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

April 30, 2009

Mr. Dale E. Young
Vice President
Crystal River Nuclear Plant (NA1B)
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2009002

Dear Mr. Young:

On March 31, 2009, 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 13, 2009, with you and other members of your staff.

The inspection examined activities conducted under your license as they related 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 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 NCV in this report, 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.: 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 addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at Crystal River Unit 3. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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 the 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/

Marvin D. Sykes, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 50-302
License No. DPR-72

Enclosure: Inspection Report 05000302/2009002
w/Attachment: Supplemental Information

cc w/encls: (See page 3)

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Letter to Dale E. Young from Marvin D. Sykes dated April 30, 2009

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2009002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-302
License Nos:	DPR-72
Report No:	05000302/2009002
Licensee:	Progress Energy (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	Crystal River, FL
Dates:	January 1, 2009 – March 31, 2009
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector R. Aiello, Senior Operations Engineer (Section 1R11.2) R. Berryman, Senior Reactor Inspector (Section 1R17) S. Rose, Senior Operations Engineer (Section 1R17) D. Mas-Penaranda, Reactor Inspector (Section 1R17) J. Hamman, Reactor Inspector (Section 1R17) A. Alen, Reactor Inspector (trainee) (Section 1R17)
Approved by:	M. Sykes, Chief, Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2009002; 01/01/2009-03/31/2009; Crystal River Unit 3; Problem Identification and Resolution; Follow-up of Events and Notices of Enforcement Discretion.

The report covered a three month period of inspection by resident inspectors. Two Green findings and one Green NCV were identified. The significance of most findings is identified by their color (Green, White, Yellow, Red) using 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 4, dated December 2006.

A. NRC Identified & Self-Revealing Findings

Green. 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. (Section 4OA.2.2)

Green. 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.

Green. 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.

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status:

Crystal River 3 began the inspection period at 100 percent rated thermal power (RTP). On January 27, the unit was manually tripped after loss of two circulating water pumps, one condensate pump and one feedwater booster pump due the actuation of their undervoltage protection relays during maintenance. The unit was restarted on January 28. On January 29, reactor power was limited to approximately 57 percent RTP due to a damaged feedwater block valve. On February 2, the unit was shutdown in order to perform post-maintenance testing of the repaired feedwater block valve. On February 3, the unit was restarted and obtained 100 percent RTP on February 4. The unit operated at essentially 100 percent RTP for the remainder of the inspection period.

REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

During the periods listed below, the inspectors verified that the licensee implemented Administrative Instruction AI-513, Seasonal Weather Preparations, Sections 4.2 (Freezing Weather) and/or 4.3 (Freezing Weather Monitoring). The inspectors walked down portions of the emergency feedwater pump EFP-3; A and B emergency diesel generator (EGDG); and the alternate AC diesel generator systems to check for any unidentified susceptibilities to cold weather. Nuclear condition reports were reviewed to check that the licensee was identifying and correcting cold weather protection issues. This completed three samples for a site specific weather related condition.

- January 14 -17 with outside temperature near freezing (approximately 33 degrees (F) Fahrenheit)
- January 21 - 22 with outside temperature just below freezing (approximately 30 F)
- February 5 - 6 with outside temperature in the upper 20’s (F)

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment 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:

- Nuclear service water (SW) system using operating procedure OP-408, Nuclear Services Cooling System, while raw water (RW) pump RWP-3B and decay heat closed cycle cooling (DC) heat exchanger DCHE-1B were out of service for maintenance
- A train DC and RW systems using operating procedures OP-404, Decay Heat Removal System and OP-408, Nuclear Services Cooling System, while RWP-3B and heat exchanger DCHE-1B were out of service for emergent maintenance
- Emergency feed water pump EFP-3 using operating procedure OP-450, Emergency Feed Water System, while the emergency feed water pump EFP-2 was out of service for maintenance
- A train decay heat removal (DHR) system and DC system using OP-404, Decay Heat Removal System, while RWP-3B and DCHE-1B were out of service for planned maintenance

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted a detailed walkdown/review of the alignment and condition of emergency feed pump systems, specifically the turbine driven EFP-2 pump and the diesel driven EFP-3 pump. The inspectors used licensee operating procedure, OP-450, Emergency Feed Water System, as well as design documents, and reviewed the applicable portions of the Final Safety Analysis Report (FSAR) to verify proper system alignment. This completes one sample of a complete system alignment.

The walkdown 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
- Temporary scaffolding had been installed per station procedures

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. The system walkdown report, Administrative Instruction AI-1701, Quarterly Walkdown of the Emergency Feed System, dated January 19, 2009, was reviewed by the inspectors. This inspection sample was completed using the guidance listed in Operating Experience Smart Sample FY2009-02.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Area 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 FSAR Section 9.8, Plant Fire Protection Program, were met. Documents reviewed are listed in the attachment. The inspectors toured the following six areas important to reactor safety:

- Emergency feedwater initiation and control (EFIC) rooms
- High pressure injection area
- Cable spreading room
- Off-site power transformer, B emergency service transformer, and Main step-up transformer areas.
- A train DHR and building spray vault
- Emergency feedwater tank EFT-2 building

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

On February 18 and on February 26, the inspectors observed licensee fire brigade response to a simulated fire. The first fire drill involved a main step up transformer explosion and fire. The second drill was associated with a feedwater pump lube oil fire. 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, was 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. Documents reviewed are listed in the attachment. The inspectors observed that:

- 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 were completed
- 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 plan strategies were utilized
- The drill scenario was followed, and the drill acceptance criteria were met
- All fire fighting equipment was retuned to a condition of readiness

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On February 6, the inspectors observed and assessed licensed operator crew response and actions for the Crystal River Unit 3 licensed operator simulator evaluated session SES-12. Session SES-12 involved a loss of all service water raw water, a loss of off-site power, and loss of other safety-related equipment. The inspectors observed the operator's use of abnormal procedures AP-330, Loss of Nuclear Service Cooling and AP-770, Emergency Diesel Generator Actuation. Additionally, emergency operating procedures used during the scenario included EOP-02, Vital System Status Verification and EOP-04, Inadequate Heat Transfer. The operator's actions were verified to be in accordance with the above procedures. Event classification and notifications were verified to be in accordance with emergency management procedure EM-202, Duties of the Emergency Coordinator. The simulator instrumentation and controls were verified to closely parallel those in the actual control room. The inspectors attended the management crew critique and evaluation to verify the licensee had entered any adverse conditions into the corrective action program. The inspectors evaluated the following attributes related to crew performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of abnormal and emergency operation procedures; and emergency plan implementing procedures
- Control board operation and manipulation, including high-risk operator actions
- Oversight and direction provided by supervision, including ability to identify and implement appropriate technical specification actions, regulatory reporting requirements, and emergency plan classification and notification
- Crew overall performance and interactions

b. Findings

No findings of significance were identified.

.2 Annual Review of Requalification Examination Results by a Regional Specialist

a. Inspection Scope

On February 15, 2008, the licensee completed the requalification annual operating tests, required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the individual operating tests and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609, Appendix I, Operator Requalification Human Performance Significance Determination Process.

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. The review included 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 (MR) per 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The documents reviewed are listed in the attachment. The inspectors conducted this inspection for the following three issues:

- NCR 319944, Maintenance Rule A1 for B train RW/DC system
- NCR 309669, Feedwater valve FWV-30 failure to close
- NCR 280017, Main feedwater system entered into A1 due to flow oscillations

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 seven work week assessments.

- Work Week 09W01, Operations in yellow risk condition while raw water pump RWP-3B and heat exchanger DCHE-1B were out of service for emergent work

- Work Week 09W04, Operations in yellow risk condition while raw water pump RWP-3B and heat exchanger DCHE-1B were out of service for maintenance

- Work Week 09W05, Operations with emergency diesel generator EGDG-1B out of service for testing and separately, a reactor startup

- Work Week 09W06, Operations with emergency feed pump EFP-2 out of service, and separately, off-normal breaker alignment on the B emergency service 4160-volt bus to perform maintenance on breaker 3206

- Work Week 09W07, Operations with the A train control complex chiller out of service for planned maintenance and EGDG-1A out of service for testing

- Work Week 09W09, Operations in yellow risk condition, with the B train raw water system unavailable for planned maintenance

- Work Week 09W11, Operations with the A train EFIC air handler unavailable for planned maintenance, and the A emergency diesel generator out of service for testing

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

The inspectors reviewed the following six NCRs to verify 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 ITS, 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 316048, and NCR 265002, Emergency diesel engine water jacket leak
- NCR 316682, Raw water pump RWP-1 discharge check valve not fully closed
- NCR 313742, RWP-3B flowrate lower than required for in-service testing
- NCR 317302, Feedwater block valve (FWV-29) failed to open

- NCR 324146, Found partial o-ring in diesel jacket water valve DJV-112
- NCR 326879, Emergency feed valve EFV-148 full flow recirculation stuck closed

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility, FSAR, or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for six changes and additional information, such as drawings, calculations, supporting analyses, the FSAR, and ITS to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The six evaluations reviewed are listed.

The inspectors reviewed samples of changes for which the licensee had determined that evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The 18 "screened out" changes reviewed are in the attachment.

The inspectors evaluated engineering design change packages for nine material and design based modifications to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The nine modifications and the associated attributes reviewed are as follows:

EC 55315, Alternate AC Diesel Generator (Mitigating Systems)
- Timing
- Control Signals
- Equipment Protection
- Operations
- Licensing Basis

EC 59476, Reactor Building (RB) Sump Level Instrumentation Modification (Mitigating Systems)
- Post Modification Testing
- Energy Needs
- Material/Replacement Components
- Process Medium

EC 62722, MS Safety Relief Valve Manual Lifting Device Removal (Mitigating Systems)
- Equipment Protection
- Operations
- Failure Modes
- Licensing Basis
- Post Modification Testing

EC 63496, Domestic Water System (DO) System Re-Route Component Cooling (Mitigating Systems)

- Material/Replacement Components
- Licensing Basis
- Failure Modes
- Flowpaths
- Pressure Boundary
- Process Medium
- Seismic Qualification
- Post Modification Testing

EC 66385, CWTS-2 (Circulating Water Traveling Screens) Motor Trip Lever (Initiating Events)

- Post Modification testing
- Control Signals
- Materials/Replacement Components

EC 68587, Part 21 Replace MS-86-PT Transmitter (Mitigating Systems)

- Energy Needs
- Materials/Replacement Components
- Structural
- Process Medium
- Licensing Basis

EC 68926, SCP-2 (Secondary Service Closed Cycle Pump) Motor Replacement (Initiating Events)

- Post Modification Testing
- Energy Needs
- Material/Replacement Components
- Process Medium

EC 71126, Replace Existing FW-328-IB with new Signal Isolator, Converter for Input to Woodward 505 Controller for FWP-2A (Initiating Events)

- Materials/Replacement Components
- Pressure Boundary
- Licensing Basis
- Failure Modes

EC 71213, RWP-2A Weight Increase (Mitigating Systems)

- Equipment Protection
- Failure Modes
- Licensing Basis
- Post Modification Testing

Documents reviewed included procedures, engineering calculations, modification design and implementation packages, work orders, site drawings, corrective action documents, applicable sections of the living FSAR, supporting analyses, ITS, and design basis information. The inspectors additionally reviewed test documentation to ensure adequacy in scope and conclusion. The inspectors' review was also intended to verify that all details were incorporated in licensing and design basis documents and associated plant procedures.

The inspectors also reviewed selected NCRs and the licensee's recent self-assessment associated with modifications and screening/evaluation issues to confirm that problems were identified at an appropriate threshold were entered into the corrective action process, and appropriate corrective actions had been initiated and tracked to completion.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed two temporary modification listed below 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.

- EC 72314R0, FWV-29 Gagging Device (Stem Clamp)
- EC 71858, EGDG-1B lube oil temperature temporary setpoint change

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the one design change package listed below to verify it met the requirements of procedures EGR-NGGC-0003, Design Review Requirements and EGR-NGGC-0005, Engineering Change. The inspectors observed the as-built configuration of the modification and observed installation, and reviewed testing activities associated with the modification. Documents reviewed included surveillance procedures, design and implementation packages, work orders, system drawings, corrective action documents, applicable sections of the updated final safety analysis report, technical specifications, and design basis information. Post maintenance testing data and acceptance criteria were reviewed. The inspectors verified that issues found during the course of the installation and testing associated with the modification were entered and properly dispositioned in the corrective action program.

- EC 69433 DCP-1A motor replacement

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 an/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 six post-maintenance tests reviewed are listed below:

- SP-344A, SWP-1A and Valve Surveillance, after performing maintenance on RW check valve RWV-36 per work order (WO) 1410063

- SP-435, Valve Testing During Cold Shut Down, and OP-209T, Special Procedure Stoke Test FWV-29 Open/Closed, after performing maintenance on main feedwater block valve FWV-29 per WO 1490281

- SP-340D, RWP-3B, DCP-1B and Valve Surveillance, after performing maintenance on the B train raw water discharge piping and heat exchanger, per WO 1420529

- OP-404, Decay Heat Removal (DCP-1B 480V breaker only), after completing maintenance on the DCP-1B breaker per WO 221097

- SP-354C, Functional Test of The Alternate AC Diesel Generator EGDG-1C, after completing maintenance on the jacket water cooling system per WO 1330063

- SP-340D, RWP-3B, DCP-1B and Valve Surveillance, after performing maintenance on B train RW system per WO’s 1299351 and 1298950

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed six surveillance tests listed below to verify that ITS 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.

In-Service Test:

- SP- 340C, MUP-1A, MUP1B, and Valve Surveillance

Surveillance Test:

- SP-145A, EFIC Electronic Noise Transmitter Frequency Analysis Response Time Testing
- SP-112T, Reactor Protection System Main Turbine and MFP Anticipatory Trip Calibration
- SP-457A, ECCS Response to a Safety Injection Test Signal (Modes 1-3), (B portion only)
- SP-349C, EFP-3 and Valve Surveillance

Reactor Coolant system Leak Determination Test

- SP-317, RC System Water Inventory Balance

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed and reviewed two emergency response activities 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.

- February 6, license operator simulator evaluated session SES-12, involving the loss of service water raw water and off-site power
- March 30, unannounced staffing drill involving a steam generator tube rupture and an open atmospheric dump valve (limited scope drill involving only the technical support center (TSC) making classification, notifications and protective action recommendations)

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Initiating Events and Mitigating Systems Cornerstones

a. Inspection Scope

The inspectors checked licensee submittals for the PIs listed below for the period January 1, 2008 through December 31, 2008 to verify accuracy. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 5, were used to check the reporting for each data element. The inspector checked licensee events reports (LERs), operator logs, and daily plant status reports to verify the licensee accurately reported the data including the number of critical hours reported. 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
- Unplanned Power Changes per 7000 Critical Hours
- Unplanned Scrams with Complications

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Review

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify 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 NCR 307873 for a detailed review and discussion with the licensee. The NCR was written to address an EGDG-1B low lube oil temperature condition that resulted in the EGDG being declared inoperable. The inspectors verified the issue had been completely and accurately identified in the licensee's CAP, and that safety concerns were properly classified and prioritized for resolution, the apparent cause determination 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 licensee corrective action procedure CAP-NGGC-200, Corrective Action Program.

b. Findings and Observations

The inspectors determined that the investigation was not thorough. The licensee had determined that fire watches stationed in the EGDG room blocked open the door between the warm EGDG room and its cold radiator room causing both EGDG room and lube oil temperature to lower. This apparent cause was based on not finding any equipment problems and on an NRC inspector observation of finding the door blocked open two days after the event. No other corroborating evidence supported this conclusion. The inspectors questioned the licensee on why fire watches stationed in the EGDG room at the time of the event were not interviewed. As a result of this questioning, the licensee conducted interviews with the fire watches and determined that the door was routinely left open to support fire service maintenance and for the comfort of the fire watches. The additional supportive information was added to the investigation report. Also, as a result of the inspector's questions, the licensee added a corrective action enhancement to change administrative instruction AI-513, Seasonal Weather Preparations, to check closed EGDG doors as part of the cold weather checklist. A finding associated with the cause of the inoperable emergency diesel generator is documented below.

Introduction: A Green 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 EGDG operability. As a result, during cold weather conditions, an open door caused a low EGDG-1B lube oil temperature and inoperability of the EGDG.

Description: On November 22, 2008, the control room operators received the EGDG-1B low lube oil temperature announciator indicating a lube oil temperature of less than 115 degrees Fahrenheit (F). At the EGDG, the operators found the standby lube oil circulating pump secured and a lube oil temperature of 114 degrees F. A low temperature interlock turned the pump off as-designed for this low lube oil temperature condition. The operators, utilizing the EGDG operating procedure, bypassed this low temperature interlock and started the pump in manual to begin warming up the lube oil. As a result of mixing, lube oil temperature lowered to less than the minimum required temperature of 110 degrees F. The EGDG fuel racks were tripped and the EGDG was declared inoperable. During troubleshooting, the lube oil keep warm system was checked and verified to be operating properly. The standby lube oil pump and heaters were placed in service. Approximately 13 hours later, the fuel rack was reset and the EGDG was available for use. The EGDG lube oil system was monitored for proper operation for an additional 13 hours before the EGDG was declared operable. Since troubleshooting was unable to determine the cause of the low lube oil temperature condition, the licensee, on November 26, implemented a temporary modification to raise the lube oil set point 10 degrees F to add additional margin.

The licensee's investigation later determined that the cause of the low lube oil temperature condition was due to cold outside air (less than 40 degrees F) entering the diesel engine room from the radiator room through a propped open door. The lube oil keep warm system was not designed to maintain acceptable temperatures under these conditions. The licensee found that the door was open for periods of time to support fire piping modification in the diesel engine room and also was opened for the personal comfort of the continuous fire watches stationed in the diesel engine room.

Analysis: The performance deficiency associated with this finding was the failure to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support EGDG operability. 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. Using NRC Inspection Manual Chapter (IMC) 0609.04, Significance Determination Process (SDP), Phase I worksheet, the finding was determined to be 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 technical specifications, 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. 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.

Enforcement: The failure to have adequate controls in place to ensure the temperature of the emergency diesel room was maintained to support EGDG operability 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 307873. This finding is identified as FIN 05000302/2009002-01, Failure to Have Adequate Controls in Place to ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability.

4OA3 Follow-up 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 plant procedures.

- January 27, manual reactor trip in accordance with emergency operating procedures EOP-2, Vital System Status Verification; EOP-10, Post-Trip Stabilization and EOP-14, Emergency Operating Procedure Enclosures (enclosures 1 and 2 post-trip actions)
- January 28-29, Reactor startup and power ascension in accordance with operating procedures OP-210, Reactor Startup and OP-204, Power Operations
- February 2-3, Reactor startup and power ascension in accordance with operating procedures OP-210, Reactor Startup and OP-204, Power Operations

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 05000302/2008-003-00: Manual Reactor Trip due to Main Feedwater System Oscillations Caused by an Inconsistent Procedure

(Closed) LER 05000302/2008-003-00-01: Manual Reactor Trip due to Main Feedwater System Oscillations Caused by an Inadequate Design

The LER documented that a manual reactor trip was initiated due unstable feedwater flow oscillations. The inspectors reviewed the LER and NCR 293080 documenting the event. The inspectors checked the accuracy and completeness of the LER and the appropriateness of the licensee's corrective actions. No findings of significance or violations of NRC requirements were identified.

.3 (Closed) LER 05000302/2008-004-00: Motor-Operated Main Feedwater Isolation Valve Inoperable due to Motor Rotor Oxidation/Corrosion

a. Inspection Scope

The inspectors reviewed the root cause evaluation associated with LER 05000302/2008-004-00 to determine whether a performance deficiency was involved, corrective actions were adequate, and the safety significance.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance (Green) involving a non-cited violation (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. The valve was found to be inoperable for a period of time longer than allowed by Improved Technical Specifications (ITS).

Description. On December 5, 2008, during a planned reduction of reactor power, MFIV FWV-30 failed to automatically close when it received a closed signal. The valve was declared inoperable and ITS 3.7.3, condition A was entered. Condition A required the affected flow path to be isolated within 72 hours. This condition was satisfied after the motor-operated valve (MOV) was manually closed.

The licensee determined the root cause of the FWV-30 failure to be oxidation and corrosion of the magnesium rotor in the MOV's actuator. Based on the amount of degradation of the magnesium rotor, the licensee determined that FWV-30 became inoperable sometime between its last successful operation on August 29, 2008 and when it failed to close on December 5, 2008. The licensee determined that it was likely that the valve was inoperable for greater than the 72 hours allowed by ITS.

Crystal River Unit 3 experienced an identical failure in October 2005 when MFIV FWV-29 failed to automatically close (reference LER 05000302/2005-004-00 and NRC inspection report IR 05000302/2006-002). Since non-magnesium rotors were not available for FWV-29 and FWV-30, corrective actions to inspect/monitor the magnesium rotors were implemented. The intent of the inspection program was to replace motors that exhibited early signs of magnesium rotor corrosion prior to any failures.

On November 20, 2006, NRC Information Notice (IN) 2006-26, Failure of Magnesium Rotors in Motor-Operated Valve Actuators, was issued. IN 2006-026 details several examples of magnesium rotor failures in the industry including the Crystal River Unit 3 FWV-29 failure. The IN identified the need to have an adequate magnesium rotor inspection/preventative maintenance program in place. The licensee entered IN 2006-26 into their operating experience program as action request (AR) 214943. Licensee review of IN 2006-26 identified the need for additional training for inspectors, enhanced inspection criteria, and the use of high quality bore scope equipment to ensure that inspections will identify magnesium rotor degradation. These actions were not implemented in time for the FWV-30 inspection during the fall 2007 refueling outage. As a result, there was a missed opportunity to identify magnesium rotor degradation during the inspection.

Analysis. The inspectors determined that the failure to implement corrective actions associated with operating experience item AR 21493 (IN 2006-026) in time for the fall 2007 FWV-30 inspection was a performance deficiency that resulted in a repeat failure of a MFIV due to magnesium rotor corrosion. The inspectors determined 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 Significance Determination Process (SDP) Phase 2 analysis was required. Based upon the Phase 2 results, a regional Senior Reactor Analyst performed a Phase 3 evaluation. The NRC's most current Standardized Plant Analysis Risk Model for Crystal River Unit 3 was used in the evaluation with the basic events for failing to isolate a faulted Steam Generator set to always fail. Since an additional check valve, not in the computer model, must fail to cause the loss of this function, a hand calculation was used to include its failure in the applicable accident sequences. The dominant accident sequence involved a steam generator tube rupture, failure to isolate the rupture generator and failure of the decay heat removal system through various human and equipment failures. Limiting the significance of the performance deficiency was that another valve, other than FWV-30, had to fail before a ruptured Steam Generator could not be isolated. An exposure time of 49 days was used in the 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. Corrective actions for the failure of FWV-30 include: installing a new motor; develop and implement engineering changes to replace the station's MOV magnesium rotor motors with aluminum rotor motors (when available); ensure the engineering staff is trained on effective correction action plans; and revise MOV maintenance procedures to include those actions of operating experience item AR 214943 prior to the next MOV inspections.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires, in part, that in the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. Contrary to the above, corrective actions associated with IN 2006-26 (AR

214943) were not implemented in a timely manner to preclude a second failure of a MFIV due to magnesium rotor corrosion. Specifically, training and inspection requirements identified by the licensee in operating experience item AR 214943 necessary to identify magnesium rotor degradation were not implemented in time for the inspection of FWV-30 in fall 2007. As a result, FWV-30 failed due to magnesium rotor degradation and was found inoperable on December 5, 2008. Because the finding is of very low significance and has been entered into the licensee's CAP as NCR 309669, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000302/2009002-02, 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.

.4 (Closed) LER 05000302/2009-001-00: Manual Reactor Trip due to Loss of A 4160V Unit Bus Loads Caused by Incorrectly Connected Test Leads

a. Inspection Scope

The inspectors reviewed the root cause evaluation associated with LER 05000302/2009-001-00 to determine whether a performance deficiency was involved, corrective actions were adequate and to determine the safety significance. The inspectors also reviewed the LER to verify its accuracy and completeness.

b. Findings

Introduction. A Green self-revealing finding was identified for the failure to follow procedure HUM-NGGC-0001, Human Performance Program, which required workers to perform self checks 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.

Description. On January 27, 2009, the reactor was manually tripped when reactor coolant system (RCS) pressure was observed increasing towards the automatic trip set point. Emergency operating procedure, EOP-2, Vital System Status Verification, was entered and the unit was stabilized in Mode 3, Hot Standby.

The licensee investigation revealed that relay technicians performing voltage checks to support meter calibrations, incorrectly connected a Doble relay test set across the secondary side of the non-safety A Unit 4160V bus' potential transformer (PT) resulting in a short circuit. The short circuit caused two fuses to blow on the secondary side of the bus' PT and an indication of a loss of bus voltage. The bus' undervoltage motor protection relays actuated causing loss of the bus loads including a feedwater booster pump. RCS pressure increased due to the reduction of feedwater flow.

The licensee determined there were two root causes associated with this event. The first root cause was the inadequate use of human performance tools as specified in HUM-NGGC-0001, Human Performance Program. Specifically, the relay technicians failed to properly utilize self and peer checking to ensure the leads from the multimeter and not the Doble relay test set were connected during voltage checks. In addition, the performance of the voltage check was outside of the work order guidance. The second root cause was determined to be inadequate closure of corrective actions for a similar 2004 event documented in NCR 133661. In 2004, a Doble test set was incorrectly used during voltage checks during similar meter calibrations resulting in one blown fuse and a loss of meter indication. One of the corrective actions for the 2004 event was to complete a risk assessment to determine whether this type of work should be completed with the unit shutdown. The licensee determined that the assessment was completed; however it should have resulted in moving relay work activities that could cause plant transients to an off-line or outage activity.

Analysis. The inspectors determined that the failure to follow procedure HUM-NGGC-0001 was a performance deficiency and therefore a finding. This was selected as the performance deficiency since it was more reflective of current licensee performance than the inadequate closure of corrective actions in 2004. 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, Attachment 5, Revision 2, resulted in the connection of incorrect test equipment, the loss of several secondary plant pumps and ultimately led to manual reactor trip. Using NRC Inspection Manual Chapter (IMC) 0609.04, Significance Determination Process (SDP), Phase I 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. 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. Corrective actions for the plant trip event 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.

Enforcement. No violation of NRC regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because the performance deficiency involved non-safety related equipment. This finding was determined to be of very low safety significance (Green) and was entered into the corrective action program as NCR 316543. This finding is identified as FIN 05000302/2009002-03, Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Finding

No findings of significance were identified.

.2 (Closed) Temporary Instruction (TI) 2515/176, EDG TS Surveillance Requirements Regarding Endurance and Margin Testing

Inspection activities for TI 2515/176 were previously completed and documented in inspection report 05000302/2008004, and this TI is considered closed at the Crystal River Nuclear Plant; however, TI 2515/176 will not expire until August 31, 2009. The information gathered while completing this temporary instruction was forwarded to the Office of Nuclear Reactor Regulation for review and evaluation.

4OA6 Exit

.1 Exit Meeting Summary

On April 13, 2009, the resident inspectors presented the inspection results to Mr. D. Young, Site Vice President and other members of licensee management. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On April 13, 2009, the Senior Resident Inspector met with Mr. Dale Young and other members of the licensee staff to discuss the NRC's annual assessment of the Crystal River Nuclear Plant's safety performance for the period of January 1 through December 31, 2008. The annual assessment results were previously provided to Progress Energy/Florida Power Corporation (FPC) via letter dated March 4, 2009.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee personnel:</u>
J. Holt, Plant General Manager
W. Brewer, Manager, Maintenance
S. Cahill, Manager, Engineering
P. Dixon, Manager, Nuclear Assessment
J. Franke, Director of Site Operations
R. Hons, Manager Training
C. Morris, Manager, Operations
D. Westcott, Supervisor, Licensing
B. Akins, Superintendent, Radiation Protection
S Gangi, Supervisor, Emergency Preparedness (interim)
I. Wilson, Manager Outage and Scheduling
D. Young, Vice President, Crystal River Nuclear Plant

<u>NRC personnel:</u>
M. Sykes, Chief, Branch 3, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED

<u>Opened and Closed</u>

05000302/2009002-01	FIN	Failure to Have Adequate Controls in Place to Ensure the Temperature of the Emergency Diesel Room was Maintained to Support EGDG Operability (Section 4OA2.2)
05000302/2009002-02	NCV	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 (Section 4OA3.3)
05000302/2009002-03	FIN	Inadequate Peer and Peer Checking Resulted in Connecting Improper Test Equipment and a Manual Plant Trip (Section 4OA3.4)

<u>Closed</u>

05000302/2008003-00-00	LER	Manual Reactor Trip due to Main Feedwater System Oscillations Caused by an Inconsistent Procedure (Section 4OA3.2)

05000302/2008003-00-01	LER	Manual Reactor Trip due to Main Feedwater System Oscillations Caused by an Inadequate Design (Section 4OA3.2)
05000302/2008004-00	LER	Motor-Operated Main Feedwater Isolation Valve Inoperable due to Motor Rotor Oxidation/Corrosion (Section 4OA3.3)
05000302/2009001-00	LER	Manual Reactor Trip due to Loss of a 4160V Unit Bus Loads Caused by Incorrectly Connected Test Leads (Section 4OA3.4)
2515/176	TI	EDG TS Surveillance Requirements Regarding Endurance and Margin Testing (Section 4OA5)

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 174428, FWV-29 Failure due to Magnesium Rotor Corrosion
NCR 317302, Feed Water Valve FWV-29 Failed to Stroke Open

Section 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

Full Evaluations

Action Report (AR) 222154, Support AHF-1C NCON for AR218670, dated February 12, 2007
AR 236919, Change in Analysis of Record for Containment Pressure, dated June 26, 2007
AR 242159, Evaluation of EOP-14 EOP Enclosure Rev. 21, dated August 22, 2007
AR 243429, 50.59 Review for CR3C16 Core Reload EC 64022, dated November 25, 2007
AR 251797, CP-14 Operations Evolution 2007-10-02, dated October 24, 2007
AR 276295, FSAR Change Request for Section 8.2.3.1.3, dated April 23, 2008

Screened Out Items

AR 210754, Perform a 50.59 Determination on EOP-3 Rev. 13, dated November 17, 2006
AR 210756, Perform 50.59 for EOP-4 Inadequate Heat Transfer Rev. 12, dated November 17, 2006
AR 210757, Perform 50.59 for EOP-7 Inadequate Core Cooling Rev. 13, dated November 11, 2006
AR 210758, Perform 50.59 for EOP-8 LOCA Cooldown Rev. 16, dated November 24, 2006
AR 210759, Perform 50.59 for EOP-14 EOP Enclosures Rev. 17, dated November 18, 2006
AR 219134, 10CFR 50.59 Screen for EC 63029, dated February 26, 2007
AR 225589, 50.59 Screen for Revision to FSAR 1.2.7 Snubbers, dated March 13, 2007
AR 229733, EC 66695 TBP-9 Motor Replacement, dated April 19, 2007
AR 237323, Elimination of Spent Fuel Pool Tornado Missile Shielding
AR 237595, Replace Relays “CR”, “LS-13”, and “74” in HPI MCC’s MUMC-1 and MUMC-2, dated December 19, 2007
AR 238901, EC 67387 FWM-1B Motor Replacement, dated September 19, 2007
AR 249222, RPS Calibration Procedure Revision
AR 268779, EC 69541 Correct Beede Meter Input Impedance, dated March 5, 2008
AR 269004, 50.59 Screen for Temporary EC 69549, RW Pipe Internal Coating Repair, dated March 5, 2008
AR 291250, 50.59 Evaluation of the Proposed Change to FSAR Table 1-3, dated August 18, 2008
AR 296890, GSF-1A/B (Gland Steam Fan) Motor Replacement
AR 295516, Screen to Support Bases Change 08-09, dated September 16, 2008
AR 302326, Screen for ITS Bases Change B08-10, dated October 23, 2008

Engineering Changes

EC 55315, Alternate AC Diesel Generator, Rev. 13
EC 59476, Reactor Building (RB) Sump Level Instrumentation Modification, Rev. 18
EC 62722, MS Safety Relief Valve Manual Lifting Device Removal, Rev. 2
EC 63496, Domestic Water System (DO) System Re-Route Component Cooling, Rev. 1
EC 64487, Remove the Need to Install Missile Shields over the Spent Fuel Pool
EC 66385, CWTS-2 (Circulating Water Traveling Screens) Motor Trip Lever, Rev. 0
EC 68587, Replace MS-86-PT Transmitter, Rev. 0
EC 68926, SCP-2 (Secondary Service Closed Cycle Pump) Motor Replacement, Rev. 0
EC 69549, RW Pipe Internal Coating Repairs, Rev. 0
EC 71126, Replace Existing FW-328-IB with new Signal Isolator/Converter for input to Woodward 505 Controller for FWP-2A, Rev. 1
EC 71213, RWP-2A Weight Increase, Rev. 0
EC 71278, Replacement Motor for GSF-1A
EC 71279, Replacement Motor for GSF-1B

Condition Reports

NCR 37777, RWP-2A Bearing Horizontal Vibration Identified to be in Alert Range, dated February 9, 2000
NCR 94259, RWP-2A Failed to meet SP-344A IST Criteria, dated May 22, 2003

NCR 175822, Test procedure needs to be updated
NCR 176122, Incorrect direction in procedure
NCR 185374, Change surveillance frequency
NCR 215401, Add calibration data sheets
NCR 237108, Non-Conservatism Identified in Peak RB Pressure Analysis, dated June 20, 2007
NCR 253367, Snubber RCH-69 Fluid Not Visible in Viewport, dated November 5, 2007
NCR 253725, Snubber RCH-530 Fluid Level Not Visible in Sightglass, dated November 7, 2007
NCR 253727, Snubber RCH-73 Fluid Level Not Visible in Sightglass, dated November 7, 2007
NCR 253841, MSH-147 Fluid Level Low, dated November 7, 2007
NCR 253886, Snubber FWH-164 Fluid Level is not Visible, dated November 8, 2007
NCR 271622, RWP-2A in Alert Range per SP-344A, dated March 22, 2008
NCR 293080, Manual Reactor Trip Following A CDP Uncouple, dated August 24, 2008
NCR 293104, Loss of Signal Alarm on a MFWP Controller, dated August 25, 2008
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