UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 13, 2007

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000302/2007002

Dear Mr. Young:

On March 31, 2007, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Crystal River Unit 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 9, 2007, with you and members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, three self-revealing findings of very low safety significance (Green) were identified. One of the findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance of the issue and because it was entered into your corrective action program, the NRC is treating the issue as a Non-Cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, NRC Region II; The Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Crystal River Unit 3 site.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system

(ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael E. Ernstes, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No.: 50-302
License No.: DPR-72
Enclosure: Inspection Report 05000302/2007002
w/Attachment: Supplemental Information
cc w/encl: (See page 3)

(ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael E. Ernstes, Chief
Reactor Projects Branch 3
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Docket No.: 50-302
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Report to Dale E. Young from Michael E. Ernstes dated April 13, 2007

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000302/2007002

Distribution w/encl:
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No:	05000302/2007002
Licensee:	Progress Energy Florida (Florida Power Corporation)
Facility	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	January 1, 2007 - March 31, 2007
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector
Approved by:	Michael E. Ernstes, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2007-002; 01/01/2007 - 03/31/2007; Crystal River Unit 3; Problem Identification and Resolution; Followup of Events and Notices of Enforcement Discretion.

The report covered a three-month period of inspection by the resident inspectors. Three Green self-revealing findings, one of which was a non-cited violation (NCV), were identified during this inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events and Mitigating Systems

- Green: 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)

- Green: 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)

- Green: 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)

B. Licensee-identified Violations

None

REPORT DETAILS

Summary of Plant Status:

The unit began the inspection period at 100 percent power. On February 17, the unit was reduced to approximately 74 percent power to support a planned circulating water system pump outage. On February 19, the unit automatically ran back to approximately 52 percent power when a main feedwater pump tripped due to water intrusion into an electrical connection. The condition was corrected and power was increased to 74 percent on February 20. On February 21, the unit experienced an automatic trip due to a failure of an integrated control system electronic module. The module was replaced and the unit was restarted on February 23 and returned to approximately 73% power to complete planned circulating water system maintenance. The unit resumed full power operation on February 26. The unit was at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On January 29 and 30, when outdoor temperatures fell below 40 degrees Fahrenheit (F), the inspectors verified that the licensee implemented Administrative Instruction AI-513, Seasonal Weather Preparations, Sections 4.2 (Freezing Weather) and 4.3 (Freezing Weather Monitoring). The inspectors walked down portions of the emergency feedwater pump EFP-3 system to check for any unidentified susceptibilities. Nuclear condition reports were reviewed to check that the licensee was identifying and correcting cold weather protection issues. This completed one sample for a site specific weather related condition.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the critical portions of the selected trains to verify correct system alignment. The inspectors reviewed plant documents to determine the correct system and power alignments, and the required positions of select valves and breakers. The inspectors verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact mitigating system availability. The inspectors verified the following four partial system alignments in system walkdowns using the listed documents:

- January 17, Emergency Feedwater Pump EFP-3, using OP-450, Emergency Feedwater System, while the turbine driven emergency feedwater pump EFP-2 was out of service for maintenance

- January 22, Feedwater Pump FWP-7, using OP- 605, Feedwater System, while the control complex chiller 1A was out of service for maintenance

- January 31 and February 6, Boration flow path via the boric acid storage tanks and the borated water storage tank, using OP- 403B, Chemical Addition Boric Acid System, and OP- 402, Make-up and Purification System

- January 26, Raw water pump RWP-2B, using OP-408, Nuclear Services Cooling System , while RWP-2A was out of service due to emergent work on raw water valve RWV-24

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of both trains of the nuclear services closed cycle cooling system (SW). The inspectors used licensee operating procedure, OP- 408, Nuclear Services Cooling System, as well as Final Safety Analysis Report (FSAR) Chapter 9.5, Cooling Water Systems, to verify proper system alignment. This completes one sample of a complete system alignment.

The walkdowns also included evaluation of selected system piping and supports against the following considerations:

- Piping and pipe supports did not show evidence of water hammer
- Oil reservoir levels indicated normal
- Snubbers did not indicate any observable hydraulic fluid leakage
- Component foundations were not degraded
- No fire protection hazards

A review of outstanding maintenance work orders was performed to verify that any deficiencies did not significantly affect the system function. In addition, the inspectors reviewed nuclear condition reports (NCRs) to verify that system problems were being identified and appropriately resolved. Inspectors reviewed the system walkdown report, Administrative Instruction AI-1701, and the Quarterly Walkdown of SW and RW systems, dated January 08, 2007. The inspectors routinely checked operability of the heat removal system during heat exchanger maintenance using the licensee’s operating procedure OP-103B, Operating Curves, Curve 15, SW System Heat Transfer Capability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection Walkdowns

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's implementation of the fire protection program. The inspectors checked that the areas were free of transient combustible material and other ignition sources. Also, fire detection and suppression capabilities, fire barriers, and compensatory measures for fire protection problems were verified. The inspectors checked fire suppression and detection equipment to determine whether conditions or deficiencies existed which could impair the function of the equipment. The inspectors selected the areas based on a review of the licensee's probabilistic risk assessment. The inspectors also reviewed the licensee's fire protection program to verify the requirements of Final Safety Analysis Report (FSAR) Section 9.8, Plant Fire Protection Program, were met. Documents reviewed are listed in the Attachment. The inspectors toured the following nine areas important to reactor safety:

- Emergency feed pump (EFP)-3 building
- Control rod drive room
- Intermediate building 119' elevation
- Unit 4160/6900V switchgear room turbine building 119' elevation
- Auxiliary building 143' elevation main exhaust filter room
- Intermediate building 95' elevation, EFP-1 and EFP-2 area
- Unit 3 Control Room
- Feed water pump FWP-7 area
- Control complex 109' elevation alternate shut down room

b. Findings

No findings of significance were identified.

.2 Annual Inspection

a. Inspection Scope

On February 8 and again on March 8, the inspectors observed the licensee fire brigade's response to unannounced simulated fires in the RCA intermediate building 95-foot level and control complex chiller room, respectively. The inspectors checked the brigade's communications, ability to set-up and execute fire operations, and their use of fire fighting equipment. The inspectors verified compensatory actions were in place to ensure that additional alarms which may be received during the drill were addressed.

Additionally, the inspectors verified that the licensee considered the aspects as described below when the brigade conducted the firefighting activities and during the post-drill critique. The inspectors attended the post-drill critique to check that the licensee's drill acceptance criteria were met and that any discrepancies were discussed and resolved. Administrative Instruction AI-2205, Administration of CR-3 Fire Brigade Organization and Duties of the Fire Brigade, and the fire drill evaluation report were reviewed to assure that acceptance criteria were evaluated and deficiencies were documented and corrected. In addition, the inspectors reviewed the storage, training, expectations for use and maintenance associated with the self-contained breathing apparatus (SCBA) program. This completed one sample representing observation of selected fire drills. Items reviewed are listed in the Attachment.

- The brigade, including the fire team leader, had a minimum of five members
- Members set out designated protective clothing and properly donned gear
- SCBA were available and properly used
- Control room personal verified fire location, dispatched fire brigade and sounded alarms. Emergency action levels were declared and notifications made
- Fire brigade leader as well as the control room senior reactor operator had copies of the pre-fire plans
- Brigade leader maintained control. Members were briefed, discussed plan of attack, received individual assignments, and completed communications checks Plan of attack discussions were consistent with pre-fire plans
- Fire brigade arrived at the fire scene in a timely manner, taking the appropriate access route specified in the strategies and procedures
- Control and command was set up near the fire scene and communications were established with the control room and the fire brigade members
- Effectiveness of radio communication between the command post, control room, plant operators and fire brigade members
- Fire hose lines reached all necessary fire hazard locations, were laid out without flow constrictions, and were simulated as being charged with water
- The fire area was entered in a controlled manner following the two person rule
- The fire brigade brought sufficient fire-fighting equipment to the scene to properly perform its fire-fighting duties
- The fire brigade checked for fire victims and fire propagation into other areas
- Effective smoke removal operations were simulated in accordance with the pre-fire plan
- The fire-fighting pre-fire plan strategies were utilized
- The drill scenario was followed, and the drill acceptance criteria were met
- All fire fighting equipment was returned to a condition of readiness

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

Resident Inspector Quarterly Review

a. Inspection Scope

On February 1, the inspectors observed licensed operators' response and actions for the Crystal River Unit 3 licensed operator annual exam simulator evaluated session SES-22, Plant Runback/Total Loss of Feedwater/High Pressure Injection/Power Operated Relief Valve Cooling to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. Emergency Operating Procedures (EOP's) used included EOP-02, Vital System Status Verification and EOP-04, Inadequate Heat Transfer. The inspectors specifically evaluated the following attributes related to operating crew performance:

- Clarity and formality of communication including crew briefings
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Implementation of emergency operating procedures (EOPs)
- Control board operation and manipulation, including operator actions
- Assessment of emergency classifications
- Oversight and direction provided by supervision, including ability to identify and notification of state authorities within the 15 minute requirement

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components within the scope of the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored, and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors conducted this inspection for three degraded equipment conditions listed below. The inspectors verified that the licensee was appropriately identifying and documenting maintenance rule issues in the corrective action program. Documents reviewed are listed in the Attachment. The inspectors attended the maintenance rule expert panel which discussed returning the service water system back to a(2). The licensee's maintenance effectiveness was evaluated for the following three degraded equipment conditions:

- NCR 214165, RWP-2B bearing failure resulted in B train raw water system being classified as maintenance rule (a)(1)

- NCR 222231, Emergency feedwater pump EFP-2 low margin to maintenance rule (a)(1)

- NCR 186661, Service Water System exceeded maintenance rule functional failure limit. SE07-0018, Service Water Radioactive Barrier Return to A(2) status

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the risk impact associated with those activities listed below and verified the licensee's associated risk management actions. This review primarily focused on equipment determined to be risk significant within the maintenance rule. The inspectors also assessed the adequacy of the licensee's identification and resolution of problems associated with risk management including emergent work activities. The licensee's implementation of compliance procedure CP-253, Power Operation Risk Assessment, was verified in each of the following six work week assessments.

- Work Week 07W02, Risk Assessment for operation with emergency feedwater pump EFP-2 out of service for maintenance

- Work Week 07W03, Risk Assessment for operation with individually out of service, control complex chiller CHHE-1A and emergency diesel generator EGDG-1A for planned maintenance and RWV-24 for emergent work

- Work Week 07W06, Risk Assessment for operation with an unavailable Auxiliary Feed Water (FWP-7) pump

- Work Week 07W08, Risk Assessment for operation with emergency feedwater pump EFP-3 out of service for maintenance during integrated control system (ICS) troubleshooting and repair activities

- Work Week 07W09, Risk Assessment for operation with B train raw water pumps out of service for planned maintenance (Yellow risk condition)

- Work Week 07W12, Risk Assessment for operation with A train raw water pumps out of service for planned maintenance (Yellow risk condition)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following six NCRs to verify that the operability of systems important to safety was properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. The inspectors determined if operability of systems or components important to safety was consistent with technical specifications, the FSAR, 10 CFR Part 50 requirements, and when applicable, NRC Inspection Manual, Part 9900, Technical Guidance, “Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety.” The inspectors reviewed licensee NCRs, work schedules, and engineering documents to check if operability issues were being identified at an appropriate threshold and documented in the corrective action program, consistent with 10 CFR 50, Appendix B requirements; and licensee procedure NGGC-CAP-200, corrective action program.

- NCR 218244, Make-up pump 1A outboard oil seal increased leakage
- NCR 219138, Degraded service water system piping supports
- NCR 221195, Raw water system train A degraded standby flush water flow
- NCR 223357, Emergency feedwater pump EFP-3 discharge flange leak
- NCR 221362, Minor oil leak at EGDG-1A control side turbo/exhaust
- NCR 227372, Velan Inc, discovers flow error on piston check valves

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and/or reviewed post-maintenance test procedures and/or test activities, as appropriate, for selected risk significant systems to verify whether: (1) testing was adequate for the maintenance performed; (2) acceptance criteria were clear, and adequately demonstrated operational readiness consistent with design and licensing basis documents; (3) test instrumentation had current calibrations, range, and accuracy consistent with the application; (4) tests were performed as written with applicable prerequisites satisfied, and (5) equipment was returned to the status required to perform its safety function. The seven post-maintenance tests reviewed are listed below:

- Surveillance Procedure SP-375A, CHP-1A and Valve Surveillance, after performing maintenance on the 1A control complex chiller per WO 977280

- Surveillance Procedure SP-354C, Functional Test Of The Alternate AC Diesel Generator EGDG-1C, after performing maintenance per WO 867838

- Surveillance Procedure SP-348A, Auxiliary Feed Water Pump (FWP-7) Testing, after performing maintenance per WO 955311

- Surveillance Procedure SP -349C, EFP-3 and Valve Surveillance, after performing maintenance on EFV-147 flange per WO 1020175

- Administrative Instruction AI-615, Troubleshooting Plant Equipment, for troubleshooting ICS downward megawatt thermal trend under WO 1020172

- Surveillance Procedures SP-340D, RWP-3B, DCP-1B and Valve Surveillance; SP-344A, RWP-2A, SWP-1A and Valve Surveillance; and SP-344B, RWP-2B, SWP-1A and Valve Surveillance; after performing maintenance on B train raw water system under WO's 837157, 1013116, 800804 and 814865

- Surveillance Procedures SP-340A, RWP-3A, DCP-1A and Valve Surveillance; SP-344B, RWP-2B,SWP-1B and Valve Surveillance; and SP-344A, RWP-2A, SWP-1A and Valve Surveillance; after performing maintenance on A train raw water system under WO's 8775036 and 1003329

b. Findings

No findings of significance were identified.

1R20 Refueling and other outage activities

Unplanned outage as a result of an integrated control system (ICS) failure

a. Inspection Scope

On February 21, as a result of an ICS circuit module failure, the unit tripped and was stabilized in Mode 3. The inspectors reviewed the forced outage schedule to confirm that the licensee had appropriately considered risk in developing and implementing the plans. The inspectors verified that the reactor building was inspected to ensure no active leakage was present. During the short outage, the inspectors verified the status and configuration of mechanical and electrical systems met TS requirements. The unit was restarted on February 23. Licensee documents reviewed for the inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests listed below to verify that technical specification surveillance requirements were followed and that test acceptance criteria were properly specified. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met. Additionally, the inspectors also verified that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance or as part of surveillance testing. The following seven activities were observed/reviewed:

In-Service Test:

- SP-340A, RWP-3A, DCP-1A and Valve Surveillance

Surveillance Tests:

- SP-113A, Channel A Power Range Nuclear Instrumentation Calibration
- SP-130, Engineering Safeguards monthly Functional Test
- SP-110A, "A" Channel Reactor Protection System Functional Test
- SP-354B, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1B
- SP-907A, Monthly Functional Test of 4160V ES Bus "A" Undervoltage and Degraded Grid Relaying

Reactor Coolant System Leak Detection Test:

- SP-317 RC System Water Inventory Balance

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors evaluated one temporary modification and the associated 10 CFR 50.59 screening against the system design basis documentation and FSAR to verify the modification did not adversely affect the safety functions of important safety systems. Additionally, the inspectors reviewed licensee procedure EGR-NGGC-0005, Engineering Change, to assess if the modification was properly developed and implemented.

- WO 1016568, Install temporary power supplies PS1 and PS2 in non-nuclear instrumentation cabinet NNI-Y per engineering change EC-66174

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed and reviewed one emergency response activity to verify the licensee was properly classifying emergency events, making the required notifications, and appropriate protective action recommendations. The inspectors assessed the licensee's ability to classify emergent situations and make timely notification to State and Federal officials in accordance with 10 CFR Part 50.72. Emergency activities were verified to be in accordance with the Crystal River Radiological Emergency Response Plan, Section 8.0, Emergency Classification System, and 10 CFR Part 50, Appendix E. Additionally, the inspectors verified that adequate licensee critiques were conducted in order to identify performance weaknesses and necessary improvements.

- On February 1, licensed operator Simulator Evaluated Session, SES-22, Plant Runback/Total Loss of Feedwater/High Pressure Injection/Power Operated Relief Valve Cooling.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors checked the licensee submittal for the PIs listed below for the period January 1, 2006 through December 31, 2006 to verify accuracy. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 4, were used to check the reporting for each data element. The inspector checked licensee event reports (LERs), operator logs, and daily plant status reports to verify the licensee accurately reported the data including the number of critical hours reported. In addition, the inspectors interviewed licensee personnel associated with PI data collection, evaluation, and distribution. The inspectors checked that any deficiencies affecting the licensee's performance indicator program were entered into the corrective action program (CAP) and appropriately resolved.

- Unplanned Scrams per 7000 Critical Hours

- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Screening of Items Entered Into the Corrective Action Program (CAP)

a. Inspection Scope

As required by inspection procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by attending daily plant status meetings, interviewing plant operators and applicable system engineers, and accessing the licensee's computerized database.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected NCRs 216465 and 214698 for a detailed review. The NCRs were initiated to (1) address the need to place additional focus on the raw water system due to a growing number of concerns with the system, and (2) address the failure of the circulating water screen wash system. The inspectors checked that the issue had been completely and accurately identified in the licensee's CAP, and that safety concerns were properly classified and prioritized for resolution, apparent cause determinations were sufficiently thorough, and appropriate corrective actions were implemented in a manner consistent with safety and compliance with plant technical specifications and 10 CFR 50. The inspectors also evaluated the NCR using the requirements of the licensee's CAP as delineated in corrective action procedure CAP-NGGC-200, Corrective Action Program.

b. Findings and Observations

NCR 216465 documented a need for additional focus on the raw water system due to the number of deficiencies identified with the system over the last several years. Through review of NCR 216465, the documents listed in the Attachment, system walkdowns and discussions with both the raw water system engineer and his supervisor, the inspectors determined that corrective actions completed and scheduled are timely and should be appropriate to reduce the number of equipment problems associated with the raw water system.

The inspectors found that the licensees review of the failure of the circulating water system screen wash system (NCR 214698) and corrective actions were comprehensive and thorough. A finding associated with the cause of the screen wash system failure is documented below.

Introduction: A Green self-revealing finding was identified for 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.

Description: On November 29, control room operators entered Abnormal Procedure AP-510, Rapid Power Reduction, and decreased reactor power from 100 percent to 80 percent. The power decrease was required due to condenser differential temperature exceeding procedural operating limits. Operations had identified that the traveling screens had not received an automatic start signal. Upon attempting to start the traveling screens, with the exception of one screen, all traveling screens stopped operating within minutes. The licensee found significant debris accumulation on the screens which caused an increase in differential pressure across the screens, i.e., the screens were loaded to the point that the motors were not capable of rotating the screens. After repairs were completed, reactor power was increased and restored to 100 percent. The screens were kept in operation while troubleshooting was performed over the next several days.

The licensee found that the failure of the screen auto start system was caused by failure of the differential pressure level sensing system. Specifically, the level sensors failed due to marine growth in the low side sensing tubes. The marine debris blocked the low side differential level sensing tubes, thus giving an incorrect indication of differential pressure across the screens. Although preventive maintenance was performed to check differential level set points, no maintenance was ever performed to address marine fouling in the sensing tubes. As such, the licensee's root cause evaluation concluded that there had been a failure to address the marine fouling failure mode in the scope of the existing preventive maintenance. Additionally, the licensee identified industry Operating Experience (OE) that had been processed at the station through the corrective action program, that had not been thoroughly implemented which may have prevented this issue.

Analysis: The performance deficiency associated with this finding was for failure to address the marine fouling failure mode in the scope of the existing preventive maintenance on the intake screen wash auto start system. 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 (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. A contributing cause of the finding is related to the cross-cutting area of Problem Identification and Resolution, specifically the Operating Experience 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.

Enforcement: 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 was not an activity affecting quality subject to 10 CFR Part 50, Appendix B, nor a procedure required by licensee conditions or Improved Technical Specifications. Therefore, while a performance deficiency existed, no violation of regulatory requirements occurred. This finding was determined to be of very low safety significance (Green) and was entered into the corrective action program as Nuclear Condition Report 214698. This finding is identified as FIN 05000302/2007002-001, Failure to Address Marine Fouling Resulted in a Plant Transient. 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.

4OA3 Followup of Events and Notices of Enforcement Discretion

.1 Operator performance during non-routine events

a. Inspection Scope

For the three non-routine plant evolutions described below, the inspectors reviewed the operating crew's performance, operator logs, control board indications, and the plant computer data to verify that operator response was in accordance with the associated plant procedures. Additional documents reviewed are listed in the Attachment.

- February 18, Planned reactor power decrease to 74 percent power in accordance with Operating Procedure OP-204, Power Operations
- February 19, Automatic reactor runback from 74 to 52 percent power due to the loss of a main feedwater pump
- February 21, Automatic reactor trip due to an integrated control system failure

b. Findings

Introduction: A Green 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 the multiplier module resulted in an automatic reactor trip.

Description: On February 21, 2007, at approximately 71 percent power, ICS operation became erratic causing a main feedwater transient and a low feedwater flow condition. The reactor protection system actuated on high reactor coolant system pressure causing an automatic reactor trip. The emergency feedwater initiation and control (EFIC) system actuated on low steam generator levels and started both emergency feedwater pumps. The unit was stabilized in a Hot Shutdown condition. The licensee determined that ICS module IC-384-IC had failed causing the feedwater transient and resultant reactor trip.

ICS module IC-384-IC had previously failed on March 24, 2004 causing a reactor trip. The licensee's root cause investigation determined that the failure was caused by age-related failure of the module's sub-components. A refurbishment program was established for these type of modules. Four ICS multiplier modules with controlling functions, including module IC-384-IC, were replaced with ones that had been refurbished. During refueling outage R14 (Fall 2005), refurbished module IC-384-IC failed its calibration and was replaced by a module that had not undergone refurbishment. Since a refurbished multiplier module was not available, engineering approved the temporary installation of a non-refurbished module. Engineering did not ensure a tracking mechanism was put in place to obtain and install a refurbished module. The licensee's investigation determined that the root cause of the event was engineering's failure to turnover or communicate to supervision the need to expeditiously replace the module with one that was refurbished.

Analysis: The inspectors determined that the failure to replace a known non-refurbished ICS multiplier module is a performance deficiency. The inspectors determined 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.

Enforcement: The inspectors determined that since the finding was associated with non-safety related equipment, no violation of regulatory requirements occurred. The licensee performed an extent of condition review and a formal root cause investigation to determine corrective actions. This finding was determined to be of very low safety significance (Green) and was entered into the corrective action program as Nuclear Condition Report 223337. The finding is identified as FIN 05000302/2007002-002, Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip. 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.

.2 (Closed) URI 05000302/2006005-002: Raw Water System in a Condition Prohibited by Improved Technical Specifications (ITS)

(Closed) LER 05000302/2006-001-00: Train B Raw Water System in a Condition Prohibited by Technical Specifications Due to Equipment Failure

a. Inspection Scope

The inspectors reviewed the root cause evaluation associated with LER 05000302/2006-001-00 to determine whether a performance deficiency was involved, corrective actions were adequate, and the safety significance. The resident inspectors and regional specialists reviewed a licensee evaluation documented in NCR 210023 that concluded that the degraded B train raw water flush water system would have provided adequate flow to raw water pumps RWP-2B and RWP-3B bearings for a minimum of 24 hours and most likely 30 days to determine whether both raw water pumps would have performed their safety function during a loss of offsite power (LOOP) event.

b. Findings

Introduction. A Green self-revealing, Non-cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI was identified as a result of the licensee's 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 ITS when shell debris passed through a corroded strainer and clogged the cyclone separator discharge piping.

Description: On October 20, 2006, during performance of the weekly surveillance (SP-306, Routine Surveillance Log, revision 61), a plant operator questioned whether adequate flow existed through the B train raw water bearing flush water strainer since the strainer differential pressure (DP) was 0 inches water column (WC) and the associated cyclone separator discharge piping was warmer than that of the A train discharge piping. The A train cyclone separator discharge piping, unlike the B train piping, was also moist with condensation due to the cold water flow internal to the piping. System engineering, using a portable flow meter, measured the B train cyclone separator discharge piping flow to be approximately 0.4 gallons per minute (gpm). Minimum design flow is 9 gpm. Shell debris found in the discharge port of the cyclone separator was removed and the system was returned to service.

SP-306 specified, in part, that if differential pressure was less than 2 inches WC, then ensure flow exists by verifying strainer outlet temperature is approximately the same as the ultimate heat sink (UHS) temperature. The licensee determined that the DP had been less than 2 inches WC since July 6, 2006 and the temperature guidance was ineffective in determining flow. The weekly surveillance failed to identify this adverse condition until the UHS was cold enough to show a difference in pipe temperature between the two trains. The inspectors characterized this finding as a self-revealing since SP-306 temperature guidance was inadequate to provide assurance of adequate flow and the adverse condition only became self-revealing when the UHS temperature dropped due to seasonal conditions.

The licensee's investigation determined that debris large enough to clog the cyclone separator discharge piping had passed through a corroded weld seam on a strainer basket. A basket with a corroded weld seam had been replaced on September 1, 2006. The root cause was determined to be the use of Monel alloy, susceptible to pitting and crevice corrosion in low flow conditions, in strainer fabrication.

Normally, bearing flush water is supplied by a non-safety system that does not use the baskets or cyclone separator. During a LOOP event, bearing flush water is supplied from the discharge of the raw water pumps through a strainer basket and a cyclone separator. The strainer baskets were installed during the 1997 extended design outage to prevent clogging of the cyclone separator. In 1998, the licensee recognized that Monel was a poor choice of material when corrosion of the strainer baskets was first found. An Engineering recommendation to change the strainer basket material was not implemented. Corrective actions included periodic inspections and basket replacement as needed. Until 2001, the raw water system design basis document listed the filtering ability of the cyclone separator as an operational consideration and not a required design parameter. Since making the filtering ability of the cyclone separator a required design parameter, more than 20 strainer baskets have been replaced due to corrosion issues under the preventative maintenance (PM) program. Replacing degraded baskets became routine under the PM program and were not documented in the corrective active program.

<u>Analysis</u>: The inspectors determined that the failure to identify and take appropriate corrective actions for repetitive failures of the raw water pumps' bearing flush water strainer baskets in the corrective action program was a performance deficiency. The inspectors determined the finding was more than minor because it affected the equipment reliability attribute of the Mitigating System Cornerstone and resulted in a raw water system train being inoperable for a period of time greater than allowed by ITS. 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 LOOP event. The 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.

<u>Enforcement</u>: 10 CFR, Appendix B, Criterion XVI, Corrective Action, requires, in part, the establishment of measures to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, correction of degraded raw water strainer baskets, that are required to assure proper filtered bearing flush water, was not performed until October 20, 2006 when the system was found degraded due to shell blockage. This finding was determined to be of very low safety significance and was entered into the corrective action program as Nuclear Condition Report 210023. The finding is identified as NCV 05000302/2007002-003, Failure to Identify and Correct

Repetitive Raw Water System Flush Water Strainer Baskets Degradation. New strainer baskets made of a material compatible with service conditions were installed. Additional corrective actions include: performing routine engineering reviews 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.

4OA6 Meetings

Exit Meeting Summary

On April 9, 2007, the resident inspectors presented the inspection results to Mr. D. Young, Site Vice President and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
M. Annacone, Manager, Engineering
W. Brewer, Manager, Maintenance
J. Franke, Plant General Manager
J. Hays, Manager, Outage and Scheduling
T. Hobbs, Manager, Nuclear Assessment
J. Holt, Manager, Operations
R. Hons, Manager, Training
P. Infanger, Supervisor, Licensing
M. Rigsby, Superintendent, Radiation Protection
D. Roderick, Director Site Operations
J. Stephenson, Supervisor, Emergency Preparedness
D. Young, Vice President, Crystal River Nuclear Plant

NRC personnel:
M. Ernstes, Chief, Reactor Projects Branch 3, NRC Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000302/2007002-001	FIN	Failure to Address Marine Fouling Resulted in a Plant Transient (Section 4OA2.2)
05000302/2007002-002	FIN	Failure to Replace a Non-refurbished ICS Module Resulted in a Reactor Trip (Section 4OA3.1)
05000302/2007002-003	NCV	Failure to Identify and Correct Repetitive Raw Water System Flush Water Strainer Baskets Degradation (Section 4OA3.2)

Closed

05000302/2006005-002	URI	Raw Water System in a Condition Prohibited by ITS (Section 4OA3.2)
05000302/2006-001-00	LER	Train B Raw Water System in a Condition Prohibited by Technical Specifications Due to Equipment Failure (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection

Procedures

AI-2205A, Pre Fire Plan - Control Complex
AI-2205B, Pre Fire Plan - Turbine Building
AI-2205C, Pre Fire Plan - Auxiliary Building
AI-2205F, Pre Fire Plan - Miscellaneous Buildings and Components
SP-804, Surveillance of Plant Fire Brigade Equipment
HPP-502, Respiratory Equipment Inspection And Maintenance

Section 1R12: Maintenance Effectiveness

Nuclear Condition Reports

NCR 210023, RWSP-1B Flow Blockage
NCR 181301, UHS Margin Improvement Plan
NCR 170139, RWP-2B Upper Packing has Minimum Flush Flow
NCR 178271, EFP-2 did not trip during performance of PT-350
NCR 163731, MSV-187 has leaking seal weld plug

Maintenance Work Orders

Miscellaneous

RW and EF Maintenance Rule Event List and Monitoring Status Databases

Section 1R20: Refueling and Other Outage Activities

WCP-103, Station Readiness for Reduced Inventory, Mode 4/3 Entry, and Mode 2/1 Entry
AI-704, Reactor Trip Review and Analysis
CP-253, Power Operation Risk Assessment

Section 4OA2: Problem Identification and Resolution

Miscellaneous

Equipment Performance Action Plan for the RW system
System Health Report RW System- Road to Green
Nuclear Condition Report database

Section 4OA3: Followup of Events and Notices of Enforcement Discretion

Procedures

EOP-2, Vital System Status Verification
EOP-10, Post Trip Stabilization
EOP-14, Emergency Operating Procedure Enclosures
AI-704, Reactor Trip Review and Analysis
AP-545, Plant Runback