identifying deficiencies in the corrective action programs. In addition, audit and assessment findings were consistent with the inspectors' observations. Based on interviews of plant personnel from various departments, personnel indicated that they felt free to input safety issues and conditions adverse to quality into the corrective action and employee concerns programs. A safety conscious work environment was evident at Crystal River.
Inspection Report# : 2002006*(pdf)*

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 : December 02, 2002

Crystal River 3

Initiating Events

Significance: Sep 28, 2002
Identified By: Self Disclosing
Item Type: FIN Finding
Corrective Actions
Green. The licensee's corrective actions for a failed power cable were insufficient to prevent recurrence of a partial loss of offsite power event. The finding was more than minor because it increased the likelihood of a loss of offsite power. The finding was determined to be of very low safety significance by the safety determination process because it did not involve a total loss of offsite power and power remained available for safety equipment. (Section 4OA2)
Inspection Report# : 2002003*(pdf)*

Significance: Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Corrective Actions to Address a Feedwater Transient
Green. The inspectors identified that corrective actions to address a feedwater transient which occurred on December 15, 2001, had not been implemented. This issue was more than minor because the feedwater transient required operator intervention in order to stabilize the plant and resulted in cavitation of a feedwater booster pump, which if it had tripped or become damaged, could have resulted in more severe consequences. Therefore, it was important that corrective actions should have been implemented. This finding was determined to be of very low safety significance (Green) by the significance determination process because the impact was limited to a slightly increased likelihood of a plant transient. (Section 4OA2.c)
Inspection Report# : 2002006*(pdf)*

Mitigating Systems

Significance: Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to complete an accurate risk assessment per 10 CFR 50.65(a)(4)
10 CFR 50.65 (a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, the licensee failed to assess the maintenance risk for all plant maintenance to be performed during the week of February 11, 2002. Specifically, the completed assessment failed to account for raw water pumps within licensee-established risk assessment scope that were concurrently out of service. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report 58911.
Inspection Report# : 2002002*(pdf)*

Significance: Jun 29, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to properly position a component during clearance (tagging) activities
Technical Specification 5.6.1, Procedures, states that written procedures shall be implemented covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. The regulatory guide list in Appendix A, includes procedures for equipment control (tagging). Licensee procedure OPS-NGGC-1301, Equipment Clearance implements this requirement and states, in step 4.10, that the "Tag Hanger positions components as specified on the Clearance Checklist." Contrary to the above, the tag hanger for Clearance Checklist 35216, on April 4, 2002, failed to position the emergency feedwater pump (EFP-3) fuel rack, in the "tripped" position prior to placing a "Diesel Engine Tripped" tag on the fuel rack. The rack was in the Normal, not tripped position and this was not identified by the independent second-checker. This is being treated as a Non-Cited Violation. The violation is in the licensee corrective action program as Nuclear Condition Report

58819.
Inspection Report# : 2002002*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Significance: N/A Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Identification and Resolution of Problems
Based on the results of the inspection, one finding and several negative observations were identified. The licensee was effective at identifying problems at a low threshold and putting them into the corrective action program. Although two issues were identified that the licensee had not entered into the corrective action program, these were considered isolated instances and not indicative of a weakness in this area. Generally, the licensee properly evaluated issues and implemented effective and timely corrective action. Formal root causes for issues classified as significant conditions adverse to quality were especially thorough and detailed. The inspectors identified several examples in which condition reporting evaluations lacked thoroughness or were too narrowly focused, and some corrective actions were not comprehensive or were not implemented as intended. One finding of very low safety significance was identified. The inspectors identified that corrective actions to address a feedwater transient had not been implemented. Licensee audits and self-assessments were effective in identifying deficiencies in the corrective action programs. In addition, audit and assessment findings were consistent with the inspectors' observations. Based on interviews of plant personnel from various departments, personnel indicated that they felt free to input safety issues and conditions adverse to quality into the corrective action and employee concerns programs. A safety conscious work environment was evident at Crystal River.
Inspection Report# : 2002006*(pdf)*

Last modified : March 25, 2003

Crystal River 3
1Q/2003 Plant Inspection Findings

Initiating Events

Significance: G Sep 28, 2002
Identified By: Self Disclosing
Item Type: FIN Finding
Corrective Actions
Green. The licensee's corrective actions for a failed power cable were insufficient to prevent recurrence of a partial loss of offsite power event. The finding was more than minor because it increased the likelihood of a loss of offsite power. The finding was determined to be of very low safety significance by the safety determination process because it did not involve a total loss of offsite power and power remained available for safety equipment. (Section 4OA2)
Inspection Report# : 2002003*(pdf)*

Significance: G Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Corrective Actions to Address a Feedwater Transient
Green. The inspectors identified that corrective actions to address a feedwater transient which occurred on December 15, 2001, had not been implemented. This issue was more than minor because the feedwater transient required operator intervention in order to stabilize the plant and resulted in cavitation of a feedwater booster pump, which if it had tripped or become damaged, could have resulted in more severe consequences. Therefore, it was important that corrective actions should have been implemented. This finding was determined to be of very low safety significance (Green) by the significance determination process because the impact was limited to a slightly increased likelihood of a plant transient. (Section 4OA2.c)
Inspection Report# : 2002006*(pdf)*

Mitigating Systems

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Significance: N/A Jun 21, 2002
Identified By: NRC
Item Type: FIN Finding
Identification and Resolution of Problems
Based on the results of the inspection, one finding and several negative observations were identified. The licensee was effective at identifying problems at a low threshold and putting them into the corrective action program. Although two issues were identified that the licensee had not entered into the corrective action program, these were considered isolated instances and not indicative of a weakness in this area. Generally, the licensee properly evaluated issues and implemented effective and timely corrective action. Formal root causes for issues classified as significant conditions adverse to quality were especially thorough and detailed. The inspectors identified several examples in which condition reporting evaluations lacked thoroughness or were too narrowly focused, and some corrective actions were not comprehensive or were not implemented as intended. One finding of very low safety significance was identified. The inspectors identified that corrective actions to address a feedwater transient had not been implemented. Licensee audits and self-assessments were effective in identifying deficiencies in the corrective action programs. In addition, audit and assessment findings were consistent with the inspectors' observations. Based on interviews of plant personnel from various departments, personnel indicated that they felt free to input safety issues and conditions adverse to quality into the corrective action and employee concerns programs. A safety conscious work environment was evident at Crystal River.
Inspection Report# : 2002006*(pdf)*

Last modified : May 30, 2003

Crystal River 3
2Q/2003 Plant Inspection Findings

Initiating Events

Significance: Sep 28, 2002
Identified By: Self Disclosing
Item Type: FIN Finding
Corrective Actions
Green. The licensee's corrective actions for a failed power cable were insufficient to prevent recurrence of a partial loss of offsite power event. The finding was more than minor because it increased the likelihood of a loss of offsite power. The finding was determined to be of very low safety significance by the safety determination process because it did not involve a total loss of offsite power and power remained available for safety equipment. (Section 4OA2)
Inspection Report# : 2002003*(pdf)*

Mitigating Systems

Significance: Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Implement Inservice Testing Program Requirements (Section 1R22).
A finding was identified for failure to implement increased frequency testing of a safety-related pump, after the pump differential pressure was found in the Alert range of the ASME Code, Section XI test on December 2, 2002. When tested on May 22, 2003, the pump was found in the Action range and was declared inoperable. A non-cited violation of Technical Specification 5.6.2 was identified. The finding is greater than minor because an engineering evaluation was required to assure that accident analysis requirements were met during the subsequent period of operation with differential pressure below the design minimum value. If the finding had not been corrected, pump performance could have resulted in the safety system not being capable of performing its design function to remove residual heat following an accident. The finding is of very low safety significance because the maximum period of operation below the design minimum differential pressure was of short duration and redundancy existed that assured the safety function remained available. (Section 1R22)
Inspection Report# : 2003004*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Last modified : September 04, 2003

Crystal River 3
3Q/2003 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Protect One Train of Safe Shutdown Equipment From Fire Damage
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix R, Section III.G.2, Fire Protection of Safe Shutdown Capability, for failure to protect certain electrical cables for safe shutdown equipment from fire damage in three fire areas. The licensee has corrected related identified procedural deficiencies and plans to resolve the noncompliance with cable protection through licensing correspondence with the NRC.

This finding is greater than minor safety significance because it involved a lack of required fire barriers for equipment relied upon for safe shutdown following a fire and because it affected the objectives of the Mitigating Systems Cornerstone of Reactor Safety. It affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The finding is of very low safety significance because licensee's proceduralized manual actions are reasonably accomplishable and training would have enabled operators to maintain the makeup function sufficiently to maintain reactor coolant system process variables within acceptable ranges. Therefore, the inspectors identified this issue as a Green finding as described in Inspection Procedure 71111.05, Fire Protection. (Section 4OA5)
Inspection Report# : 2003005*(pdf)*

Significance: G Sep 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Maintain Two Operable Control Complex Cooling Trains
A self-revealing non-cited violation of Crystal River 3 Technical Specification 3.7.18 was identified. Following Train B chiller maintenance on December 19, 2002, and Train A chiller maintenance on February 25, 2003, neither train of control complex cooling was operable because control complex chiller motor overload relays had been improperly set below their design values. The problem was identified on June 11, 2003, when both chiller motors tripped on overload current, when an overload current condition had not occurred.

The self-revealing finding is greater than minor safety significance because it resulted in a loss of the control complex cooling safety function and affected the availability and reliability of the Mitigating Systems Cornerstone of Reactor Safety that is used to mitigate events. The finding is of very low safety significance because the alternate non-safety Appendix R cooling system and feedwater pump (FWP-7) were available to mitigate transients involving systems that could be affected by the loss of cooling. (Section 4OA3)

Inspection Report# : 2003005*(pdf)*

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Implement Inservice Testing Program Requirements (Section 1R22).
A finding was identified for failure to implement increased frequency testing of a safety-related pump, after the pump differential pressure was found in the Alert range of the ASME Code, Section XI test on December 2, 2002. When tested on May 22, 2003, the pump was found in the Action range and was declared inoperable.

A non-cited violation of Technical Specification 5.6.2 was identified. The finding is greater than minor because an engineering evaluation was required to assure that accident analysis requirements were met during the subsequent period of operation with differential pressure below the design minimum value. If the finding had not been corrected, pump performance could have resulted in the safety system not being capable of performing its design function to remove residual heat following an accident. The finding is of very low safety significance because the maximum period of operation below the design minimum differential pressure was of short duration and redundancy existed that assured the safety function remained available. (Section 1R22)
Inspection Report# : 2003004*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Last modified : December 01, 2003

Crystal River 3
4Q/2003 Plant Inspection Findings

Initiating Events

Significance: G Dec 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Perform the Magnetic Particle Calibration
The inspector identified a non-cited violation of Technical Specification 5.6.1.1 for failure to follow procedural requirements involving incorrect calibration of a magnetic particle testing (MT) yoke. This finding could have inhibited the identification of indications or flaws on American Society of Mechanical Engineers (ASME) Class 2 Safety-Related Feed Water to Once Through Steam Generator (OTSG) "A" piping.

This finding is more than minor because if left uncorrected, it could result in a more significant safety concern. Failure to correctly perform the calibration could reduce the ability to discover indications or flaws which could lead to pipe breaks. The issue was determined to be of very low safety significance because the likelihood of a loss of coolant accident (LOCA) initiator was not affected, the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have available, and the finding did not increase the likelihood of a fire or flood. (Section 1R08)
Inspection Report# : 2003006*(pdf)*

Mitigating Systems

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Protect One Train of Safe Shutdown Equipment From Fire Damage
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix R, Section III.G.2, Fire Protection of Safe Shutdown Capability, for failure to protect certain electrical cables for safe shutdown equipment from fire damage in three fire areas. The licensee has corrected related identified procedural deficiencies and plans to resolve the noncompliance with cable protection through licensing correspondence with the NRC.

This finding is greater than minor safety significance because it involved a lack of required fire barriers for equipment relied upon for safe shutdown following a fire and because it affected the objectives of the Mitigating Systems Cornerstone of Reactor Safety. It affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The finding is of very low safety significance because licensee's proceduralized manual actions are reasonably accomplishable and training would have enabled operators to maintain the makeup function sufficiently to maintain reactor coolant system process variables within acceptable ranges. Therefore, the inspectors identified this issue as a Green finding as described in Inspection Procedure 71111.05, Fire Protection. (Section 4OA5)
Inspection Report# : 2003005*(pdf)*

Significance: G Sep 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Maintain Two Operable Control Complex Cooling Trains
A self-revealing non-cited violation of Crystal River 3 Technical Specification 3.7.18 was identified. Following Train B chiller maintenance on December 19, 2002, and Train A chiller maintenance on February 25, 2003, neither train of control complex cooling was operable because control complex chiller motor overload relays had been improperly set below their design values. The problem was identified on June 11, 2003, when both chiller motors tripped on overload current, when an overload current condition had not occurred.

The self-revealing finding is greater than minor safety significance because it resulted in a loss of the control complex cooling safety function and affected the availability and reliability of the Mitigating Systems Cornerstone of Reactor Safety that is used to mitigate events. The finding is of very low safety significance because the alternate non-safety Appendix R cooling system and feedwater pump (FWP-7) were available to mitigate transients involving systems that could be affected by the loss of cooling. (Section 4OA3)

Inspection Report# : 2003005*(pdf)*

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Implement Inservice Testing Program Requirements (Section 1R22).
A finding was identified for failure to implement increased frequency testing of a safety-related pump, after the pump differential pressure was found in the Alert range of the ASME Code, Section XI test on December 2, 2002. When tested on May 22, 2003, the pump was found in the Action range and was declared inoperable.

A non-cited violation of Technical Specification 5.6.2 was identified. The finding is greater than minor because an engineering evaluation was required to assure that accident analysis requirements were met during the subsequent period of operation with differential pressure below the design minimum value. If the finding had not been corrected, pump performance could have resulted in the safety system not being capable of performing its design function to remove residual heat following an accident. The finding is of very low safety significance because the maximum period of operation below the design minimum differential pressure was of short duration and redundancy existed that assured the safety function remained available. (Section 1R22)
Inspection Report# : 2003004*(pdf)*

Barrier Integrity

Significance: G Dec 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles
A self-revealing non-cited violation of Technical Specification 3.4.12.a was identified. Small cracks in the pressurizer upper level instrument tap nozzles resulted in pressure boundary leakage since late 2000.

The finding was greater than minor because the breach in the reactor coolant system (RCS) affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. However, the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzle. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. (Section 4OA3.3)

Inspection Report# : 2003006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Last modified : March 02, 2004

Crystal River 3
1Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Mar 27, 2004
Identified By: Self Disclosing
Item Type: FIN Finding
Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip
A self-revealing Green finding was identified for a loss of design control during maintenance on the integrated control system which later resulted in a reactor trip.

This finding is more than minor because it affected the configuration control attribute of the initiating event cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. The issue was of very low safety significance because although there was a reactor trip, mitigating systems remained available and were not affected. Because no safety systems were affected, the finding did not involve a violation of regulatory requirements. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3.1)
Inspection Report# : 2004003*(pdf)*

Significance: G Dec 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Perform the Magnetic Particle Calibration
The inspector identified a non-cited violation of Technical Specification 5.6.1.1 for failure to follow procedural requirements involving incorrect calibration of a magnetic particle testing (MT) yoke. This finding could have inhibited the identification of indications or flaws on American Society of Mechanical Engineers (ASME) Class 2 Safety-Related Feed Water to Once Through Steam Generator (OTSG) "A" piping.

This finding is more than minor because if left uncorrected, it could result in a more significant safety concern. Failure to correctly perform the calibration could reduce the ability to discover indications or flaws which could lead to pipe breaks. The issue was determined to be of very low safety significance because the likelihood of a loss of coolant accident (LOCA) initiator was not affected, the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have available, and the finding did not increase the likelihood of a fire or flood. (Section 1R08)
Inspection Report# : 2003006*(pdf)*

Mitigating Systems

Significance: G Sep 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Maintain Two Operable Control Complex Cooling Trains
A self-revealing non-cited violation of Crystal River 3 Technical Specification 3.7.18 was identified. Following Train B chiller maintenance on December 19, 2002, and Train A chiller maintenance on February 25, 2003, neither train of control complex cooling was operable because control complex chiller motor overload relays had been improperly set below their design values. The problem was identified on June 11, 2003, when both chiller motors tripped on overload current, when an overload current condition had not occurred.

The self-revealing finding is greater than minor safety significance because it resulted in a loss of the control complex cooling safety function and affected the availability and reliability of the Mitigating Systems Cornerstone of Reactor Safety that is used to mitigate events. The finding is of very low safety significance because the alternate non-safety Appendix R cooling system and feedwater pump (FWP-7) were available to mitigate transients involving systems that could be affected by the loss of cooling. (Section 4OA3)

Inspection Report# : 2003005*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Protect One Train of Safe Shutdown Equipment From Fire Damage
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix R, Section III.G.2, Fire Protection of Safe Shutdown Capability, for failure to protect certain electrical cables for safe shutdown equipment from fire damage in three fire areas. The licensee has corrected related identified procedural deficiencies and plans to resolve the noncompliance with cable protection through licensing correspondence with the NRC.

This finding is greater than minor safety significance because it involved a lack of required fire barriers for equipment relied upon for safe shutdown following a fire and because it affected the objectives of the Mitigating Systems Cornerstone of Reactor Safety. It affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The finding is of very low safety significance because licensee's proceduralized manual actions are reasonably accomplishable and training would have enabled operators to maintain the makeup function sufficiently to maintain reactor coolant system process variables within acceptable ranges. Therefore, the inspectors identified this issue as a Green finding as described in Inspection Procedure 71111.05, Fire Protection. (Section 4OA5)
Inspection Report# : 2003005*(pdf)*

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Implement Inservice Testing Program Requirements (Section 1R22).
A finding was identified for failure to implement increased frequency testing of a safety-related pump, after the pump differential pressure was found in the Alert range of the ASME Code, Section XI test on December 2, 2002. When tested on May 22, 2003, the pump was found in the Action range and was declared inoperable.

A non-cited violation of Technical Specification 5.6.2 was identified. The finding is greater than minor because an engineering evaluation was required to assure that accident analysis requirements were met during the subsequent period of operation with differential pressure below the design minimum value. If the finding had not been corrected, pump performance could have resulted in the safety system not being capable of performing its design function to remove residual heat following an accident. The finding is of very low safety significance because the maximum period of operation below the design minimum differential pressure was of short duration and redundancy existed that assured the safety function remained available. (Section 1R22)
Inspection Report# : 2003004*(pdf)*

Barrier Integrity

Significance: G Dec 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles
A self-revealing non-cited violation of Technical Specification 3.4.12.a was identified. Small cracks in the pressurizer upper level instrument tap nozzles resulted in pressure boundary leakage since late 2000.

The finding was greater than minor because the breach in the reactor coolant system (RCS) affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. However, the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzle. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. (Section 4OA3.3)

Inspection Report# : 2003006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Miscellaneous

Last modified : May 05, 2004

Crystal River 3
2Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 26, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedures in 10 CFR 50.59 Screening
An NRC identified, Non-Cited Violation (NCV) of Technical Specification 5.6.1.1 was identified for failure to fully implement a procedure which required a 10 CFR 50.59 evaluation to be completed for a one-time test of the power operated relief valve (PORV). Because the required evaluation was not completed, the licensee was unaware that the test would result in opening the PORV. As a result, the PORV unexpectedly opened for a very short period while the plant was operating and caused a reactor pressure transient.

This finding is more than minor because it affected the Primary System Loss of Coolant Accident (LOCA) Initiator attribute of the Initiating Events Cornerstone. The issue was of very low safety significance because although PORV opened for a short period of time with the reactor operating at power, mitigating systems, including the PORV block valve, were available had the valve failed to shut. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3)

Inspection Report# : 2004004*(pdf)*

Significance: G Mar 27, 2004
Identified By: Self Disclosing
Item Type: FIN Finding
Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip
A self-revealing Green finding was identified for a loss of design control during maintenance on the integrated control system which later resulted in a reactor trip.

This finding is more than minor because it affected the configuration control attribute of the initiating event cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. The issue was of very low safety significance because although there was a reactor trip, mitigating systems remained available and were not affected. Because no safety systems were affected, the finding did not involve a violation of regulatory requirements. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3.1)
Inspection Report# : 2004003*(pdf)*

Significance: G Dec 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Perform the Magnetic Particle Calibration
The inspector identified a non-cited violation of Technical Specification 5.6.1.1 for failure to follow procedural requirements involving incorrect calibration of a magnetic particle testing (MT) yoke. This finding could have inhibited the identification of indications or flaws on American Society of Mechanical Engineers (ASME) Class 2 Safety-Related Feed Water to Once Through Steam Generator (OTSG) "A" piping.

This finding is more than minor because if left uncorrected, it could result in a more significant safety concern. Failure to correctly perform the calibration could reduce the ability to discover indications or flaws which could lead to pipe breaks. The issue was determined to be of very low safety significance because the likelihood of a loss of coolant accident (LOCA) initiator was not affected, the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have available, and the finding did not increase the likelihood of a fire or flood. (Section 1R08)
Inspection Report# : 2003006*(pdf)*

Mitigating Systems

Significance: G Sep 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Maintain Two Operable Control Complex Cooling Trains

A self-revealing non-cited violation of Crystal River 3 Technical Specification 3.7.18 was identified. Following Train B chiller maintenance on December 19, 2002, and Train A chiller maintenance on February 25, 2003, neither train of control complex cooling was operable because control complex chiller motor overload relays had been improperly set below their design values. The problem was identified on June 11, 2003, when both chiller motors tripped on overload current, when an overload current condition had not occurred.

The self-revealing finding is greater than minor safety significance because it resulted in a loss of the control complex cooling safety function and affected the availability and reliability of the Mitigating Systems Cornerstone of Reactor Safety that is used to mitigate events. The finding is of very low safety significance because the alternate non-safety Appendix R cooling system and feedwater pump (FWP-7) were available to mitigate transients involving systems that could be affected by the loss of cooling. (Section 4OA3)

Inspection Report# : 2003005*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Protect One Train of Safe Shutdown Equipment From Fire Damage
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix R, Section III.G.2, Fire Protection of Safe Shutdown Capability, for failure to protect certain electrical cables for safe shutdown equipment from fire damage in three fire areas. The licensee has corrected related identified procedural deficiencies and plans to resolve the noncompliance with cable protection through licensing correspondence with the NRC.

This finding is greater than minor safety significance because it involved a lack of required fire barriers for equipment relied upon for safe shutdown following a fire and because it affected the objectives of the Mitigating Systems Cornerstone of Reactor Safety. It affected the availability and reliability of systems that mitigate initiating events to prevent undesirable consequences. The finding is of very low safety significance because licensee's proceduralized manual actions are reasonably accomplishable and training would have enabled operators to maintain the makeup function sufficiently to maintain reactor coolant system process variables within acceptable ranges. Therefore, the inspectors identified this issue as a Green finding as described in Inspection Procedure 71111.05, Fire Protection. (Section 4OA5)
Inspection Report# : 2003005*(pdf)*

Barrier Integrity

Significance: G Dec 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles
A self-revealing non-cited violation of Technical Specification 3.4.12.a was identified. Small cracks in the pressurizer upper level instrument tap nozzles resulted in pressure boundary leakage since late 2000.

The finding was greater than minor because the breach in the reactor coolant system (RCS) affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. However, the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzle. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. (Section 4OA3.3)

Inspection Report# : 2003006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : September 08, 2004

Crystal River 3
3Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 26, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedures in 10 CFR 50.59 Screening
An NRC identified, Non-Cited Violation (NCV) of Technical Specification 5.6.1.1 was identified for failure to fully implement a procedure which required a 10 CFR 50.59 evaluation to be completed for a one-time test of the power operated relief valve (PORV). Because the required evaluation was not completed, the licensee was unaware that the test would result in opening the PORV. As a result, the PORV unexpectedly opened for a very short period while the plant was operating and caused a reactor pressure transient.

This finding is more than minor because it affected the Primary System Loss of Coolant Accident (LOCA) Initiator attribute of the Initiating Events Cornerstone. The issue was of very low safety significance because although PORV opened for a short period of time with the reactor operating at power, mitigating systems, including the PORV block valve, were available had the valve failed to shut. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3)

Inspection Report# : 2004004*(pdf)*

Significance: G Mar 27, 2004
Identified By: Self Disclosing
Item Type: FIN Finding
Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip
A self-revealing Green finding was identified for a loss of design control during maintenance on the integrated control system which later resulted in a reactor trip.

This finding is more than minor because it affected the configuration control attribute of the initiating event cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. The issue was of very low safety significance because although there was a reactor trip, mitigating systems remained available and were not affected. Because no safety systems were affected, the finding did not involve a violation of regulatory requirements. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3.1)
Inspection Report# : 2004003*(pdf)*

Significance: G Dec 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Perform the Magnetic Particle Calibration
The inspector identified a non-cited violation of Technical Specification 5.6.1.1 for failure to follow procedural requirements involving incorrect calibration of a magnetic particle testing (MT) yoke. This finding could have inhibited the identification of indications or flaws on American Society of Mechanical Engineers (ASME) Class 2 Safety-Related Feed Water to Once Through Steam Generator (OTSG) "A" piping.

This finding is more than minor because if left uncorrected, it could result in a more significant safety concern. Failure to correctly perform the calibration could reduce the ability to discover indications or flaws which could lead to pipe breaks. The issue was determined to be of very low safety significance because the likelihood of a loss of coolant accident (LOCA) initiator was not affected, the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have available, and the finding did not increase the likelihood of a fire or flood. (Section 1R08)
Inspection Report# : 2003006*(pdf)*

Mitigating Systems

Significance: G Sep 25, 2004

Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Investigate Deficient Condition of Boric Acid Leakage Affecting The Low Pressure Injection System As Required By Boric Acid Corrosion Control Procedure
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, for failure to follow boric acid corrosion control program procedures that required an investigation of boric acid leakage identified on decay heat pump DHP-1B.

This finding is more than minor because if left uncorrected it could become a more significant concern, that being loss of integrity of components in the low pressure injection system. The finding was of very low safety significance because only minimal corrosion was observed when inspected. (Section 1RO4)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Establish Adequate Corrective Actions For Fire Brigade Response Results In A Recurrent Problem
The inspectors identified a Non-Cited Violation (NCV) of Crystal River 3 Operating License Condition 2.C.(9) when prompt corrective measures were not taken to ensure the availability of a fire brigade member to respond to a fire emergency.

This finding is more than minor because if left uncorrected, adequate fire response capability would be challenged which would be a more significant safety concern. A significance determination process review assumed fire confinement was affected with a low degradation rating which resulted in the finding being screened as having very low safety significance. The finding involved the cross-cutting element of problem and identification of resolution, in that interim corrective actions were narrowly focused and ineffective to prevent recurrence. (Section 1RO5)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Redundant Channels of A Post-Accident Monitoring Function Not Operable Due To Reversed Power Supplies Redundant channels of a post-accident monitoring function not operable due to reversed power supp
A self-revealing Non-Cited Violation (NCV) of Technical Specification 3.3.17 D was identified when both channels of the Degrees of Subcooling Monitor were found to have their respective power supplies crossed.

The finding was more than minor because the failure of degrees of subcooling monitor indication during certain LOCA scenarios could challenge the control room operators in taking timely action to establish the plant conditions (trip reactor coolant pumps within one minute) needed to assure safety. The finding was of very low safety significance because operators retained the ability to diagnose a loss of subcooling margin using emergency operating procedures had a loss of subcooling margin occurred. (Section 4OA3)

Inspection Report# : 2004005*(pdf)*

Barrier Integrity

Significance: G Dec 27, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles
A self-revealing non-cited violation of Technical Specification 3.4.12.a was identified. Small cracks in the pressurizer upper level instrument tap nozzles resulted in pressure boundary leakage since late 2000.

The finding was greater than minor because the breach in the reactor coolant system (RCS) affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. However, the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzle. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. (Section 4OA3.3)

Inspection Report# : 2003006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jul 02, 2004
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution
The licensee's corrective action program was generally effective at identifying problems at an appropriate threshold level and entering them into the corrective action program. Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluation for issues classified as significant conditions adverse to quality were especially comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. The inspectors generally found that the scope and depth of corrective actions implemented by the licensee were appropriate for the severity and risk significance of the problem identified. Industry operating experience items were effectively evaluated for applicability and entered into the corrective action program (CAP). Nuclear Assessment Section (NAS) 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 corrective action program and addressed. Based on discussions conducted with plant employees from various departments the inspectors did not identify any reluctance to report safety concerns. Further, the inspectors concluded that the licensee was aggressive in addressing potential chilling effect issues. However, the inspectors observed from the more recent data reviewed that several lower threshold issues had not been entered into the CAP. In addition, several examples were identified where problem evaluations lacked thoroughness or were narrowly focused.

Inspection Report# : 2004007*(pdf)*

Last modified : December 29, 2004

Crystal River 3
4Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 26, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedures in 10 CFR 50.59 Screening
An NRC identified, Non-Cited Violation (NCV) of Technical Specification 5.6.1.1 was identified for failure to fully implement a procedure which required a 10 CFR 50.59 evaluation to be completed for a one-time test of the power operated relief valve (PORV). Because the required evaluation was not completed, the licensee was unaware that the test would result in opening the PORV. As a result, the PORV unexpectedly opened for a very short period while the plant was operating and caused a reactor pressure transient.

This finding is more than minor because it affected the Primary System Loss of Coolant Accident (LOCA) Initiator attribute of the Initiating Events Cornerstone. The issue was of very low safety significance because although PORV opened for a short period of time with the reactor operating at power, mitigating systems, including the PORV block valve, were available had the valve failed to shut. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3)

Inspection Report# : 2004004*(pdf)*

Significance: G Mar 27, 2004
Identified By: Self Disclosing
Item Type: FIN Finding
Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip
A self-revealing Green finding was identified for a loss of design control during maintenance on the integrated control system which later resulted in a reactor trip.

This finding is more than minor because it affected the configuration control attribute of the initiating event cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. The issue was of very low safety significance because although there was a reactor trip, mitigating systems remained available and were not affected. Because no safety systems were affected, the finding did not involve a violation of regulatory requirements. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3.1)
Inspection Report# : 2004003*(pdf)*

Mitigating Systems

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Investigate Deficient Condition of Boric Acid Leakage Affecting The Low Pressure Injection System As Required By Boric Acid Corrosion Control Procedure
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, for failure to follow boric acid corrosion control program procedures that required an investigation of boric acid leakage identified on decay heat pump DHP-1B.

This finding is more than minor because if left uncorrected it could become a more significant concern, that being loss of integrity of components in the low pressure injection system. The finding was of very low safety significance because only minimal corrosion was observed when inspected. (Section 1RO4)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation

Redundant Channels of A Post-Accident Monitoring Function Not Operable Due To Reversed Power Supplies Redundant channels of a post-accident monitoring function not operable due to reversed power supp
A self-revealing Non-Cited Violation (NCV) of Technical Specification 3.3.17 D was identified when both channels of the Degrees of Subcooling Monitor were found to have their respective power supplies crossed.

The finding was more than minor because the failure of degrees of subcooling monitor indication during certain LOCA scenarios could challenge the control room operators in taking timely action to establish the plant conditions (trip reactor coolant pumps within one minute) needed to assure safety. The finding was of very low safety significance because operators retained the ability to diagnose a loss of subcooling margin using emergency operating procedures had a loss of subcooling margin occurred. (Section 4OA3)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Establish Adequate Corrective Actions For Fire Brigade Response Results In A Recurrent Problem
The inspectors identified a Non-Cited Violation (NCV) of Crystal River 3 Operating License Condition 2.C.(9) when prompt corrective measures were not taken to ensure the availability of a fire brigade member to respond to a fire emergency.

This finding is more than minor because if left uncorrected, adequate fire response capability would be challenged which would be a more significant safety concern. A significance determination process review assumed fire confinement was affected with a low degradation rating which resulted in the finding being screened as having very low safety significance. The finding involved the cross-cutting element of problem and identification of resolution, in that interim corrective actions were narrowly focused and ineffective to prevent recurrence. (Section 1RO5)

Inspection Report# : 2004005*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jul 02, 2004
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution
The licensee's corrective action program was generally effective at identifying problems at an appropriate threshold level and entering them into the corrective action program. Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluation

for issues classified as significant conditions adverse to quality were especially comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. The inspectors generally found that the scope and depth of corrective actions implemented by the licensee were appropriate for the severity and risk significance of the problem identified. Industry operating experience items were effectively evaluated for applicability and entered into the corrective action program (CAP). Nuclear Assessment Section (NAS) 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 corrective action program and addressed. Based on discussions conducted with plant employees from various departments the inspectors did not identify any reluctance to report safety concerns. Further, the inspectors concluded that the licensee was aggressive in addressing potential chilling effect issues. However, the inspectors observed from the more recent data reviewed that several lower threshold issues had not been entered into the CAP. In addition, several examples were identified where problem evaluations lacked thoroughness or were narrowly focused.

Inspection Report# : 2004007*(pdf)*

Last modified : March 09, 2005

Crystal River 3
1Q/2005 Plant Inspection Findings

Initiating Events

Significance: G Jun 26, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedures in 10 CFR 50.59 Screening
An NRC identified, Non-Cited Violation (NCV) of Technical Specification 5.6.1.1 was identified for failure to fully implement a procedure which required a 10 CFR 50.59 evaluation to be completed for a one-time test of the power operated relief valve (PORV). Because the required evaluation was not completed, the licensee was unaware that the test would result in opening the PORV. As a result, the PORV unexpectedly opened for a very short period while the plant was operating and caused a reactor pressure transient.

This finding is more than minor because it affected the Primary System Loss of Coolant Accident (LOCA) Initiator attribute of the Initiating Events Cornerstone. The issue was of very low safety significance because although PORV opened for a short period of time with the reactor operating at power, mitigating systems, including the PORV block valve, were available had the valve failed to shut. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3)

Inspection Report# : 2004004*(pdf)*

Mitigating Systems

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Investigate Deficient Condition of Boric Acid Leakage Affecting The Low Pressure Injection System As Required By Boric Acid Corrosion Control Procedure
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, for failure to follow boric acid corrosion control program procedures that required an investigation of boric acid leakage identified on decay heat pump DHP-1B.

This finding is more than minor because if left uncorrected it could become a more significant concern, that being loss of integrity of components in the low pressure injection system. The finding was of very low safety significance because only minimal corrosion was observed when inspected. (Section 1RO4)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Redundant Channels of A Post-Accident Monitoring Function Not Operable Due To Reversed Power Supplies Redundant channels of a post-accident monitoring function not operable due to reversed power supp
A self-revealing Non-Cited Violation (NCV) of Technical Specification 3.3.17 D was identified when both channels of the Degrees of Subcooling Monitor were found to have their respective power supplies crossed.

The finding was more than minor because the failure of degrees of subcooling monitor indication during certain LOCA scenarios could challenge the control room operators in taking timely action to establish the plant conditions (trip reactor coolant pumps within one minute) needed to assure safety. The finding was of very low safety significance because operators retained the ability to diagnose a loss of subcooling margin using emergency operating procedures had a loss of subcooling margin occurred. (Section 4OA3)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: NRC

Item Type: NCV NonCited Violation
Failure To Establish Adequate Corrective Actions For Fire Brigade Response Results In A Recurrent Problem
The inspectors identified a Non-Cited Violation (NCV) of Crystal River 3 Operating License Condition 2.C.(9) when prompt corrective measures were not taken to ensure the availability of a fire brigade member to respond to a fire emergency.

This finding is more than minor because if left uncorrected, adequate fire response capability would be challenged which would be a more significant safety concern. A significance determination process review assumed fire confinement was affected with a low degradation rating which resulted in the finding being screened as having very low safety significance. The finding involved the cross-cutting element of problem and identification of resolution, in that interim corrective actions were narrowly focused and ineffective to prevent recurrence. (Section 1RO5)

Inspection Report# : 2004005*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jul 02, 2004
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution
The licensee's corrective action program was generally effective at identifying problems at an appropriate threshold level and entering them into the corrective action program. Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluation for issues classified as significant conditions adverse to quality were especially comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. The inspectors generally found that the scope and depth of corrective actions implemented by the licensee were appropriate for the severity and risk significance of the problem identified. Industry operating experience items were effectively evaluated for applicability and entered into the corrective action program (CAP). Nuclear Assessment Section (NAS) 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 corrective action program and addressed. Based on discussions conducted with plant employees from various departments the inspectors did not identify any reluctance to report safety concerns. Further, the inspectors concluded that the licensee was aggressive in addressing potential chilling effect issues. However, the inspectors observed from the more recent data reviewed that several lower threshold issues had not been entered into the CAP. In addition, several examples were identified where problem evaluations lacked thoroughness or were narrowly focused.

Inspection Report# : 2004007*(pdf)*

Last modified : June 17, 2005

Crystal River 3
2Q/2005 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply
The inspectors identified a non-cited violation when the licensee failed to establish appropriate quantitative acceptance criteria to assure offsite power operability for compliance with Crystal River 3 Technical Specification 3.8.1.

This finding is more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The finding was of very low safety significance according to the SDP Phase 1 worksheet since none of the functions identified in phase 1 were degraded as a result of this deficiency. (Section 1R22)

Inspection Report# : 2005003*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves
A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures occurred in the ‘A' EGDG during fast start surveillance tests, conducted on April 23, 2004 (NCR 125149) and March 23, 2005 (NCR 154522). A third related failure was identified in July 5, 2001, when the ‘B' EGDG failed to start during a monthly surveillance test (NCR 44603).

This finding is more than minor because it directly affected the mitigating system cornerstone objective of ensuring the reliability and operability of a mitigating system. This issue was of very low safety significance because the condition did not affect the capability of the ‘A' EGDG to perform its design safety function. In addition, the slower fast start time was bounded by the accident analysis calculations. Corrective actions included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system. (Section 4OA3)

Inspection Report# : 2005003*(pdf)*

Significance: TBD Jun 16, 2005
Identified By: NRC
Item Type: AV Apparent Violation
Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action
Preliminary Greater than Green. An apparent violation of 10 CFR 50, Appendix R, Section III.G.2, for failure to physically protect or separate cables from fire damage and instead relying on an unapproved local manual operator action. The unprotected cables were associated with a common electrical protection and metering circuit which was installed such that fire damage to a cable in or just above the 3A 4160V engineered safeguards (ES) switchgear could result in tripping and locking out all feeder breakers to both 4160V ES busses, resulting in a loss of all safety-related alternating current power.

In addition, the local manual operator action to reset the 3B emergency diesel generator breaker lockout on the 3B 4160V ES switchgear was determined to be non-feasible. During a severe fire in the adjacent 3A 4160V Switchgear Room the fire response activities would cause the location for the operator action (the 3B 4160V Switchgear Room) to be exposed to hot smoke, water mist, and water on the floor.

This finding is greater than minor because it degraded the defense in depth for fire protection and also because it is associated with the protection against external factors attribute and degraded the reactor safety mitigating systems cornerstone objective. The finding adversely affected the reliability and capability of equipment required to achieve and maintain a safe shutdown condition following a severe fire in the 3A 4160V ES Switchgear Room. (Section 4OA5.01)

Inspection Report# : 2005007*(pdf)*

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Investigate Deficient Condition of Boric Acid Leakage Affecting The Low Pressure Injection System As Required By Boric Acid Corrosion Control Procedure
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, for failure to follow boric acid corrosion control program procedures that required an investigation of boric acid leakage identified on decay heat pump DHP-1B.

This finding is more than minor because if left uncorrected it could become a more significant concern, that being loss of integrity of components in the low pressure injection system. The finding was of very low safety significance because only minimal corrosion was observed when inspected. (Section 1RO4)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Redundant Channels of A Post-Accident Monitoring Function Not Operable Due To Reversed Power Supplies Redundant channels of a post-accident monitoring function not operable due to reversed power supp
A self-revealing Non-Cited Violation (NCV) of Technical Specification 3.3.17 D was identified when both channels of the Degrees of Subcooling Monitor were found to have their respective power supplies crossed.

The finding was more than minor because the failure of degrees of subcooling monitor indication during certain LOCA scenarios could challenge the control room operators in taking timely action to establish the plant conditions (trip reactor coolant pumps within one minute) needed to assure safety. The finding was of very low safety significance because operators retained the ability to diagnose a loss of subcooling margin using emergency operating procedures had a loss of subcooling margin occurred. (Section 4OA3)

Inspection Report# : 2004005*(pdf)*

Significance: G Sep 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Establish Adequate Corrective Actions For Fire Brigade Response Results In A Recurrent Problem
The inspectors identified a Non-Cited Violation (NCV) of Crystal River 3 Operating License Condition 2.C.(9) when prompt corrective measures were not taken to ensure the availability of a fire brigade member to respond to a fire emergency.

This finding is more than minor because if left uncorrected, adequate fire response capability would be challenged which would be a more significant safety concern. A significance determination process review assumed fire confinement was affected with a low degradation rating which resulted in the finding being screened as having very low safety significance. The finding involved the cross-cutting element of problem and identification of resolution, in that interim corrective actions were narrowly focused and ineffective to prevent recurrence. (Section 1RO5)

Inspection Report# : 2004005*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jul 02, 2004
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution
The licensee's corrective action program was generally effective at identifying problems at an appropriate threshold level and entering them into the corrective action program. Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluation for issues classified as significant conditions adverse to quality were especially comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. The inspectors generally found that the scope and depth of corrective actions implemented by the licensee were appropriate for the severity and risk significance of the problem identified. Industry operating experience items were effectively evaluated for applicability and entered into the corrective action program (CAP). Nuclear Assessment Section (NAS) 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 corrective action program and addressed. Based on discussions conducted with plant employees from various departments the inspectors did not identify any reluctance to report safety concerns. Further, the inspectors concluded that the licensee was aggressive in addressing potential chilling effect issues. However, the inspectors observed from the more recent data reviewed that several lower threshold issues had not been entered into the CAP. In addition, several examples were identified where problem evaluations lacked thoroughness or were narrowly focused.

Inspection Report# : 2004007*(pdf)*

Last modified : August 24, 2005

Crystal River 3
3Q/2005 Plant Inspection Findings

Initiating Events

Mitigating Systems

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)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply
The inspectors identified a non-cited violation when the licensee failed to establish appropriate quantitative acceptance criteria to assure offsite power operability for compliance with Crystal River 3 Technical Specification 3.8.1.

This finding is more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The finding was of very low safety significance according to the SDP Phase 1 worksheet since none of the functions identified in phase 1 were degraded as a result of this deficiency. (Section 1R22)

Inspection Report# : 2005003*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves
A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures occurred in the ‘A' EGDG during fast start surveillance tests, conducted on April 23, 2004 (NCR 125149) and March 23, 2005 (NCR 154522). A third related failure was identified in July 5, 2001, when the ‘B' EGDG failed to start during a monthly surveillance test (NCR 44603).

This finding is more than minor because it directly affected the mitigating system cornerstone objective of ensuring the reliability and operability of a mitigating system. This issue was of very low safety significance because the condition did not affect the capability of the ‘A' EGDG to perform its design safety function. In addition, the slower fast start time was bounded by the accident analysis calculations. Corrective actions included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system. (Section 4OA3)

Inspection Report# : 2005003*(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

Last modified : November 30, 2005

Crystal River 3
4Q/2005 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 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)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply
The inspectors identified a non-cited violation when the licensee failed to establish appropriate quantitative acceptance criteria to assure offsite power operability for compliance with Crystal River 3 Technical Specification 3.8.1.

This finding is more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The finding was of very low safety significance according to the SDP Phase 1 worksheet since none of the functions identified in phase 1 were degraded as a result of this deficiency. (Section 1R22)

Inspection Report# : 2005003*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves
A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures occurred in the ‘A' EGDG during fast start surveillance tests, conducted on April 23, 2004 (NCR 125149) and March 23, 2005 (NCR 154522). A third related failure was identified in July 5, 2001, when the ‘B' EGDG failed to start during a monthly surveillance test (NCR 44603).

This finding is more than minor because it directly affected the mitigating system cornerstone objective of ensuring the reliability and operability of a mitigating system. This issue was of very low safety significance because the condition did not affect the capability of the ‘A' EGDG to perform its design safety function. In addition, the slower fast start time was bounded by the accident analysis calculations. Corrective actions included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system. (Section 4OA3)

Inspection Report# : 2005003*(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

Last modified : March 03, 2006

Crystal River 3
1Q/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 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)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply
The inspectors identified a non-cited violation when the licensee failed to establish appropriate quantitative acceptance criteria to assure offsite power operability for compliance with Crystal River 3 Technical Specification 3.8.1.

This finding is more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The finding was of very low safety significance according to the SDP Phase 1 worksheet since none of the functions identified in phase 1 were degraded as a result of this deficiency. (Section 1R22)

Inspection Report# : 2005003*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves
A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures occurred in the 'A' EGDG during fast start surveillance tests, conducted on April 23, 2004 (NCR 125149) and March 23, 2005 (NCR 154522). A third related failure was identified in July 5, 2001, when the 'B' EGDG failed to start during a monthly surveillance test (NCR 44603).

This finding is more than minor because it directly affected the mitigating system cornerstone objective of ensuring the reliability and operability of a mitigating system. This issue was of very low safety significance because the condition did not affect the capability of the 'A' EGDG to perform its design safety function. In addition, the slower fast start time was bounded by the accident analysis calculations. Corrective actions included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system. (Section 4OA3)

Inspection Report# : 2005003*(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

Last modified : May 25, 2006

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

Crystal River 3
3Q/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)*

Barrier Integrity

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 : December 21, 2006

Crystal River 3
4Q/2006 Plant Inspection Findings

Initiating Events

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)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation

Failure to Conduct Adequate Surveys for Liquid Effluent Releases
The inspectors identified a non-cited violation (NCV) of 10 CFR 20.1501(a) for failure to perform accurate surveys to demonstrate compliance with Technical Specification (TS) 5.6.2.3 Offsite Dose Calculation Manual (ODCM) controls used to maintain doses to members of the public from radioactive effluents as low as reasonably achievable (ALARA) in accordance with Appendix I to 10 CFR 50 design criteria as specified in 10 CFR 50.36a. Specifically, as of December 4, 2006, the licensee failed to conduct adequate dose evaluations to demonstrate compliance with TS 5.6.2.3 for radioactive liquid effluent releases made from the station discharge tank SDT-1 to a percolation pond located within the owner controlled area. The failure to conduct accurate dose evaluations for this liquid release pathway impaired the licensee's ability to demonstrate compliance with ODCM ALARA limits for the liquid radioactive waste processing equipment and operations. The issue was entered into the licensee's corrective action program for resolution.

The violation is more than minor because it adversely affects the program and process attribute of the Public Radiation Safety cornerstone, in that it involved an occurrence in the licensee's radioactive effluent release program that is contrary to NRC regulations. The finding was determined to be of very low safety significance because preliminary calculations based on recently determined dilution factors for the settling pond demonstrated that resultant offsite dose values were small fractions of the ODCM limits (Appendix I to 10 CFR Part 50 design criteria). Further, evaluations of radionuclide concentrations in the effluent were conducted in accordance with 10 CFR 20.1302(b) (2)(i) to demonstrate compliance with 10 CFR 20.1301 limits. (Section 2PS1)

Inspection Report# : 2006005 *(pdf)*

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 : March 01, 2007

Crystal River 3
1Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Address Marine Fouling Resulted in a Plant Transient
A self-revealing finding was identified for the failure to address the marine fouling failure mode in the scope of the existing preventive maintenance on the intake screen wash auto start system. As a result, reactor power had to be decreased to 80 percent to maintain condenser operating temperature limits. The licensee entered the issue into the corrective action program. Corrective actions included cleaning both the low and high side differential level sensing tubes, replacing tubes as needed, and implementing preventive maintenance procedures to periodically clean the tubes.

The finding was more than minor since it affected the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically the Operating Experience (OE) Program, in that, the licensee did not adequately implement OE through changes to station procedures to provide instructions to clean the sensing tubes during preventive maintenance on the system. (Section 4OA2.2)
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip
A self-revealing finding was identified for failure to replace a non-refurbished integrated control system (ICS) multiplier module that had been temporarily installed during the Fall 2005 refueling outage. As a result, an age-related failure of a multiplier module resulted in an automatic reactor trip. The licensee entered the issue into the corrective action program. Corrective actions completed and/or proposed include: installation of a refurbished multiplier module; development of an engineering refueling outage turnover checklist to ensure formal followup actions are implemented whenever components not of desired quality are installed; and briefing of engineering personnel of this event.

The finding was more than minor because it affected the equipment reliability attribute of the Initiating Events Cornerstone and resulted in an automatic reactor trip that upset plant stability and challenged critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance, specifically Decision Making in that the licensee did not adequately communicate decisions and the basis for decisions to personnel who have a need to know the information. (Section 4OA3.1)

Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Identify and Correct Repetitive Raw Water System Flush Water Strainer Baskets Degradation
A self-revealing, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI was identified for failure to identify and take appropriate corrective actions for repetitive failures of the raw water pumps bearing flush water strainer baskets. As a result, both raw water pumps, RWP-2B and RWP-3B, were inoperable for a period greater than that allowed by Improved Technical Specifications when shell debris passed through a corroded strainer and clogged the cyclone separator discharge piping. The licensee entered the issue into the corrective action program. New strainer baskets made of a material compatible with service conditions were installed. Additional corrective actions include: performing routine engineering review of degraded conditions found during preventative maintenance activities; revision to applicable surveillance procedures, and counseling of maintenance and engineering personnel on the need to identify and document adverse conditions in the corrective action program.

The finding was more than minor because it affected the equipment reliability attribute of the Mitigating System Cornerstone and resulted in a raw water train being inoperable for a period of time greater than allowed by Improved Technical Specifications. The finding was assessed through the Significance Determination Process (SDP) Phase 1 screening worksheet and determined to be of very low safety significance since the raw water pumps with a degraded flush water system had a very high likelihood of performing their safety function during a loss of offsite power event. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically, the licensee did not document the adverse condition of degraded strainer baskets in the corrective action program after it was determined that the filtering ability of the cyclone separator was a required design function. (Section 4OA3.2)

Inspection Report# : 2007002 *(pdf)*

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)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Conduct Adequate Surveys for Liquid Effluent Releases
The inspectors identified a non-cited violation (NCV) of 10 CFR 20.1501(a) for failure to perform accurate surveys to demonstrate compliance with Technical Specification (TS) 5.6.2.3 Offsite Dose Calculation Manual (ODCM) controls used to maintain doses to members of the public from radioactive effluents as low as reasonably achievable (ALARA) in accordance with Appendix I to 10 CFR 50 design criteria as specified in 10 CFR 50.36a. Specifically, as of December 4, 2006, the licensee failed to conduct adequate dose evaluations to demonstrate compliance with TS 5.6.2.3 for radioactive liquid effluent releases made from the station discharge tank SDT-1 to a percolation pond located within the owner controlled area. The failure to conduct accurate dose evaluations for this liquid release pathway impaired the licensee's ability to demonstrate compliance with ODCM ALARA limits for the liquid radioactive waste processing equipment and operations. The issue was entered into the licensee's corrective action program for resolution.

The violation is more than minor because it adversely affects the program and process attribute of the Public Radiation Safety cornerstone, in that it involved an occurrence in the licensee's radioactive effluent release program that is contrary to NRC regulations. The finding was determined to be of very low safety significance because preliminary calculations based on recently determined dilution factors for the settling pond demonstrated that resultant offsite dose values were small fractions of the ODCM limits (Appendix I to 10 CFR Part 50 design criteria). Further, evaluations of radionuclide concentrations in the effluent were conducted in accordance with 10 CFR 20.1302(b) (2)(i) to demonstrate compliance with 10 CFR 20.1301 limits. (Section 2PS1)

Inspection Report# : 2006005 *(pdf)*

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 : June 01, 2007

Crystal River 3
2Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Address Marine Fouling Resulted in a Plant Transient
A self-revealing finding was identified for the failure to address the marine fouling failure mode in the scope of the existing preventive maintenance on the intake screen wash auto start system. As a result, reactor power had to be decreased to 80 percent to maintain condenser operating temperature limits. The licensee entered the issue into the corrective action program. Corrective actions included cleaning both the low and high side differential level sensing tubes, replacing tubes as needed, and implementing preventive maintenance procedures to periodically clean the tubes.

The finding was more than minor since it affected the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically the Operating Experience (OE) Program, in that, the licensee did not adequately implement OE through changes to station procedures to provide instructions to clean the sensing tubes during preventive maintenance on the system. (Section 4OA2.2)
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip
A self-revealing finding was identified for failure to replace a non-refurbished integrated control system (ICS) multiplier module that had been temporarily installed during the Fall 2005 refueling outage. As a result, an age-related failure of a multiplier module resulted in an automatic reactor trip. The licensee entered the issue into the corrective action program. Corrective actions completed and/or proposed include: installation of a refurbished multiplier module; development of an engineering refueling outage turnover checklist to ensure formal followup actions are implemented whenever components not of desired quality are installed; and briefing of engineering personnel of this event.

The finding was more than minor because it affected the equipment reliability attribute of the Initiating Events Cornerstone and resulted in an automatic reactor trip that upset plant stability and challenged critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance, specifically Decision Making in that the licensee did not adequately communicate decisions and the basis for decisions to personnel who have a need to know the information. (Section 4OA3.1)

Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish Preventative Maintenance Procedures for Hydrostatic Seals Necessary to Protect Safety-Related Equipment from Internal Flooding
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately establish and implement procedures required by Regulatory Guide 1.33, Appendix A, Section 9, Procedures for Performing Maintenance (PM). Specifically, no procedure, program or process existed to periodically inspect hydrostatic barriers to identify and repair any degradation of the seals which provide protection of safety-related equipment from internal flooding. Corrective actions completed or planned include: Repair and qualify applicable fire seals as hydrostatic barriers and establish a hydrostatic penetration seal preventative maintenance program.

The finding is more than minor because it affected the protection against external factors (i.e. flood hazard) attribute of the Mitigating System cornerstone and could have impacted the availability of mitigating equipment during an internal flood event if left uncorrected. The inspectors determined that several degraded fire barrier seals did not meet hydrostatic barrier acceptability requirements. The finding was assessed through the SDP Phase 1 screening and determined to be of very low safety significance since the as-found condition of the hydrostatic barriers would not have resulted in the loss or degradation of safety-related mitigating equipment in the event of an internal flood.

Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Identify and Correct Repetitive Raw Water System Flush Water Strainer Baskets Degradation
A self-revealing, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI was identified for failure to identify and take appropriate corrective actions for repetitive failures of the raw water pumps bearing flush water strainer baskets. As a result, both raw water pumps, RWP-2B and RWP-3B, were inoperable for a period greater than that allowed by Improved Technical Specifications when shell debris passed through a corroded strainer and clogged the cyclone separator discharge piping. The licensee entered the issue into the corrective action program. New strainer baskets made of a material compatible with service conditions were installed. Additional corrective actions include: performing routine engineering review of degraded conditions found during preventative maintenance activities; revision to applicable surveillance procedures, and counseling of maintenance and engineering personnel on the need to identify and document adverse conditions in the corrective action program.

The finding was more than minor because it affected the equipment reliability attribute of the Mitigating System Cornerstone and resulted in a raw water train being inoperable for a period of time greater than allowed by Improved Technical Specifications. The finding was assessed through the Significance Determination Process (SDP) Phase 1 screening worksheet and determined to be of very low safety significance since the raw water pumps with a degraded flush water system had a very high likelihood of performing their safety function during a loss of offsite power event. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically, the licensee did not document the adverse condition of degraded strainer baskets in the corrective action program after it was determined that the filtering ability of the cyclone separator was a required design function. (Section 4OA3.2)

Inspection Report# : 2007002 *(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Conduct Adequate Surveys for Liquid Effluent Releases
The inspectors identified a non-cited violation (NCV) of 10 CFR 20.1501(a) for failure to perform accurate surveys to demonstrate compliance with Technical Specification (TS) 5.6.2.3 Offsite Dose Calculation Manual (ODCM) controls used to maintain doses to members of the public from radioactive effluents as low as reasonably achievable (ALARA) in accordance with Appendix I to 10 CFR 50 design criteria as specified in 10 CFR 50.36a. Specifically, as of December 4, 2006, the licensee failed to conduct adequate dose evaluations to demonstrate compliance with TS 5.6.2.3 for radioactive liquid effluent releases made from the station discharge tank SDT-1 to a percolation pond located within the owner controlled area. The failure to conduct accurate dose evaluations for this liquid release pathway impaired the licensee's ability to demonstrate compliance with ODCM ALARA limits for the liquid radioactive waste processing equipment and operations. The issue was entered into the licensee's corrective action program for resolution.

The violation is more than minor because it adversely affects the program and process attribute of the Public Radiation Safety cornerstone, in that it involved an occurrence in the licensee's radioactive effluent release program that is contrary to NRC regulations. The finding was determined to be of very low safety significance because preliminary calculations based on recently determined dilution factors for the settling pond demonstrated that resultant offsite dose values were small fractions of the ODCM limits (Appendix I to 10 CFR Part 50 design criteria). Further, evaluations of radionuclide concentrations in the effluent were conducted in accordance with 10 CFR 20.1302(b) (2) (i) to demonstrate compliance with 10 CFR 20.1301 limits. (Section 2PS1)

Inspection Report# : 2006005 *(pdf)*

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

Last modified : August 24, 2007

Crystal River 3
3Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Address Marine Fouling Resulted in a Plant Transient
A self-revealing finding was identified for the failure to address the marine fouling failure mode in the scope of the existing preventive maintenance on the intake screen wash auto start system. As a result, reactor power had to be decreased to 80 percent to maintain condenser operating temperature limits. The licensee entered the issue into the corrective action program. Corrective actions included cleaning both the low and high side differential level sensing tubes, replacing tubes as needed, and implementing preventive maintenance procedures to periodically clean the tubes.

The finding was more than minor since it affected the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically the Operating Experience (OE) Program, in that, the licensee did not adequately implement OE through changes to station procedures to provide instructions to clean the sensing tubes during preventive maintenance on the system. (Section 4OA2.2)
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip
A self-revealing finding was identified for failure to replace a non-refurbished integrated control system (ICS) multiplier module that had been temporarily installed during the Fall 2005 refueling outage. As a result, an age-related failure of a multiplier module resulted in an automatic reactor trip. The licensee entered the issue into the corrective action program. Corrective actions completed and/or proposed include: installation of a refurbished multiplier module; development of an engineering refueling outage turnover checklist to ensure formal followup actions are implemented whenever components not of desired quality are installed; and briefing of engineering personnel of this event.

The finding was more than minor because it affected the equipment reliability attribute of the Initiating Events Cornerstone and resulted in an automatic reactor trip that upset plant stability and challenged critical safety functions. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance, specifically Decision Making in that the licensee did not adequately communicate decisions and the basis for decisions to personnel who have a need to know the information. (Section 4OA3.1)

Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish Preventative Maintenance Procedures for Hydrostatic Seals Necessary to Protect Safety-Related Equipment from Internal Flooding
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately establish and implement procedures required by Regulatory Guide 1.33, Appendix A, Section 9, Procedures for Performing Maintenance (PM). Specifically, no procedure, program or process existed to periodically inspect hydrostatic barriers to identify and repair any degradation of the seals which provide protection of safety-related equipment from internal flooding. Corrective actions completed or planned include: Repair and qualify applicable fire seals as hydrostatic barriers and establish a hydrostatic penetration seal preventative maintenance program.

The finding is more than minor because it affected the protection against external factors (i.e. flood hazard) attribute of the Mitigating System cornerstone and could have impacted the availability of mitigating equipment during an internal flood event if left uncorrected. The inspectors determined that several degraded fire barrier seals did not meet hydrostatic barrier acceptability requirements. The finding was assessed through the SDP Phase 1 screening and determined to be of very low safety significance since the as-found condition of the hydrostatic barriers would not have resulted in the loss or degradation of safety-related mitigating equipment in the event of an internal flood.

Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Identify and Correct Repetitive Raw Water System Flush Water Strainer Baskets Degradation
A self-revealing, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI was identified for failure to identify and take appropriate corrective actions for repetitive failures of the raw water pumps bearing flush water strainer baskets. As a result, both raw water pumps, RWP-2B and RWP-3B, were inoperable for a period greater than that allowed by Improved Technical Specifications when shell debris passed through a corroded strainer and clogged the cyclone separator discharge piping. The licensee entered the issue into the corrective action program. New strainer baskets made of a material compatible with service conditions were installed. Additional corrective actions include: performing routine engineering review of degraded conditions found during preventative maintenance activities; revision to applicable surveillance procedures, and counseling of maintenance and engineering personnel on the need to identify and document adverse conditions in the corrective action program.

The finding was more than minor because it affected the equipment reliability attribute of the Mitigating System Cornerstone and resulted in a raw water train being inoperable for a period of time greater than allowed by Improved Technical Specifications. The finding was assessed through the Significance Determination Process (SDP) Phase 1 screening worksheet and determined to be of very low safety significance since the raw water pumps with a degraded flush water system had a very high likelihood of performing their safety function during a loss of offsite power event. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically, the licensee did not document the adverse condition of degraded strainer baskets in the corrective action program after it was determined that the filtering ability of the cyclone separator was a required design function. (Section 4OA3.2)

Inspection Report# : 2007002 *(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Conduct Adequate Surveys for Liquid Effluent Releases
The inspectors identified a non-cited violation (NCV) of 10 CFR 20.1501(a) for failure to perform accurate surveys to demonstrate compliance with Technical Specification (TS) 5.6.2.3 Offsite Dose Calculation Manual (ODCM) controls used to maintain doses to members of the public from radioactive effluents as low as reasonably achievable (ALARA) in accordance with Appendix I to 10 CFR 50 design criteria as specified in 10 CFR 50.36a. Specifically, as of December 4, 2006, the licensee failed to conduct adequate dose evaluations to demonstrate compliance with TS 5.6.2.3 for radioactive liquid effluent releases made from the station discharge tank SDT-1 to a percolation pond located within the owner controlled area. The failure to conduct accurate dose evaluations for this liquid release pathway impaired the licensee's ability to demonstrate compliance with ODCM ALARA limits for the liquid radioactive waste processing equipment and operations. The issue was entered into the licensee's corrective action program for resolution.

The violation is more than minor because it adversely affects the program and process attribute of the Public Radiation Safety cornerstone, in that it involved an occurrence in the licensee's radioactive effluent release program that is contrary to NRC regulations. The finding was determined to be of very low safety significance because preliminary calculations based on recently determined dilution factors for the settling pond demonstrated that resultant offsite dose values were small fractions of the ODCM limits (Appendix I to 10 CFR Part 50 design criteria). Further, evaluations of radionuclide concentrations in the effluent were conducted in accordance with 10 CFR 20.1302(b) (2) (i) to demonstrate compliance with 10 CFR 20.1301 limits. (Section 2PS1)

Inspection Report# : 2006005 *(pdf)*

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

Last modified : December 07, 2007

Crystal River 3
4Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Address Marine Fouling Resulted in a Plant Transient
A self-revealing finding was identified for the failure to address the marine fouling failure mode in the scope of the existing preventive maintenance on the intake screen wash auto start system. As a result, reactor power had to be decreased to 80 percent to maintain condenser operating temperature limits. The licensee entered the issue into the corrective action program. Corrective actions included cleaning both the low and high side differential level sensing tubes, replacing tubes as needed, and implementing preventive maintenance procedures to periodically clean the tubes.

The finding was more than minor since it affected the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Using the NRC Manual Chapter 0609, “Significance Determination Process,” Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically the Operating Experience (OE) Program, in that, the licensee did not adequately implement OE through changes to station procedures to provide instructions to clean the sensing tubes during preventive maintenance on the system. (Section 4OA2.2)
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip
A self-revealing finding was identified for failure to replace a non-refurbished integrated control system (ICS) multiplier module that had been temporarily installed during the Fall 2005 refueling outage. As a result, an age-related failure of a multiplier module resulted in an automatic reactor trip. The licensee entered the issue into the corrective action program. Corrective actions completed and/or proposed include: installation of a refurbished multiplier module; development of an engineering refueling outage turnover checklist to ensure formal followup actions are implemented whenever components not of desired quality are installed; and briefing of engineering personnel of this event.

The finding was more than minor because it affected the equipment reliability attribute of the Initiating Events Cornerstone and resulted in an automatic reactor trip that upset plant stability and challenged critical safety functions. Using the NRC Manual Chapter 0609, “Significance Determination Process,” Phase 1 screening worksheet, the finding was determined to be of very low safety significance since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance, specifically Decision Making in that the licensee did not adequately communicate decisions and the basis for decisions to personnel who have a need to know the information. (Section 4OA3.1)

Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA
The inspectors identified a finding of very low safety significance involving a violation of Technical Specifications (TS) 5.6.1 for failure to implement an adequate procedure for manual starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) following a Large Break Loss of Coolant Accident (LBLOCA). The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 degrees Fahrenheit (°F). The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that Nuclear Services Closed Cycle Cooling (SW) temperature falls below 104 °F no later than 84 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This issue is documented in the corrective action program as nuclear condition report (NCR) 247908. This finding was reviewed for cross-cutting aspects and none were identified.(Section 1R21.2.3
Inspection Report# : 2007006 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing
The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of Emergency Feedwater Pump (EFP)-2 in accordance with the approved In-service Testing (IST) program.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This issue is documented in the corrective actions program as NCR 248036. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.7)

Inspection Report# : 2007006 *(pdf)*

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish Preventative Maintenance Procedures for Hydrostatic Seals Necessary to Protect Safety-Related Equipment from Internal Flooding
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately establish and implement procedures required by Regulatory Guide 1.33, Appendix A, Section 9, Procedures for Performing Maintenance (PM). Specifically, no procedure, program or process existed to periodically inspect hydrostatic barriers to identify and repair any degradation of the seals which provide protection of safety-related equipment from internal flooding. Corrective actions completed or planned include: Repair and qualify applicable fire seals as hydrostatic barriers and establish a hydrostatic penetration seal preventative maintenance program.

The finding is more than minor because it affected the protection against external factors (i.e. flood hazard) attribute of the Mitigating System cornerstone and could have impacted the availability of mitigating equipment during an internal flood event if left uncorrected. The inspectors determined that several degraded fire barrier seals did not meet hydrostatic barrier acceptability requirements. The finding was assessed through the SDP Phase 1 screening and determined to be of very low safety significance since the as-found condition of the hydrostatic barriers would not have resulted in the loss or degradation of safety-related mitigating equipment in the event of an internal flood.

Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Identify and Correct Repetitive Raw Water System Flush Water Strainer Baskets Degradation
A self-revealing, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI was identified for failure to identify and take appropriate corrective actions for repetitive failures of the raw water pumps bearing flush water strainer baskets. As a result, both raw water pumps, RWP-2B and RWP-3B, were inoperable for a period greater than that allowed by Improved Technical Specifications when shell debris passed through a corroded strainer and clogged the cyclone separator discharge piping. The licensee entered the issue into the corrective action program. New strainer baskets made of a material compatible with service conditions were installed. Additional corrective actions include: performing routine engineering review of degraded conditions found during preventative maintenance activities; revision to applicable surveillance procedures, and counseling of maintenance and engineering personnel on the need to identify and document adverse conditions in the corrective action program.

The finding was more than minor because it affected the equipment reliability attribute of the Mitigating System Cornerstone and resulted in a raw water train being inoperable for a period of time greater than allowed by Improved Technical Specifications. The finding was assessed through the Significance Determination Process (SDP) Phase 1 screening worksheet and determined to be of very low safety significance since the raw water pumps with a degraded flush water system had a very high likelihood of performing their safety function during a loss of offsite power event. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically, the licensee did not document the adverse condition of degraded strainer baskets in the corrective action program after it was determined that the filtering ability of the cyclone separator was a required design function. (Section 4OA3.2)

Inspection Report# : 2007002 *(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

Last modified : February 04, 2008

Crystal River 3
1Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Oct 29, 2007
Identified By: NRC
Item Type: FIN Finding
Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient
A self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control switch during maintenance activities. As a result of bumping the control switch, a condensate pump had to be secured and reactor power was rapidly reduced to 61 percent to prevent a reactor trip. Corrective actions included removing the control switch handle to prevent it from being bumped.

The finding was more than minor since it affected the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenged critical safety functions. The inspectors referenced Inspection manual Chapter 0609.04, Significance Determination process (SDP), Phase 1 screening and determined the finding to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. A contributing cause of this finding is related to the crosscutting area of human performance, with a work control component. Specifically, the licensee did not adequately plan work activities to protect the condensate pump control switch from being bumped.
Inspection Report# : 2008002 *(pdf)*

Mitigating Systems

Significance: G Feb 22, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inoperable Fire Penetration Seal
The inspectors identified a Green non-cited violation (NCV) of Crystal River Unit 3 Operating License Condition 2.C (9), Fire Protection Program. The NCV was associated with an inoperable fire penetration seal in the 3-hour fire rated ceiling of the makeup system valve alley. The licensee declared the penetration seal inoperable. Corrective actions included establishing an hourly fire watch and repairing the penetration to its designed condition.

The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater 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. Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the finding was determined to have a very low safety significance since the gap in the fire penetration seal was small (less than 1/8 inch in width).

Inspection Report# : 2008002 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedural Guidance Associated with Removal of Containment Debris
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately implement procedures required by Regulatory Guide 1.33, Appendix A, Section 3, Procedures for Startup,

Operation, and Shutdown of Safety-Related PWR Systems. Specifically, the licensee failed to verify no latent debris was present in containment that could impact the emergency core cooling system (ECCS) sump. Corrective actions completed include: removal of the debris identified by the inspectors and performing additional inspection and cleaning of containment.

The finding is more than minor because it could be reasonably viewed as a precursor to a significant event involving debris accumulation on the containment sump screens which could cause impairment to ECCS recirculation flow during a design basis loss of coolant accident. The inspectors referenced Inspection Manual Chapter 0609, Significance Determination Process (SDP), Phase 1 screening and determined the finding to be of very low safety significance. Although the debris impacted the mitigating system cornerstone, it was unlikely to have resulted in an actual loss of safety function and was not potentially risk significant due to possible external events. A contributing cause of this finding is related to the crosscutting area of Human Performance, specifically Work Practices in that the licensee did not adequately comply with a containment inspection procedure. (IMC 305, H.4(b))

Inspection Report# : 2007005 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA
The inspectors identified a finding of very low safety significance involving a violation of Technical Specifications (TS) 5.6.1 for failure to implement an adequate procedure for manual starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) following a Large Break Loss of Coolant Accident (LBLOCA). The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 degrees Fahrenheit (°F). The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that Nuclear Services Closed Cycle Cooling (SW) temperature falls below 104 °F no later than 84 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This issue is documented in the corrective action program as nuclear condition report (NCR) 247908. This finding was reviewed for cross-cutting aspects and none were identified.(Section 1R21.2.3
Inspection Report# : 2007006 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing
The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of Emergency Feedwater Pump (EFP)-2 in accordance with the approved In-service Testing (IST) program.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This issue is documented in the corrective actions program as NCR 248036. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.7)

Inspection Report# : 2007006 *(pdf)*

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish Preventative Maintenance Procedures for Hydrostatic Seals Necessary to Protect Safety-Related Equipment from Internal Flooding
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately establish and implement procedures required by Regulatory Guide 1.33, Appendix A, Section 9, Procedures for Performing Maintenance (PM). Specifically, no procedure, program or process existed to periodically inspect hydrostatic barriers to identify and repair any degradation of the seals which provide protection of safety-related equipment from internal flooding. Corrective actions completed or planned include: Repair and qualify applicable fire seals as hydrostatic barriers and establish a hydrostatic penetration seal preventative maintenance program.

The finding is more than minor because it affected the protection against external factors (i.e. flood hazard) attribute of the Mitigating System cornerstone and could have impacted the availability of mitigating equipment during an internal flood event if left uncorrected. The inspectors determined that several degraded fire barrier seals did not meet hydrostatic barrier acceptability requirements. The finding was assessed through the SDP Phase 1 screening and determined to be of very low safety significance since the as-found condition of the hydrostatic barriers would not have resulted in the loss or degradation of safety-related mitigating equipment in the event of an internal flood.

Inspection Report# : 2007003 *(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

Last modified : June 05, 2008

Crystal River 3
2Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Control Transient Combustibles
The team identified a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C.(9), for the licensee's failure to properly implement fire protection program procedures for control of transient combustible materials. Specifically, transient combustible materials were left unattended for four days in the 3B 480V ES Switchgear Room after work had been completed, which was a violation of the licensee's administrative procedures for control of transient combustibles. Once identified, the licensee removed the combustible materials and initiated a nuclear condition report to address the issue.

The finding is more than minor because the transient combustible materials presented a credible fire scenario involving equipment important to safety, which degraded the reactor safety Initiating Events cornerstone objective to limit the likelihood of those events that may upset plant stability and challenge critical safety functions. The amount of unattended transient combustible materials did not violate the licensee's transient combustible control limits for the fire area. Therefore, the finding was assigned a low degradation rating against the combustible controls program. The finding was of very low safety significance (Green) based on the low degradation rating. This finding has a cross-cutting aspect in the Work Practices component of the Human Performance area because the licensee failed to effectively communicate expectations regarding procedural compliance and personnel following procedures (NRC Inspection Manual Chapter 0305, H.4(b)).
Inspection Report# : 2008006 *(pdf)*

Significance: G Feb 15, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Reactor Coolant Pump 1B Lube Oil Collection System Leakage
A self-revealing non-cited violation of 10 CFR 50, Appendix R, Section III.O, was identified for failure of the reactor coolant pump (RCP) oil collection system to collect and drain RCP oil leakage to a vented closed container. Specifically, the licensee found an estimated one to two gallons of oil on the reactor building floor beneath RCP-1B. The licensee initiated a nuclear condition report for this issue.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Initiating Events cornerstone objective. The team completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding was of very low safety significance (Green) because the amount of oil identified in 2008 was bounded by the licensee's 2004 analysis (which assumed a 21 gallon oil leak). This finding has a cross-cutting aspect in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee did not take appropriate corrective actions in a timely manner to address the adverse trend related to oil leakage for RCP-1B (NRC Inspection Manual Chapter 0305, P.1(d)).

Inspection Report# : 2008006 *(pdf)*

Significance: G Oct 29, 2007
Identified By: NRC
Item Type: FIN Finding
Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient
A self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control

switch during maintenance activities. As a result of bumping the control switch, a condensate pump had to be secured and reactor power was rapidly reduced to 61 percent to prevent a reactor trip. Corrective actions included removing the control switch handle to prevent it from being bumped.

The finding was more than minor since it affected the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenged critical safety functions. The inspectors referenced Inspection manual Chapter 0609.04, Significance Determination process (SDP), Phase 1 screening and determined the finding to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. A contributing cause of this finding is related to the crosscutting area of human performance, with a work control component. Specifically, the licensee did not adequately plan work activities to protect the condensate pump control switch from being bumped.
Inspection Report# : 2008002 *(pdf)*

Mitigating Systems

Significance: G Feb 22, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inoperable Fire Penetration Seal
The inspectors identified a Green non-cited violation (NCV) of Crystal River Unit 3 Operating License Condition 2.C (9), Fire Protection Program. The NCV was associated with an inoperable fire penetration seal in the 3-hour fire rated ceiling of the makeup system valve alley. The licensee declared the penetration seal inoperable. Corrective actions included establishing an hourly fire watch and repairing the penetration to its designed condition.

The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater 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. Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the finding was determined to have a very low safety significance since the gap in the fire penetration seal was small (less than 1/8 inch in width).

Inspection Report# : 2008002 *(pdf)*

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Protect Cables for Valve DHV-42
The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.2., for failure to protect cables from fire damage for components required for safe shutdown. Specifically, the Mecatiss MTS-3 fire wrap installed around the cables for valve DHV-42 (suction from the reactor building sump to the Train A decay heat pump) was not installed in accordance with the vendor’s tested configuration. The licensee initiated a nuclear condition report and implemented an hourly roving fire watch to address this issue. Additionally, the licensee implemented repairs during the March 2008 forced outage to upgrade the Mecatiss MTS-3 fire wrap to comply with the vendor tested configuration.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Mitigating Systems cornerstone objective. The inspectors completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding, when given credit for the fixed automatic suppression system in the area, was of very low safety significance (Green).

Inspection Report# : 2008006 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedural Guidance Associated with Removal of Containment Debris
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately implement procedures required by Regulatory Guide 1.33, Appendix A, Section 3, Procedures for Startup, Operation, and Shutdown of Safety-Related PWR Systems. Specifically, the licensee failed to verify no latent debris was present in containment that could impact the emergency core cooling system (ECCS) sump. Corrective actions completed include: removal of the debris identified by the inspectors and performing additional inspection and cleaning of containment.

The finding is more than minor because it could be reasonably viewed as a precursor to a significant event involving debris accumulation on the containment sump screens which could cause impairment to ECCS recirculation flow during a design basis loss of coolant accident. The inspectors referenced Inspection Manual Chapter 0609, Significance Determination Process (SDP), Phase 1 screening and determined the finding to be of very low safety significance. Although the debris impacted the mitigating system cornerstone, it was unlikely to have resulted in an actual loss of safety function and was not potentially risk significant due to possible external events. A contributing cause of this finding is related to the crosscutting area of Human Performance, specifically Work Practices in that the licensee did not adequately comply with a containment inspection procedure. (IMC 305, H.4(b))

Inspection Report# : 2007005 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA
The inspectors identified a finding of very low safety significance involving a violation of Technical Specifications (TS) 5.6.1 for failure to implement an adequate procedure for manual starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) following a Large Break Loss of Coolant Accident (LBLOCA). The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 degrees Fahrenheit (°F). The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that Nuclear Services Closed Cycle Cooling (SW) temperature falls below 104 °F no later than 84 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This issue is documented in the corrective action program as nuclear condition report (NCR) 247908. This finding was reviewed for cross-cutting aspects and none were identified.(Section 1R21.2.3
Inspection Report# : 2007006 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing
The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of Emergency Feedwater Pump (EFP)-2 in accordance with the approved In-service Testing (IST) program.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems

Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This issue is documented in the corrective actions program as NCR 248036. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.7)

Inspection Report# : 2007006 *(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 25, 2008
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 corrective action program (CAP). Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluations for issues classified as significant adverse conditions were comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. However, the team identified a few examples where corrective actions have not been entirely effective.

The team determined that thresholds for identifying issues were appropriately low. 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.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors did not identify any reluctance to report safety concerns. The team concluded that the employee concerns program (ECP) was functioning as intended.

Inspection Report# : 2008007 *(pdf)*

Last modified : August 29, 2008

Crystal River 3
3Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Control Transient Combustibles
The team identified a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C.(9), for the licensee's failure to properly implement fire protection program procedures for control of transient combustible materials. Specifically, transient combustible materials were left unattended for four days in the 3B 480V ES Switchgear Room after work had been completed, which was a violation of the licensee's administrative procedures for control of transient combustibles. Once identified, the licensee removed the combustible materials and initiated a nuclear condition report to address the issue.

The finding is more than minor because the transient combustible materials presented a credible fire scenario involving equipment important to safety, which degraded the reactor safety Initiating Events cornerstone objective to limit the likelihood of those events that may upset plant stability and challenge critical safety functions. The amount of unattended transient combustible materials did not violate the licensee's transient combustible control limits for the fire area. Therefore, the finding was assigned a low degradation rating against the combustible controls program. The finding was of very low safety significance (Green) based on the low degradation rating. This finding has a cross-cutting aspect in the Work Practices component of the Human Performance area because the licensee failed to effectively communicate expectations regarding procedural compliance and personnel following procedures (NRC Inspection Manual Chapter 0305, H.4(b)).
Inspection Report# : 2008006 *(pdf)*

Significance: G Feb 15, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Reactor Coolant Pump 1B Lube Oil Collection System Leakage
A self-revealing non-cited violation of 10 CFR 50, Appendix R, Section III.O, was identified for failure of the reactor coolant pump (RCP) oil collection system to collect and drain RCP oil leakage to a vented closed container. Specifically, the licensee found an estimated one to two gallons of oil on the reactor building floor beneath RCP-1B. The licensee initiated a nuclear condition report for this issue.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Initiating Events cornerstone objective. The team completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding was of very low safety significance (Green) because the amount of oil identified in 2008 was bounded by the licensee's 2004 analysis (which assumed a 21 gallon oil leak). This finding has a cross-cutting aspect in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee did not take appropriate corrective actions in a timely manner to address the adverse trend related to oil leakage for RCP-1B (NRC Inspection Manual Chapter 0305, P.1(d)).

Inspection Report# : 2008006 *(pdf)*

Significance: G Oct 29, 2007
Identified By: NRC
Item Type: FIN Finding
Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient
A self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control switch during maintenance activities. As a result of bumping the control switch, a condensate pump had to be secured and reactor power was rapidly reduced to 61 percent to prevent a reactor trip. Corrective actions included removing the control switch handle to prevent it from being bumped.

The finding was more than minor since it affected the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenged critical safety functions. The inspectors referenced Inspection manual Chapter 0609.04, Significance Determination process (SDP), Phase 1 screening and determined the finding to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. A contributing cause of this finding is related to the crosscutting area of human performance, with a work control component. Specifically, the licensee did not adequately plan work activities to protect the condensate pump control switch from being bumped.
Inspection Report# : 2008002 *(pdf)*

Mitigating Systems

Significance: G Feb 22, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inoperable Fire Penetration Seal
The inspectors identified a Green non-cited violation (NCV) of Crystal River Unit 3 Operating License Condition 2.C(9), Fire Protection Program. The NCV was associated with an inoperable fire penetration seal in the 3-hour fire rated ceiling of the makeup system valve alley. The licensee declared the penetration seal inoperable. Corrective actions included establishing an hourly fire watch and repairing the penetration to its designed condition.

The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater 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. Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the finding was determined to have a very low safety significance since the gap in the fire penetration seal was small (less than 1/8 inch in width).

Inspection Report# : 2008002 *(pdf)*

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Protect Cables for Valve DHV-42
The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.2., for failure to protect cables from fire damage for components required for safe shutdown. Specifically, the Mecatiss MTS-3 fire wrap installed around the cables for valve DHV-42 (suction from the reactor building sump to the Train A decay heat pump) was not installed in accordance with the vendor's tested configuration. The licensee initiated a nuclear condition report and implemented an hourly roving fire watch to address this issue. Additionally, the licensee implemented repairs during the March 2008 forced outage to upgrade the Mecatiss MTS-3 fire wrap to comply with the vendor tested configuration.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Mitigating Systems cornerstone objective. The inspectors completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding, when given credit for the fixed automatic suppression system in the area, was of very low safety significance (Green).

Inspection Report# : 2008006 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedural Guidance Associated with Removal of Containment Debris
The inspectors identified a non-cited violation (NCV) of Improved Technical Specification 5.6.1.1.a, for failure to adequately implement procedures required by Regulatory Guide 1.33, Appendix A, Section 3, Procedures for Startup, Operation, and Shutdown of Safety-Related PWR Systems. Specifically, the licensee failed to verify no latent debris was present in containment that could impact the emergency core cooling system (ECCS) sump. Corrective actions completed include: removal of the debris identified by the inspectors and performing additional inspection and cleaning of containment.

The finding is more than minor because it could be reasonably viewed as a precursor to a significant event involving debris accumulation on the containment sump screens which could cause impairment to ECCS recirculation flow during a design basis loss of coolant accident. The inspectors referenced Inspection Manual Chapter 0609, Significance Determination Process (SDP), Phase 1 screening and determined the finding to be of very low safety significance. Although the debris impacted the mitigating system cornerstone, it was unlikely to have resulted in an actual loss of safety function and was not potentially risk significant due to possible external events. A contributing cause of this finding is related to the crosscutting area of Human Performance, specifically Work Practices in that the licensee did not adequately comply with a containment inspection procedure. (IMC 305, H.4(b))

Inspection Report# : 2007005 *(pdf)*

Significance: G Oct 05, 2007

Identified By: NRC
Item Type: NCV NonCited Violation
Violation of Technical Specification 5.6.1 for Failure to Implement an Adequate Procedure for Manual Starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) Following a LBLOCA
The inspectors identified a finding of very low safety significance involving a violation of Technical Specifications (TS) 5.6.1 for failure to implement an adequate procedure for manual starting of the Control Complex Chilled Water Chiller Units (CHHE-1A/1B) following a Large Break Loss of Coolant Accident (LBLOCA). The chiller units are required to be restarted prior to 127 minutes after the accident to ensure adequate cooling to components within the control complex.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of CHHE-1A/1B to perform the intended safety function during a design basis event. The vendor for CHHE-1A/1B provided a maximum temperature for restarting the chiller units of 104 degrees Fahrenheit (°F). The basis for this limitation is to prevent an inadvertent chiller unit trip due to high chiller freon condenser pressure. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found that Nuclear Services Closed Cycle Cooling (SW) temperature falls below 104 °F no later than 84 minutes after a LBLOCA. This affords operators at least 40 minutes to successfully restart the chiller units. This issue is documented in the corrective action program as nuclear condition report (NCR) 247908. This finding was reviewed for cross-cutting aspects and none were identified.(Section 1R21.2.3
Inspection Report# : 2007006 *(pdf)*

Significance: G Oct 05, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Violation of 10 CFR 50, Appendix B, Criterion XI for Failure to Account for Instrument Uncertainty During EFP-2 Testing
The inspectors identified a finding of very low safety significance involving a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, for failure to implement a test program which accounted for the effects of instrument uncertainty on surveillance testing of Emergency Feedwater Pump (EFP)-2 in accordance with the approved In-service Testing (IST) program.

This finding is more than minor because it affects the Procedure Quality attribute of the Mitigating Systems Cornerstone. It impacts the cornerstone objective of ensuring the availability, reliability, and operability of EFP-2 to perform the intended safety function during a design basis event. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of in-service failures of EFP-2 rendering safety-related equipment inoperative. This issue is documented in the corrective actions program as NCR 248036. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.7)

Inspection Report# : 2007006 *(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 25, 2008
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 corrective action program (CAP). Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluations for issues classified as significant adverse conditions were comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. However, the team identified a few examples where corrective actions have not been entirely effective.

The team determined that thresholds for identifying issues were appropriately low. 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.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors did not identify any reluctance to report safety concerns. The team concluded that the employee concerns program (ECP) was functioning as intended.

Inspection Report# : 2008007 *(pdf)*

Last modified : November 26, 2008

Crystal River 3
4Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2008
Identified By: NRC
Item Type: FIN Finding
Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient
A self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control switch during maintenance activities. As a result of bumping the control switch, a condensate pump had to be secured and reactor power was rapidly reduced to 61 percent to prevent a reactor trip. Corrective actions included removing the control switch handle to prevent it from being bumped.

The finding was more than minor since it affected the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenged critical safety functions. The inspectors referenced Inspection manual Chapter 0609.04, Significance Determination process (SDP), Phase 1 screening and determined the finding to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. A contributing cause of this finding is related to the crosscutting area of human performance, with a work control component. Specifically, the licensee did not adequately plan work activities to protect the condensate pump control switch from being bumped. (Section 4OA2.2)

Inspection Report# : 2008002 *(pdf)*

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Control Transient Combustibles
The team identified a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C.(9), for the licensee's failure to properly implement fire protection program procedures for control of transient combustible materials. Specifically, transient combustible materials were left unattended for four days in the 3B 480V ES Switchgear Room after work had been completed, which was a violation of the licensee's administrative procedures for control of transient combustibles. Once identified, the licensee removed the combustible materials and initiated a nuclear condition report to address the issue.

The finding is more than minor because the transient combustible materials presented a credible fire scenario involving equipment important to safety, which degraded the reactor safety Initiating Events cornerstone objective to limit the likelihood of those events that may upset plant stability and challenge critical safety functions. The amount of unattended transient combustible materials did not violate the licensee's transient combustible control limits for the fire area. Therefore, the finding was assigned a low degradation rating against the combustible controls program. The finding was of very low safety significance (Green) based on the low degradation rating. This finding has a cross-cutting aspect in the Work Practices component of the Human Performance area because the licensee failed to effectively communicate expectations regarding procedural compliance and personnel following procedures (NRC Inspection Manual Chapter 0305, H.4(b)).
Inspection Report# : 2008006 *(pdf)*

Significance: G Feb 15, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation

Reactor Coolant Pump 1B Lube Oil Collection System Leakage
A self-revealing non-cited violation of 10 CFR 50, Appendix R, Section III.O, was identified for failure of the reactor coolant pump (RCP) oil collection system to collect and drain RCP oil leakage to a vented closed container. Specifically, the licensee found an estimated one to two gallons of oil on the reactor building floor beneath RCP-1B. The licensee initiated a nuclear condition report for this issue.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Initiating Events cornerstone objective. The team completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding was of very low safety significance (Green) because the amount of oil identified in 2008 was bounded by the licensee's 2004 analysis (which assumed a 21 gallon oil leak). This finding has a cross-cutting aspect in the Corrective Action Program component of the Problem Identification and Resolution area because the licensee did not take appropriate corrective actions in a timely manner to address the adverse trend related to oil leakage for RCP-1B (NRC Inspection Manual Chapter 0305, P.1(d)).

Inspection Report# : 2008006 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inoperable Fire Penetration Seal
The inspectors identified a Green non-cited violation (NCV) of Crystal River Unit 3 Operating License Condition 2.C (9), Fire Protection Program. The NCV was associated with an inoperable fire penetration seal in the 3-hour fire rated ceiling of the makeup system valve alley. The licensee declared the penetration seal inoperable. Corrective actions included establishing an hourly fire watch and repairing the penetration to its designed condition.

The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater 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. Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the finding was determined to have a very low safety significance since the gap in the fire penetration seal was small (less than 1/8 inch in width).

Inspection Report# : 2008002 *(pdf)*

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Protect Cables for Valve DHV-42
The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.2., for failure to protect cables from fire damage for components required for safe shutdown. Specifically, the Mecatiss MTS-3 fire wrap installed around the cables for valve DHV-42 (suction from the reactor building sump to the Train A decay heat pump) was not installed in accordance with the vendor's tested configuration. The licensee initiated a nuclear condition report and implemented an hourly roving fire watch to address this issue. Additionally, the licensee implemented repairs during the March 2008 forced outage to upgrade the Mecatiss MTS-3 fire wrap to comply with the vendor tested configuration.

This finding is more than minor because it is associated with the external factors attribute, i.e., fire, and it degraded the reactor safety Mitigating Systems cornerstone objective. The inspectors completed a Phase 1 screening of the finding in accordance with IMC 0609, Appendix F, Attachment 1, Step 1.3, Qualitative Screening Approach, and concluded that the finding, when given credit for the fixed automatic suppression system in the area, was of very low safety significance (Green).

Inspection Report# : 2008006 *(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 25, 2008
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 corrective action program (CAP). Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluations for issues classified as significant adverse conditions were comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. However, the team identified a few examples where corrective actions have not been entirely effective.

The team determined that thresholds for identifying issues were appropriately low. 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.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors did not identify any reluctance to report safety concerns. The team concluded that the employee concerns program (ECP) was functioning as intended.

Inspection Report# : 2008007 *(pdf)*

Last modified : April 07, 2009

Crystal River 3
1Q/2009 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip
A self-revealing finding was identified for the failure to follow procedure HUM-NGGC-0001, Human Performance Program, which required workers to perform self and peer checks to ensure the correct action is performed on the correct component. Specifically, during meter calibration activities, workers performing voltage checks failed to perform adequate self and peer checks when connecting test equipment. As a result, incorrect test equipment was connected resulting in blown fuses, the loss of several secondary plant pumps, and ultimately a manual plant trip. Corrective actions include: move relay work identified in the extent of condition review from on-line to outage to prevent recurrence, revise maintenance procedures associated with calibration of meters and relays to incorporate human factoring from lessons learned from this event, and perform an analysis of and incorporate best practices in procedures regarding how plant risk is assessed for activities that could cause transients.

The finding was more than minor since it affected the human performance attribute of the Initiating Event Cornerstone and resulted in an event that upset plant stability. Specifically, the failure to properly utilize human performance tools such as self and peer checking as specified in HUM-GGC-0001, Revision 2, resulted in the connection of incorrect test equipment, the loss of several secondary plant pumps and ultimately led to a manual reactor trip. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance with a work practices aspect (H.4(a)). Specifically, workers did not utilize proper self and peer checking.

Inspection Report# : 2009002 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Have Adequate Controls in Place to Ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability
A self-revealing finding was identified for failing to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support emergency diesel generator (EGDG) operability. As a result, during cold weather conditions, licensee personnel did not close an access door which caused a low EGDG-1B lube oil temperature condition and inoperability of the EGDG. Corrective actions include: posting signs on all external doors of both safety and non-safety EGDGs rooms indicating that the doors should not be left open, discussing the event with site personnel; and initiation of changes to the site’s cold weather checklist to check closed EGDG room doors during cold weather conditions.

The finding was more than minor since it affected the equipment availability attribute of the Mitigating System

Cornerstone and resulted in an unavailable emergency diesel generator train for approximately 13 hours. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it was not a design or qualification deficiency, did not result in a loss of a system safety function, did not result in an actual loss of safety function of a single train for greater than allowed by improved technical specifications (ITS), did not represent an actual loss of safety function of risk-significant, non-technical specification equipment, and did not screen as risk significant due to external events. The inspectors found that the cause of this finding was not reflective of current performance since the EGDG door lacked the proper signage since initial plant operation. Therefore, a cross-cutting aspect was not assigned.

Inspection Report# : 2009002 *(pdf)*

Significance: G Mar 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Timely and Effective Corrective Actions Resulted in a Repeat Failure of a Main Feedwater Isolation Valve due to Magnesium Rotor Oxidation/Corrosion
The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for failure to take timely and effective corrective actions to prevent a second failure of a main feedwater isolation valve (MFIV) due to corrosion of the valve actuator's magnesium rotor. Specifically, corrective actions associated with a similar failure of a MFIV in 2005 were not enhanced when additional information became available through NRC Information Notice (IN) 2006-026, Failure of Magnesium Rotors in Motor-Operator Valve Actuators. As a result, in December 2008, a MFIV failed to operate due to magnesium rotor degradation. Corrective actions for the failure of FWV-30 include: installation of a new motor; development and implementation of engineering changes to replace the station's motor-operated valve (MOV) magnesium rotor motors with aluminum rotor motors (when available); ensuring the engineering staff is trained on effective correction action plans; and revision of MOV maintenance procedures to include information obtained from IN 2006-026 prior to the next MOV inspections.

The finding was more than minor because it affected the equipment availability attribute of the Mitigating System cornerstone and resulted in a MFIV being inoperable for a period of time greater than allowed by ITS. Since the valve would not have performed its safety function for greater than the ITS' allowed outage time, a SDP Phase 2 analysis was required. Based upon the Phase 2 results, a regional senior reactor analyst performed a Phase 3 evaluation. The Phase 3 evaluation concluded that the finding was of very low safety significance (Green). A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution with an operating experience component (P.2(b)). Specifically, the licensee did not implement and institutionalize, in a timely manner, IN 2006-26 in station procedures and training programs associated with magnesium rotor inspections.

Inspection Report# : 2009002 *(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 25, 2008
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 corrective action program (CAP). Evaluation of issues was generally comprehensive and technically adequate. Formal root cause evaluations for issues classified as significant adverse conditions were comprehensive and detailed. Overall, corrective actions developed and implemented for issues were effective in correcting the problems. However, the team identified a few examples where corrective actions have not been entirely effective.

The team determined that thresholds for identifying issues were appropriately low. 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.

Based on discussions and interviews conducted with plant employees from various departments, the inspectors did not identify any reluctance to report safety concerns. The team concluded that the employee concerns program (ECP) was functioning as intended.

Inspection Report# : 2008007 *(pdf)*

Last modified : May 28, 2009

Crystal River 3
2Q/2009 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip
A self-revealing finding was identified for the failure to follow procedure HUM-NGGC-0001, Human Performance Program, which required workers to perform self and peer checks to ensure the correct action is performed on the correct component. Specifically, during meter calibration activities, workers performing voltage checks failed to perform adequate self and peer checks when connecting test equipment. As a result, incorrect test equipment was connected resulting in blown fuses, the loss of several secondary plant pumps, and ultimately a manual plant trip. Corrective actions include: move relay work identified in the extent of condition review from on-line to outage to prevent recurrence, revise maintenance procedures associated with calibration of meters and relays to incorporate human factoring from lessons learned from this event, and perform an analysis of and incorporate best practices in procedures regarding how plant risk is assessed for activities that could cause transients.

The finding was more than minor since it affected the human performance attribute of the Initiating Event Cornerstone and resulted in an event that upset plant stability. Specifically, the failure to properly utilize human performance tools such as self and peer checking as specified in HUM-GGC-0001, Revision 2, resulted in the connection of incorrect test equipment, the loss of several secondary plant pumps and ultimately led to a manual reactor trip. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance with a work practices aspect (H.4(a)). Specifically, workers did not utilize proper self and peer checking.

Inspection Report# : 2009002 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Have Adequate Controls in Place to Ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability
A self-revealing finding was identified for failing to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support emergency diesel generator (EGDG) operability. As a result, during cold weather conditions, licensee personnel did not close an access door which caused a low EGDG-1B lube oil temperature condition and inoperability of the EGDG. Corrective actions include: posting signs on all external doors of both safety and non-safety EGDGs rooms indicating that the doors should not be left open, discussing the event with site personnel; and initiation of changes to the site's cold weather checklist to check closed EGDG room doors during cold weather conditions.

The finding was more than minor since it affected the equipment availability attribute of the Mitigating System

Cornerstone and resulted in an unavailable emergency diesel generator train for approximately 13 hours. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it was not a design or qualification deficiency, did not result in a loss of a system safety function, did not result in an actual loss of safety function of a single train for greater than allowed by improved technical specifications (ITS), did not represent an actual loss of safety function of risk-significant, non-technical specification equipment, and did not screen as risk significant due to external events. The inspectors found that the cause of this finding was not reflective of current performance since the EGDG door lacked the proper signage since initial plant operation. Therefore, a cross-cutting aspect was not assigned.

Inspection Report# : 2009002 *(pdf)*

Significance: G Mar 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Timely and Effective Corrective Actions Resulted in a Repeat Failure of a Main Feedwater Isolation Valve due to Magnesium Rotor Oxidation/Corrosion
The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for failure to take timely and effective corrective actions to prevent a second failure of a main feedwater isolation valve (MFIV) due to corrosion of the valve actuator's magnesium rotor. Specifically, corrective actions associated with a similar failure of a MFIV in 2005 were not enhanced when additional information became available through NRC Information Notice (IN) 2006-026, Failure of Magnesium Rotors in Motor-Operator Valve Actuators. As a result, in December 2008, a MFIV failed to operate due to magnesium rotor degradation. Corrective actions for the failure of FWV-30 include: installation of a new motor; development and implementation of engineering
changes to replace the station's motor-operated valve (MOV) magnesium rotor motors with aluminum rotor motors (when available); ensuring the engineering staff is trained on effective correction action plans; and revision of MOV maintenance procedures to include information obtained from IN 2006-026 prior to the next MOV inspections.

The finding was more than minor because it affected the equipment availability attribute of the Mitigating System cornerstone and resulted in a MFIV being inoperable for a period of time greater than allowed by ITS. Since the valve would not have performed its safety function for greater than the ITS' allowed outage time, a SDP Phase 2 analysis was required. Based upon the Phase 2 results, a regional senior reactor analyst performed a Phase 3 evaluation. The Phase 3 evaluation concluded that the finding was of very low safety significance (Green). A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution with an operating experience component (P.2(b)). Specifically, the licensee did not implement and institutionalize, in a timely manner, IN 2006-26 in station procedures and training programs associated with magnesium rotor inspections.

Inspection Report# : 2009002 *(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

Last modified : August 31, 2009

Crystal River 3
3Q/2009 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip
A self-revealing finding was identified for the failure to follow procedure HUM-NGGC-0001, Human Performance Program, which required workers to perform self and peer checks to ensure the correct action is performed on the correct component. Specifically, during meter calibration activities, workers performing voltage checks failed to perform adequate self and peer checks when connecting test equipment. As a result, incorrect test equipment was connected resulting in blown fuses, the loss of several secondary plant pumps, and ultimately a manual plant trip. Corrective actions include: move relay work identified in the extent of condition review from on-line to outage to prevent recurrence, revise maintenance procedures associated with calibration of meters and relays to incorporate human factoring from lessons learned from this event, and perform an analysis of and incorporate best practices in procedures regarding how plant risk is assessed for activities that could cause transients.

The finding was more than minor since it affected the human performance attribute of the Initiating Event Cornerstone and resulted in an event that upset plant stability. Specifically, the failure to properly utilize human performance tools such as self and peer checking as specified in HUM-GGC-0001, Revision 2, resulted in the connection of incorrect test equipment, the loss of several secondary plant pumps and ultimately led to a manual reactor trip. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance with a work practices aspect (H.4(a)). Specifically, workers did not utilize proper self and peer checking.

Inspection Report# : 2009002 *(pdf)*

Mitigating Systems

Significance: G 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)*

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Have Adequate Controls in Place to Ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability
A self-revealing finding was identified for failing to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support emergency diesel generator (EGDG) operability. As a result, during cold weather conditions, licensee personnel did not close an access door which caused a low EGDG-1B lube oil temperature condition and inoperability of the EGDG. Corrective actions include: posting signs on all external doors of both safety and non-safety EGDGs rooms indicating that the doors should not be left open, discussing the event with site personnel; and initiation of changes to the site's cold weather checklist to check closed EGDG room doors during cold weather conditions.

The finding was more than minor since it affected the equipment availability attribute of the Mitigating System Cornerstone and resulted in an unavailable emergency diesel generator train for approximately 13 hours. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it was not a design or qualification deficiency, did not result in a loss of a system safety function, did not result in an actual loss of safety function of a single train for greater than allowed by improved technical specifications (ITS), did not represent an actual loss of safety function of risk-significant, non-technical specification equipment, and did not screen as risk significant due to external events. The inspectors found that the cause of this finding was not reflective of current performance since the EGDG door lacked the proper signage since initial plant operation. Therefore, a cross-cutting aspect was not assigned.

Inspection Report# : 2009002 *(pdf)*

Significance: G Mar 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Timely and Effective Corrective Actions Resulted in a Repeat Failure of a Main Feedwater Isolation Valve due to Magnesium Rotor Oxidation/Corrosion
The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for failure to take timely and effective corrective actions to prevent a second failure of a main feedwater isolation valve (MFIV) due to corrosion of the valve actuator's magnesium rotor. Specifically, corrective actions associated with a similar failure of a MFIV in 2005 were not enhanced when additional information became available through NRC Information Notice (IN) 2006-026, Failure of Magnesium Rotors in Motor-Operator Valve Actuators. As a result, in December 2008, a MFIV failed to operate due to magnesium rotor degradation. Corrective actions for the failure of FWV-30 include: installation of a new motor; development and implementation of engineering
changes to replace the station's motor-operated valve (MOV) magnesium rotor motors with aluminum rotor motors (when available); ensuring the engineering staff is trained on effective correction action plans; and revision of MOV maintenance procedures to include information obtained from IN 2006-026 prior to the next MOV inspections.

The finding was more than minor because it affected the equipment availability attribute of the Mitigating System cornerstone and resulted in a MFIV being inoperable for a period of time greater than allowed by ITS. Since the valve would not have performed its safety function for greater than the ITS' allowed outage time, a SDP Phase 2 analysis was required. Based upon the Phase 2 results, a regional senior reactor analyst performed a Phase 3 evaluation. The Phase 3 evaluation concluded that the finding was of very low safety significance (Green). A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution with an operating experience component (P.2(b)). Specifically, the licensee did not implement and institutionalize, in a timely manner, IN 2006-26 in station procedures and training programs associated with magnesium rotor inspections.

Inspection Report# : 2009002 *(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

Last modified : December 10, 2009

Crystal River 3
4Q/2009 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)*

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip
A self-revealing finding was identified for the failure to follow procedure HUM-NGGC-0001, Human Performance Program, which required workers to perform self and peer checks to ensure the correct action is performed on the correct component. Specifically, during meter calibration activities, workers performing voltage checks failed to perform adequate self and peer checks when connecting test equipment. As a result, incorrect test equipment was connected resulting in blown fuses, the loss of several secondary plant pumps, and ultimately a manual plant trip. Corrective actions include: move relay work identified in the extent of condition review from on-line to outage to prevent recurrence, revise maintenance procedures associated with calibration of meters and relays to incorporate human factoring from lessons learned from this event, and perform an analysis of and incorporate best practices in procedures regarding how plant risk is assessed for activities that could cause transients.

The finding was more than minor since it affected the human performance attribute of the Initiating Event Cornerstone and resulted in an event that upset plant stability. Specifically, the failure to properly utilize human performance tools such as self and peer checking as specified in HUM-GGC-0001, Revision 2, resulted in the connection of incorrect test equipment, the loss of several secondary plant pumps and ultimately led to a manual reactor trip. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. The cause of the finding is related to the cross-cutting area of Human Performance with a work practices aspect (H.4(a)). Specifically, workers did not utilize proper self and peer checking.

Inspection Report# : 2009002 *(pdf)*

Mitigating Systems

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)*

Significance: G 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)*

Significance: G Mar 31, 2009
Identified By: Self-Revealing
Item Type: FIN Finding
Failure to Have Adequate Controls in Place to Ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability
A self-revealing finding was identified for failing to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support emergency diesel generator (EGDG) operability. As a result, during cold weather conditions, licensee personnel did not close an access door which caused a low EGDG-1B lube oil temperature condition and inoperability of the EGDG. Corrective actions include: posting signs on all external doors of both safety and non-safety EGDGs rooms indicating that the doors should not be left open, discussing the event with site personnel; and initiation of changes to the site's cold weather checklist to check closed EGDG room doors during cold weather conditions.

The finding was more than minor since it affected the equipment availability attribute of the Mitigating System Cornerstone and resulted in an unavailable emergency diesel generator train for approximately 13 hours. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green) since it was not a design or qualification deficiency, did not result in a loss of a system safety function, did not result in an actual loss of safety function of a single train for greater than allowed by improved technical specifications (ITS), did not represent an actual loss of safety function of risk-significant, non-technical specification equipment, and did not screen as risk significant due to external events. The inspectors found that the cause of this finding was not reflective of current performance since the EGDG door lacked the proper signage since initial plant operation. Therefore, a cross-cutting aspect was not assigned.

Inspection Report# : 2009002 *(pdf)*

Significance: G Mar 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Take Timely and Effective Corrective Actions Resulted in a Repeat Failure of a Main Feedwater Isolation Valve due to Magnesium Rotor Oxidation/Corrosion
The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for failure to take timely and effective corrective actions to prevent a second failure of a main feedwater isolation valve (MFIV) due to corrosion of the valve actuator's magnesium rotor. Specifically, corrective actions associated with a similar failure of a MFIV in 2005 were not enhanced when additional information became available through NRC Information Notice (IN) 2006-026, Failure of Magnesium Rotors in Motor-Operator Valve Actuators. As a result, in December 2008, a MFIV failed to operate due to magnesium rotor degradation. Corrective actions for the failure of FWV-30 include: installation of a new motor; development and implementation of engineering changes to replace the station's motor-operated valve (MOV) magnesium rotor motors with aluminum rotor motors (when available); ensuring the engineering staff is trained on effective correction action plans; and revision of MOV maintenance procedures to include information obtained from IN 2006-026 prior to the next MOV inspections.

The finding was more than minor because it affected the equipment availability attribute of the Mitigating System cornerstone and resulted in a MFIV being inoperable for a period of time greater than allowed by ITS. Since the valve would not have performed its safety function for greater than the ITS' allowed outage time, a SDP Phase 2 analysis was required. Based upon the Phase 2 results, a regional senior reactor analyst performed a Phase 3 evaluation. The Phase 3 evaluation concluded that the finding was of very low safety significance (Green). A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution with an operating experience component (P.2(b)). Specifically, the licensee did not implement and institutionalize, in a timely manner, IN 2006-26 in station procedures and training programs associated with magnesium rotor inspections.

Inspection Report# : 2009002 *(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

Last modified : March 01, 2010

Crystal River 3
1Q/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: 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)*

Significance: G 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

Last modified : May 26, 2010

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: 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)*

Significance: G 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

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)*

Last modified : November 29, 2010

Crystal River 3
4Q/2010 Plant Inspection Findings

Initiating Events

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 Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Preconditioning of Safety-Related Air Operated Valves
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control for preconditioning of a safety-related air operated valve prior to surveillance testing. The licensee entered this deficiency into their corrective action program for resolution.

The licensee’s preconditioning of air operated valves prior to performing as-found testing is a performance deficiency. This finding is more than minor because if left uncorrected the performance deficiency has the potential to lead to a more significant safety concern in that safety-related valve performance deficiencies could be masked. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. The finding involved the cross-cutting aspect of complete and accurate procedures under the Resources component of the Human Performance area [H.2(c)]. [Section 1R21.2.2]

Inspection Report# : 2010007 *(pdf)*

Significance: G Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Incorporate Requirements of Recovering from a Station Blackout into Calculations
The team identified a non-cited violation of 10 CFR 50.63, Loss of all alternating current power, for failure to ensure Regulatory Guide 1.155, Station Blackout commitments were implemented in calculations for restoring off-site power. The licensee entered this deficiency into their corrective action program for resolution.

The licensee’s failure to maintain calculations to assure adequate voltage for the remote closing of switchyard breakers during a station blackout event is a performance deficiency. The team determined that the finding is more than minor because it adversely affected the design control attribute of the mitigating reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.13]

Inspection Report# : 2010007 *(pdf)*

Significance: G Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Monitor the Service Water and Decay Cooling Expansion Tank Check Valves
The team identified a non-cited violation of 10 CFR 50.65(a)(1) for the licensee’s failure to monitor service water and decay heat cooling expansion tank level indicator check valves. In response to this concern, the licensee closed the isolation valves as an interim action, performed an in situ check valve test with satisfactory results, and entered the deficiency into their corrective action program for resolution.

The licensee’s failure to perform appropriate maintenance on the check valves was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.17]

Inspection Report# : 2010007 *(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)*

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)*

Last modified : March 03, 2011

Crystal River 3
1Q/2011 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Mar 31, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
Operating Crew Failures on the 2011 Annual Requalification Operating Test
A self-revealing Green finding, associated with operating crew performance on the simulator during facility-administered requalification examination was identified. Two of the eight crews evaluated failed to pass their simulator examinations. As immediate corrective action, the failed operating crews were remediated (i.e., the operating crews were re-trained and successfully retested) prior to returning to shift. The licensee has entered this issue into the corrective action program as Nuclear Condition Report (NRC) 450196.

The inspectors determined that the crew failures constituted a performance deficiency based on the fact that licensed operators are expected to operate the plant with acceptable standards of knowledge and abilities demonstrated through periodic testing as required by 10 CFR 55.59(a)(2). Two out of eight crews of licensed operators failed to demonstrate a satisfactory understanding of the required actions and mitigating strategies required to safely operate the facility under normal, abnormal, and emergency conditions. The finding is greater than minor because the performance deficiency potentially affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding reflected the crew's potential inability to take timely actions in response to actual abnormal and emergency conditions. The cause of this finding was directly related to the cross-cutting aspect of personnel training and qualifications in the Resources component of the Human Performance area, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b)) (Section 1R11)

Inspection Report# : 2011002 *(pdf)*

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: 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: Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Preconditioning of Safety-Related Air Operated Valves
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control for preconditioning of a safety-related air operated valve prior to surveillance testing. The licensee entered this deficiency into their corrective action program for resolution.

The licensee's preconditioning of air operated valves prior to performing as-found testing is a performance deficiency. This finding is more than minor because if left uncorrected the performance deficiency has the potential to lead to a more significant safety concern in that safety-related valve performance deficiencies could be masked. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. The finding involved the cross-cutting aspect of complete and accurate procedures under the Resources component of the Human Performance area [H.2(c)]. [Section 1R21.2.2]

Inspection Report# : 2010007 *(pdf)*

Significance: Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Incorporate Requirements of Recovering from a Station Blackout into Calculations
The team identified a non-cited violation of 10 CFR 50.63, Loss of all alternating current power, for failure to ensure Regulatory Guide 1.155, Station Blackout commitments were implemented in calculations for restoring off-site power. The licensee entered this deficiency into their corrective action program for resolution.

The licensee's failure to maintain calculations to assure adequate voltage for the remote closing of switchyard breakers during a station blackout event is a performance deficiency. The team determined that the finding is more than minor because it adversely affected the design control attribute of the mitigating reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.13]

Inspection Report# : 2010007 *(pdf)*

Significance: G Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Monitor the Service Water and Decay Cooling Expansion Tank Check Valves
The team identified a non-cited violation of 10 CFR 50.65(a)(1) for the licensee's failure to monitor service water and decay heat cooling expansion tank level indicator check valves. In response to this concern, the licensee closed the isolation valves as an interim action, performed an in situ check valve test with satisfactory results, and entered the deficiency into their corrective action program for resolution.

The licensee's failure to perform appropriate maintenance on the check valves was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.17]

Inspection Report# : 2010007 *(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)*

Crystal River 3
2Q/2011 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Mar 31, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
Operating Crew Failures on the 2011 Annual Requalification Operating Test
A self-revealing Green finding, associated with operating crew performance on the simulator during facility-administered requalification examination was identified. Two of the eight crews evaluated failed to pass their simulator examinations. As immediate corrective action, the failed operating crews were remediated (i.e., the operating crews were re-trained and successfully retested) prior to returning to shift. The licensee has entered this issue into the corrective action program as Nuclear Condition Report (NRC) 450196.

The inspectors determined that the crew failures constituted a performance deficiency based on the fact that licensed operators are expected to operate the plant with acceptable standards of knowledge and abilities demonstrated through periodic testing as required by 10 CFR 55.59(a)(2). Two out of eight crews of licensed operators failed to demonstrate a satisfactory understanding of the required actions and mitigating strategies required to safely operate the facility under normal, abnormal, and emergency conditions. The finding is greater than minor because the performance deficiency potentially affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding reflected the crew's potential inability to take timely actions in response to actual abnormal and emergency conditions. The cause of this finding was directly related to the cross-cutting aspect of personnel training and qualifications in the Resources component of the Human Performance area, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b)) (Section 1R11)

Inspection Report# : 2011002 *(pdf)*

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 Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Preconditioning of Safety-Related Air Operated Valves
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control for preconditioning of a safety-related air operated valve prior to surveillance testing. The licensee entered this deficiency into their corrective action program for resolution.

The licensee's preconditioning of air operated valves prior to performing as-found testing is a performance deficiency. This finding is more than minor because if left uncorrected the performance deficiency has the potential to lead to a more significant safety concern in that safety-related valve performance deficiencies could be masked. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. The finding involved the cross-cutting aspect of complete and accurate procedures under the Resources component of the Human Performance area [H.2(c)]. [Section 1R21.2.2]

Inspection Report# : 2010007 *(pdf)*

Significance: G Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Incorporate Requirements of Recovering from a Station Blackout into Calculations
The team identified a non-cited violation of 10 CFR 50.63, Loss of all alternating current power, for failure to ensure Regulatory Guide 1.155, Station Blackout commitments were implemented in calculations for restoring off-site power. The licensee entered this deficiency into their corrective action program for resolution.

The licensee's failure to maintain calculations to assure adequate voltage for the remote closing of switchyard breakers during a station blackout event is a performance deficiency. The team determined that the finding is more than minor because it adversely affected the design control attribute of the mitigating reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.13]

Inspection Report# : 2010007 *(pdf)*

Significance: G Aug 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Monitor the Service Water and Decay Cooling Expansion Tank Check Valves
The team identified a non-cited violation of 10 CFR 50.65(a)(1) for the licensee's failure to monitor service water and decay heat cooling expansion tank level indicator check valves. In response to this concern, the licensee closed the isolation valves as an interim action, performed an in situ check valve test with satisfactory results, and entered the deficiency into their corrective action program for resolution.

The licensee's failure to perform appropriate maintenance on the check valves was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) using the SDP because it did not represent a loss of system or safety function. A cross-cutting aspect was not identified because the finding does not represent current performance. [Section 1R21.2.17]

Inspection Report# : 2010007 *(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

Last modified : October 14, 2011

Crystal River 3
3Q/2011 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Mar 31, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
Operating Crew Failures on the 2011 Annual Requalification Operating Test
A self-revealing Green finding, associated with operating crew performance on the simulator during facility-administered requalification examination was identified. Two of the eight crews evaluated failed to pass their simulator examinations. As immediate corrective action, the failed operating crews were remediated (i.e., the operating crews were re-trained and successfully retested) prior to returning to shift. The licensee has entered this issue into the corrective action program as Nuclear Condition Report (NRC) 450196.

The inspectors determined that the crew failures constituted a performance deficiency based on the fact that licensed operators are expected to operate the plant with acceptable standards of knowledge and abilities demonstrated through periodic testing as required by 10 CFR 55.59(a)(2). Two out of eight crews of licensed operators failed to demonstrate a satisfactory understanding of the required actions and mitigating strategies required to safely operate the facility under normal, abnormal, and emergency conditions. The finding is greater than minor because the performance deficiency potentially affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding reflected the crew's potential inability to take timely actions in response to actual abnormal and emergency conditions. The cause of this finding was directly related to the cross-cutting aspect of personnel training and qualifications in the Resources component of the Human Performance area, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b)) (Section 1R11)

Inspection Report# : 2011002 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: W Jul 15, 2011
Identified By: NRC
Item Type: AV Apparent Violation
Failure to Maintain a Standard EAL Scheme
TBD. An AV was identified for failure to follow and maintain in effect emergency plans which use a standard emergency classification and action level scheme. Specifically, the licensee's emergency plan emergency action level (EAL) 1.4, General Emergency - Gaseous Effluent, specified instrument values that were beyond the limits of the effluent radiation monitors capabilities to accurately measure.
This finding was considered more than minor because the licensee is required to be capable of implementing adequate

measures to protect public health and safety in the event of a radiological emergency. Regulations require a standard emergency classification and action level scheme, the bases which include facility system and effluent parameters, in use by the licensee and State and local response plans call for reliance on information provided by the licensee for determination of minimum initial offsite response measures. As a result of having General Emergency EAL threshold values that were beyond the range of the associated effluent radiation monitors, Crystal River Unit 3 personnel may not have been able to perform timely and accurate classification of an emergency based upon an effluent radioactive material release. Emergency response actions directed by the State and local emergency response plans, which rely on information provided by the licensee, could have potentially been delayed.
The cause of the finding is related to the human performance cross-cutting element of Decision-making (H.1(a)) for ensuring that risk-significant decisions are made using a systematic process and obtaining interdisciplinary input and reviews.
Inspection Report# : 2011501 *(pdf)*

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

Last modified : January 04, 2012

Crystal River 3
4Q/2011 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Fire Loading Within Allowable Limits
• Green. The inspectors identified two examples of a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C (9), for the failure to adequately evaluate changes to the approved Fire Protection Program. Specifically, in 1999, and 2003, the licensee revised their fire protection program to increase the combustible loading beyond the maximum permissible limits for FA CC-124-116, 480V ES Switchgear Bus Room 3B and FA CC-124-117, 480V ES Switchgear Bus Room 3A, respectively without performing an evaluation to ensure compliance with the approved Fire Protection Program. The licensee initiated nuclear condition reports 461209, and 476342 to address this issue.

The finding was more than minor because it affected the reactor safety mitigating system cornerstone attribute of protection against external events (i.e. fire). For both examples the selection of a "low" degradation rating was supported by screening criteria provided in Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process" as well as IMC 0609, Appendix F, Attachment 2 "Degradation Rating Guide Specific to Various Fire Protection Elements." Based on the above criteria, this finding is screened as having very low safety significance (Green) in Phase 1 of the Significance Determination Process.

The performance deficiency was not assigned a cross cutting aspect as this deficiency occurred over three years ago and is therefore not reflective of current plant performance.

Inspection Report# : 2011008 *(pdf)*

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Procedure OP-880B for Turbine Building Post-Fire Safe Shutdown
• Green. The inspectors identified a non-cited violation of Crystal River Unit 3 (CR3) Technical Specification 5.6.1.1.a., for inadequate guidance in procedure OP-880B, Appendix "R" Turbine Building Post-Fire Safe Shutdown Information. Specifically, the procedure could not have been performed as written because it did not identify the appropriate equipment that was to be manipulated to ensure that the reactor coolant pumps remained de-energized after being secured in the event of a fire in turbine building Fire Zones TB-95-400A, TB-119-400E, or TB-145-400F. Additionally, procedure OP-880B did not provide adequate guidance regarding how CR3 operators would communicate with Crystal River Unit1/Unit 2 (CR1/CR2) operators, and did not specify if a reliable means of communications was available. The licensee initiated nuclear condition reports 460602, and 461736 to address this issue.
The inspectors determined that inadequate safe shutdown procedure guidance was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone and it affected the cornerstone objective of protection against external events (i.e., fire). The inspectors assessed this finding using NRC Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process. The inspectors determined that this finding was of very low safety significance (Green)

because during the time that procedure OP-880B was issued and in effect (April 16, 2010, to April 22, 2011), CR3 was in cold shutdown and procedure OP-880B was not applicable. The inspectors determined that the cause of this finding had a cross-cutting aspect in the Human Performance Area, Work Control Component, in that, the licensee did not address the need for CR3 work groups to maintain interfaces with offsite organizations (i.e., CR1/CR2), to communicate and coordinate with each other during activities in which interdepartmental coordination was necessary to ensure plant and human performance.
Inspection Report# : 2011008 *(pdf)*

Significance: G Mar 31, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
Operating Crew Failures on the 2011 Annual Requalification Operating Test
A self-revealing Green finding, associated with operating crew performance on the simulator during facility-administered requalification examination was identified. Two of the eight crews evaluated failed to pass their simulator examinations. As immediate corrective action, the failed operating crews were remediated (i.e., the operating crews were re-trained and successfully retested) prior to returning to shift. The licensee has entered this issue into the corrective action program as Nuclear Condition Report (NRC) 450196.

The inspectors determined that the crew failures constituted a performance deficiency based on the fact that licensed operators are expected to operate the plant with acceptable standards of knowledge and abilities demonstrated through periodic testing as required by 10 CFR 55.59(a)(2). Two out of eight crews of licensed operators failed to demonstrate a satisfactory understanding of the required actions and mitigating strategies required to safely operate the facility under normal, abnormal, and emergency conditions. The finding is greater than minor because the performance deficiency potentially affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding reflected the crew’s potential inability to take timely actions in response to actual abnormal and emergency conditions. The cause of this finding was directly related to the cross-cutting aspect of personnel training and qualifications in the Resources component of the Human Performance area, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b)) (Section 1R11)

Inspection Report# : 2011002 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: W Jul 15, 2011
Identified By: NRC
Item Type: VIO Violation
Failure to Maintain a Standard EAL Scheme
TBD. An AV was identified for failure to follow and maintain in effect emergency plans which use a standard emergency classification and action level scheme. Specifically, the licensee's emergency plan emergency action level (EAL) 1.4, General Emergency - Gaseous Effluent, specified instrument values that were beyond the limits of the effluent radiation monitors capabilities to accurately measure.
This finding was considered more than minor because the licensee is required to be capable of implementing adequate measures to protect public health and safety in the event of a radiological emergency. Regulations require a standard emergency classification and action level scheme, the bases which include facility system and effluent parameters, in use by the licensee and State and local response plans call for reliance on information provided by the licensee for determination of minimum initial offsite response measures. As a result of having General Emergency EAL threshold values that were beyond the range of the associated effluent radiation monitors, Crystal River Unit 3 personnel may

not have been able to perform timely and accurate classification of an emergency based upon an effluent radioactive material release. Emergency response actions directed by the State and local emergency response plans, which rely on information provided by the licensee, could have potentially been delayed.
The cause of the finding is related to the human performance cross-cutting element of Decision-making (H.1(a)) for ensuring that risk-significant decisions are made using a systematic process and obtaining interdisciplinary input and reviews.
Inspection Report# : 2011504 *(pdf)*
Inspection Report# : 2011501 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

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Miscellaneous

Last modified : March 02, 2012

Crystal River 3
1Q/2012 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Fire Loading Within Allowable Limits
• Green. The inspectors identified two examples of a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C (9), for the failure to adequately evaluate changes to the approved Fire Protection Program. Specifically, in 1999, and 2003, the licensee revised their fire protection program to increase the combustible loading beyond the maximum permissible limits for FA CC-124-116, 480V ES Switchgear Bus Room 3B and FA CC-124-117, 480V ES Switchgear Bus Room 3A, respectively without performing an evaluation to ensure compliance with the approved Fire Protection Program. The licensee initiated nuclear condition reports 461209, and 476342 to address this issue.

The finding was more than minor because it affected the reactor safety mitigating system cornerstone attribute of protection against external events (i.e. fire). For both examples the selection of a "low" degradation rating was supported by screening criteria provided in Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process" as well as IMC 0609, Appendix F, Attachment 2 "Degradation Rating Guide Specific to Various Fire Protection Elements." Based on the above criteria, this finding is screened as having very low safety significance (Green) in Phase 1 of the Significance Determination Process.

The performance deficiency was not assigned a cross cutting aspect as this deficiency occurred over three years ago and is therefore not reflective of current plant performance.

Inspection Report# : 2011008 *(pdf)*

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Procedure OP-880B for Turbine Building Post-Fire Safe Shutdown
• Green. The inspectors identified a non-cited violation of Crystal River Unit 3 (CR3) Technical Specification 5.6.1.1.a., for inadequate guidance in procedure OP-880B, Appendix "R" Turbine Building Post-Fire Safe Shutdown Information. Specifically, the procedure could not have been performed as written because it did not identify the appropriate equipment that was to be manipulated to ensure that the reactor coolant pumps remained de-energized after being secured in the event of a fire in turbine building Fire Zones TB-95-400A, TB-119-400E, or TB-145-400F. Additionally, procedure OP-880B did not provide adequate guidance regarding how CR3 operators would communicate with Crystal River Unit1/Unit 2 (CR1/CR2) operators, and did not specify if a reliable means of communications was available. The licensee initiated nuclear condition reports 460602, and 461736 to address this issue.
The inspectors determined that inadequate safe shutdown procedure guidance was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone and it affected the cornerstone objective of protection against external events (i.e., fire). The inspectors assessed this finding using NRC Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process. The inspectors determined that this finding was of very low safety significance (Green)

because during the time that procedure OP-880B was issued and in effect (April 16, 2010, to April 22, 2011), CR3 was in cold shutdown and procedure OP-880B was not applicable. The inspectors determined that the cause of this finding had a cross-cutting aspect in the Human Performance Area, Work Control Component, in that, the licensee did not address the need for CR3 work groups to maintain interfaces with offsite organizations (i.e., CR1/CR2), to communicate and coordinate with each other during activities in which interdepartmental coordination was necessary to ensure plant and human performance.
Inspection Report# : 2011008 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: W Jul 15, 2011
Identified By: NRC
Item Type: VIO Violation
Failure to Maintain a Standard EAL Scheme
TBD. An AV was identified for failure to follow and maintain in effect emergency plans which use a standard emergency classification and action level scheme. Specifically, the licensee's emergency plan emergency action level (EAL) 1.4, General Emergency - Gaseous Effluent, specified instrument values that were beyond the limits of the effluent radiation monitors capabilities to accurately measure.
This finding was considered more than minor because the licensee is required to be capable of implementing adequate measures to protect public health and safety in the event of a radiological emergency. Regulations require a standard emergency classification and action level scheme, the bases which include facility system and effluent parameters, in use by the licensee and State and local response plans call for reliance on information provided by the licensee for determination of minimum initial offsite response measures. As a result of having General Emergency EAL threshold values that were beyond the range of the associated effluent radiation monitors, Crystal River Unit 3 personnel may not have been able to perform timely and accurate classification of an emergency based upon an effluent radioactive material release. Emergency response actions directed by the State and local emergency response plans, which rely on information provided by the licensee, could have potentially been delayed.
The cause of the finding is related to the human performance cross-cutting element of Decision-making (H.1(a)) for ensuring that risk-significant decisions are made using a systematic process and obtaining interdisciplinary input and reviews.
Inspection Report# : 2011501 *(pdf)*
Inspection Report# : 2011504 *(pdf)*

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

Last modified : May 29, 2012

Crystal River 3
2Q/2012 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Fire Loading Within Allowable Limits
• Green. The inspectors identified two examples of a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C (9), for the failure to adequately evaluate changes to the approved Fire Protection Program. Specifically, in 1999, and 2003, the licensee revised their fire protection program to increase the combustible loading beyond the maximum permissible limits for FA CC-124-116, 480V ES Switchgear Bus Room 3B and FA CC-124-117, 480V ES Switchgear Bus Room 3A, respectively without performing an evaluation to ensure compliance with the approved Fire Protection Program. The licensee initiated nuclear condition reports 461209, and 476342 to address this issue.

The finding was more than minor because it affected the reactor safety mitigating system cornerstone attribute of protection against external events (i.e. fire). For both examples the selection of a "low" degradation rating was supported by screening criteria provided in Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process" as well as IMC 0609, Appendix F, Attachment 2 "Degradation Rating Guide Specific to Various Fire Protection Elements." Based on the above criteria, this finding is screened as having very low safety significance (Green) in Phase 1 of the Significance Determination Process.

The performance deficiency was not assigned a cross cutting aspect as this deficiency occurred over three years ago and is therefore not reflective of current plant performance.

Inspection Report# : 2011008 *(pdf)*

Significance: G Jul 14, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Procedure OP-880B for Turbine Building Post-Fire Safe Shutdown
• Green. The inspectors identified a non-cited violation of Crystal River Unit 3 (CR3) Technical Specification 5.6.1.1.a., for inadequate guidance in procedure OP-880B, Appendix "R" Turbine Building Post-Fire Safe Shutdown Information. Specifically, the procedure could not have been performed as written because it did not identify the appropriate equipment that was to be manipulated to ensure that the reactor coolant pumps remained de-energized after being secured in the event of a fire in turbine building Fire Zones TB-95-400A, TB-119-400E, or TB-145-400F. Additionally, procedure OP-880B did not provide adequate guidance regarding how CR3 operators would communicate with Crystal River Unit1/Unit 2 (CR1/CR2) operators, and did not specify if a reliable means of communications was available. The licensee initiated nuclear condition reports 460602, and 461736 to address this issue.
The inspectors determined that inadequate safe shutdown procedure guidance was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone and it affected the cornerstone objective of protection against external events (i.e., fire). The inspectors assessed this finding using NRC Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance

Determination Process. The inspectors determined that this finding was of very low safety significance (Green) because during the time that procedure OP-880B was issued and in effect (April 16, 2010, to April 22, 2011), CR3 was in cold shutdown and procedure OP-880B was not applicable. The inspectors determined that the cause of this finding had a cross-cutting aspect in the Human Performance Area, Work Control Component, in that, the licensee did not address the need for CR3 work groups to maintain interfaces with offsite organizations (i.e., CR1/CR2), to communicate and coordinate with each other during activities in which interdepartmental coordination was necessary to ensure plant and human performance.
Inspection Report# : 2011008 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: W Jul 15, 2011
Identified By: NRC
Item Type: VIO Violation
Failure to Maintain a Standard EAL Scheme
TBD. An AV was identified for failure to follow and maintain in effect emergency plans which use a standard emergency classification and action level scheme. Specifically, the licensee's emergency plan emergency action level (EAL) 1.4, General Emergency - Gaseous Effluent, specified instrument values that were beyond the limits of the effluent radiation monitors capabilities to accurately measure.
This finding was considered more than minor because the licensee is required to be capable of implementing adequate measures to protect public health and safety in the event of a radiological emergency. Regulations require a standard emergency classification and action level scheme, the bases which include facility system and effluent parameters, in use by the licensee and State and local response plans call for reliance on information provided by the licensee for determination of minimum initial offsite response measures. As a result of having General Emergency EAL threshold values that were beyond the range of the associated effluent radiation monitors, Crystal River Unit 3 personnel may not have been able to perform timely and accurate classification of an emergency based upon an effluent radioactive material release. Emergency response actions directed by the State and local emergency response plans, which rely on information provided by the licensee, could have potentially been delayed.
The cause of the finding is related to the human performance cross-cutting element of Decision-making (H.1(a)) for ensuring that risk-significant decisions are made using a systematic process and obtaining interdisciplinary input and reviews.

The NRC staff conducted this supplemental inspection in accordance with Inspection Procedure 95001, “Inspection for one or two White inputs in a Strategic Performance Area,” to assess the licensee’s evaluation associated with the failure to follow and maintain a standardized Emergency Action Level (EAL) scheme. The NRC staff previously characterized this issue as having low to moderate risk significance (White) in NRC Inspection Report (IR) 05000302/2011504. During this supplemental inspection, the inspectors determined that the licensee had performed a comprehensive evaluation of the licensee-identified failure to follow and maintain a standardized EAL scheme for several years prior to June 2011. The licensee identified the root cause of the issue to be insufficient procedural guidance to direct the EAL change process. All immediate and long term corrective actions have been completed except for: (1) Evaluate/change EAL threshold values based on updated assumptions coincident with replacing the RM-A1 and RM-A2 instruments (expected completion date (ECD) of October 15, 2012); (2) Complete corrective action effectiveness reviews (ECD September 1, 2012); and (3) Assess Emergency Preparedness (EP) staff including workload leveling and degree of interaction with other site groups (ECD December 29, 2012). Given the licensee’s acceptable performance in addressing the failure to follow and maintain a standardized EAL scheme, the White finding associated with this issue will only be considered in assessing plant performance until the end of the second quarter 2012, in accordance with the guidance in Inspection Manual Chapter (IMC) 0305, “Operating Reactor Assessment Program.” Implementation of the licensee’s corrective actions will be reviewed during a future inspection.

Inspection Report# : 2011504 *(pdf)*
Inspection Report# : 2011501 *(pdf)*
Inspection Report# : 2012503 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : September 12, 2012

Crystal River 3
3Q/2012 Plant Inspection Findings

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Security

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Last modified : November 30, 2012

Crystal River 3
4Q/2012 Plant Inspection Findings

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Last modified : February 28, 2013

Crystal River 3
1Q/2013 Plant Inspection Findings

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Miscellaneous

Last modified : June 04, 2013

Crystal River 3
2Q/2013 Plant Inspection Findings

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Security

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Last modified : September 03, 2013

Crystal River 3
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Last modified : September 03, 2013

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Last modified : September 03, 2013

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Last modified : September 03, 2013

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Last modified : September 03, 2013

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Last modified : September 03, 2013

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Miscellaneous

Last modified : September 03, 2013

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Crystal River 3 – Quarterly Plant Inspection Findings

2Q/2013 Plant Inspection Findings

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Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Current data as of : September 03, 2013

Page Last Reviewed/Updated Wednesday, June 07, 2017

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Crystal River 3 – Quarterly Plant Inspection Findings

2Q/2013 Plant Inspection Findings

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Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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Current data as of : September 03, 2013

Page Last Reviewed/Updated Monday, November 06, 2017