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

April 30, 2008

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 05000302/2008002

Dear Mr. Young:

On March 31, 2008, 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 7, 2008, with Mr. D. Young 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, one NRC-identified and one self-revealing finding of very low safety significance (Green) were identified. The NRC-identified finding 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. Also, three licensee identified violations which were of very low safety significance are listed in Section 4OA7 of the report. If you contest the 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 20555-001; with copies to the Regional Administrator 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 1- 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 05000302/2008002
w/Attachment: Supplemental Information

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Letter to Dale E. Young from Marvin D. Sykes dated April 30, 2008

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2008002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-302
License Nos.:	DPR-72
Report No:	05000302/2008002
Licensee:	Progress Energy (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	Crystal River Florida
Dates:	January 1, 2008 – March 31, 2008
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector R. Aiello, Senior Operations Inspector (Section R11.2 and 4OA2.3) B. Caballero, Operations Inspector (Section R11.2 and 4OA2.3) L. Lake, Senior Reactor Inspector (Sections 4OA2.4 and 4OA5) J. Rivera-Ortiz, Reactor Inspector (Section 4OA5) B. Collins, Reactor Inspector (In Training)
Approved by:	Marvin D. Sykes, Chief, Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2008002; 01/01/2008 – 03/31/2008; Crystal River Unit 3; Fire Protection; Problem Identification and Resolution.

The report covered a three month period of inspection by resident inspectors and region based reactor inspectors. 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. Inspector Identified & Self-Revealing Findings

Cornerstone: Initiating Events and Mitigating Systems

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

The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater than minor because it is associated with the protection against external factors attribute, i.e., fire, and degraded the mitigating systems cornerstone objective to ensure the availability of systems that respond to initiating events. Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the finding was determined to have a very low safety significance since the gap in the fire penetration seal was small (less than 1/8 inch in width). (Section 1R05.1)

Green: A self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control switch during maintenance activities. As a result of bumping the control switch, a condensate pump had to be secured and reactor power was rapidly reduced to 61 percent to prevent a reactor trip. Corrective actions included removing the control switch handle to prevent it from being bumped.

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

B. Licensee Identified Violations

Three violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

The unit began the inspection period at 100 percent rated thermal power (RTP). On January 8th, the unit power was manually reduced when flow from the B condensate pump (CDP) was lost due to a degraded pump controller. After the CDP was placed on its spare controller, the unit was returned to 100 percent RTP on January 9th. Power was reduced to approximately 63 percent RTP on January 27th to complete CDP controller repairs. The unit was returned to 100 percent RTP on January 28th. On March 1st the unit was shut down to replace the seal for reactor coolant pump RCP-1C which had been experiencing increased seal leakage. The unit was restarted on March 20th and resumed 100 percent RTP on March 22nd. 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 2nd and February 14th, when outdoor temperatures were expected to fall 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/or 4.3 (Freezing Weather Monitoring). The inspectors walked down portions of the emergency feedwater pump EFP-3 and 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. Two inspection samples for a site specific weather related condition were completed.

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 three partial system alignments in system walkdowns using the listed documents:

- Nuclear service water and raw water (RW) systems using OP- 408, Nuclear Services Cooling System, while feedwater pump FWP-7, the alternate AC emergency diesel EGDG-1C, and the MTSW-2G buss were out of service for maintenance.

- Emergency feedwater pump EFP-2 using OP- 450, Emergency Feedwater System, and feed water pump FWP-7 using OP- 605, Feedwater System, while the emergency feedwater pump EFP-3 was out of service for maintenance.

- Train A decay heat closed cycle (DC) and decay heat removal (DHR) systems using OP-404, Decay Heat Removal System, while B train emergency core cooling systems (DC, DHR and RW) were out of service for 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 both trains of the control complex chilled water system. The inspectors used licensee operating procedure, OP-409, Plant Ventilation 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

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 Walk down of the control complex chilled water system, dated January 07, 2008, was reviewed by the inspectors.

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 licensee implementation of the fire protection program. The inspectors checked that the inspected 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 reviewed the area fire plans to verify the plans contained the hazards and fire protection defense-in-depth features in the fire areas. 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 feed pump EFP-3 building
- A and B Emergency diesel generators (EGDG) engine rooms and associated local engine control rooms
- 4160-Volt switch gear rooms, and emergency battery rooms
- Vital battery inverter rooms
- Makeup pump valve alley
- Spent Fuel Pool area

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) of Crystal River Unit 3 Operating License Condition 2.C(9), fire protection program. The NCV was associated with an inoperable fire penetration seal in the 3-hour fire rated ceiling of the makeup system valve alley.

Description: On February 22, during a walkdown of the auxiliary building 119' elevation in the vicinity of the spent fuel cooling pumps (fire zone AB-119-6B), the inspectors observed that the silicon foam sealant around spent fuel coolant piping in 3-hour rated fire penetration PAB-61 had partially pulled away from the pipe. Closer examination revealed a through-penetration gap (less than 1/8 inch in width) between the pipe and the silicon foam. The gap was approximately 5 inches around the circumference of the pipe. Light was observed passing through the penetration from the makeup system valve alley below (fire zone AB-95-3D). The licensee declared fire penetration PAB-061 inoperable; initiated an hourly fire watch in accordance with Fire Protection Plan, Table 6.7a; and documented the condition in the corrective action program.

Analysis: The inoperable fire penetration seal represented a licensee performance deficiency since the gap in the seal would be expected to be identified and corrected by the licensee to the criteria specified in SP-407, Fire and Flood Barrier Penetration Seals. The finding adversely affected the fire confinement capability defense-in-depth element. The finding is greater than minor because it is associated with the protection against external factors attribute, i.e., fire, and degraded the mitigating systems cornerstone objective to ensure the availability of systems that respond to initiating events. With this degraded fire penetration, a fire in zone AB-119-6B could impact components in the makeup system valve alley that are utilized to respond to initiating events.

Using NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process, the inspectors assessed the defense-in-depth (DID) element of fire barrier degradation in the fire confinement category. Since the gap in the silicon foam fire penetration seal was small (less than 1/8 inch in width), the degradation level was categorized as low (IMC 0609, Appendix F, Attachment 2, Table A2.2). IMC 0609, Appendix F, Attachment 1, Task 1.3.1, Qualitative Screening for All Finding Categories, showed that the finding was of very low safety significance (Green) due to the low degradation rating.

Enforcement: Crystal River Unit 3 Operating License Condition 2.C(9) requires, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program. The Crystal River Unit 1 Fire Protection Plan, revision 25, Section 6.5.1.3, Penetration Seals, specifies that surveillance procedure SP-407, Fire and Flood Barrier Penetration Seals, revision 33, be utilized to ensure the seal functions as an approved fire barrier. SP-407, section 3.6.2.1, states, in part, that there will be NO passage of light through the sealant.

Contrary to the above, on February 22, 2008, the inspectors found a through-penetration gap that allowed passage of light through fire seal PAB-061. The licensee initiated an hourly fire watch and documented the inoperable fire seal in the corrective action program as NCR 267299. Repairs to the fire seal were completed on February 29, 2008. Because this finding is of very low safety significance and was entered into the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding is identified as NCV 05000302/2008002-01, Inoperable Fire Barrier Penetration Seal.

.2 Annual Fire Drill

a. Inspection Scope

On January 3 and January 11 the inspectors observed the two licensee fire brigades responses to a simulated fire on feedwater pump FWP-2A. In each case, the inspectors checked 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 brigades 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 AL-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. 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 plan strategies were utilized.
- The drill scenario was followed, and the frill 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 January 24th the inspectors observed and assessed licensed operator continuing training re-qualification activities. The simulated events were done using the licensees' plant specific simulator per Simulator Examination Scenario SES-51, Loss of All Feedwater and Loss of Off-site Power. The inspectors observed the operator's use of abnormal operating procedure AP-770, Emergency Diesel Generator Actuation; EOP-02, Vital System Status Verification; and EOP-04, Inadequate Heat Transfer. The operator's actions were checked to be in accordance with licensee procedures. Event classifications (including General Emergency) were checked for proper classification and protective action recommendations. The licensee simulated emergency plan notifications. The simulator board configurations were compared with actual plant control board configurations concerning recent plant modifications. The inspectors specifically evaluated the following attributes related to operating 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 off-normal 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 Biennial Review by Regional Specialist

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of February 4 - 8, 2008, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program."

The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1998, "American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination." The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, Job Performance Measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records and performance test records, the feedback process, licensed operator qualification records, remediation plans, watchstanding, and medical records. The records were inspected using the criteria listed in Inspection Procedure 71111.11. One licensee identified violation associated with an operator not complying with a newly issued license condition to take prescribed medication while performing licensed duties is documented in Section 4OA7. Documents reviewed are listed in the attachment.

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 per 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. Documents reviewed are listed in the attachment. The inspectors conducted this inspection for two degraded equipment conditions listed below:

- NCR 260238, Incore Monitoring system exceeds Maintenance Rule performance criteria
- NCR 264799, Emergency diesel generators exceed Maintenance Rule performance criteria

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 evaluated 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 four work week assessments.

- Work Week 08W01, risk assessment for operation with B train control complex chiller (CHHE-1B) out of service for maintenance, and emergent work when the B train CDP failed causing control room operators to perform a rapid power decrease to 65 percent power.

- Work Week 08W02, risk assessment for operation with feedwater pump FWP-7 and C emergency diesel generator out of service for planned maintenance.

- Work Week 08W05, risk assessment for operation in yellow risk condition while the B train emergency core cooling system (ECCS) system was out of service for planned maintenance.

- Work Week 08W08, risk assessment for pre-outage maintenance activities alternating between the B train and A ECCS train work in the same work week.

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 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 262206, reactor incore closure leakage
- NCR 258531, increased RCP-1C seal leakage
- NCR 259782, possible degradation of RWV-22
- NCR 265002, EGDG 1B jacket water leak
- NCR 268933, Indication found in DHR system drop line weld
- NCR 268696, RWV-36 Failed the as-found inspection for the two year PM

1R18 Plant Modifications

1. Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the three design change packages listed below to verify they 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.

- Engineering change (EC) 68977, Butterfly Valve Replacement with a 800-lb Heavier Valve (RWV-22)
- EC 68963, Makeup Tank Temperature Band Change
- EC 65043, Electrical Circuit Configuration Change needed to Limit the Unavailability on EGDG-1B during SP-907B

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 five post-maintenance tests reviewed are listed below:

- SP-375, CHP-1B and Valve Surveillance, after performing maintenance on the CHHE-1B control complex chiller per work orders (WO's) 1128980 and 881927

- SP-354C, Functional Test of the Alternate AC Diesel Generator EGDG-1C, after performing emergent maintenance on the lubricating oil system, per WO 1279938

- SP-340D, RWP-3B, DCP-1B and Valve Surveillance, after performing maintenance on the emergency core cooling bravo train, per WO 1142772
- EC 65043 TP1 and TP2, EC Functional Test Procedure for Electrical Circuit Configuration Needed to Limit the Unavailability on EGDG-1B during SP-907B after EC installation per WO 1098715
- SP-344A, RWP-2A, SWP-1A and Valve Surveillance, and maintenance procedure MP-550, Maintenance of Anchor/Darling Swing Check Valves, after replacing the disk assembly of raw water pump RWP-1 discharge check valve RWV-36 per WO 1032736.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

1. Planned Outage to Replace RCP-1C Seal

a. Inspection Scope

The inspectors reviewed the licensee's outage risk assessment report to confirm the licensee had appropriately considered risk, industry experience, and previous site-specific problems, in developing and implementing the outage plan. During the forced outage, the inspectors observed and monitored licensee controls over the outage activities listed below. Documents reviewed are listed in the Attachment.

- Outage related risk assessment monitoring
- Controls associated with shutdown cooling, reactivity management, electrical power alignments, containment closure and integrity, and spent fuel pool cooling
- Implementation of equipment clearance activities
- Reduced inventory activities
- Reactor mode changes
- Reactor heat-up and pressurization
- Containment cleanup and closeout inspection
- Reactor startup
- Reactor power ascension and related testing

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 six activities were observed/reviewed:

In-Service Test:

- SP-340E, DHP-1B, BSP-1B and Valve Surveillance
- SP-349B, EFP-2 and Valve Surveillance

Surveillance Tests:

- SP-113G, Power Range Nuclear Instrument Gain Adjustment
- SP-354B, Monthly Functional Test of the Emergency Diesel Generator EGDG-1B
- SP-332, Monthly Steam Line and Feedwater Isolation Functional Test

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 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.

- January 24, licensed operator simulator evaluated session SES-51, Loss of all Feedwater and Loss of Off-site Power.

b. Findings

No findings of significance were identified

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Initiating Events Cornerstone

a. Inspection Scope

The inspectors checked licensee submittals for the PIs listed below for the period January 1, 2007 through December 31, 2007 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 resolve.

- 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 to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a screening of items entered daily into the licensee's corrective action program. This review was accomplished by reviewing daily printed summaries of condition reports and/or by reviewing the licensee's electronic condition report database.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected NCR-252450 for a detailed review and discussion with the licensee. The NCR was written to address the failure of a condensate pump which resulted in a reactor power decrease to 61 percent. 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 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 the licensee's CAP as delineated in corrective action procedure CAP-NGGC-200, Corrective Action Program.

b. Findings and Observations

The inspectors found that the licensee's review of the failure of the condensate pump and corrective actions were comprehensive and thorough. A finding associated with the cause of the condensate pump failure is documented below.

Introduction: A Green self-revealing finding was identified for failure to prevent inadvertent bumping of the condensate pump control switch during pre-outage maintenance activities. Consequently, after the switch was bumped, condensate pump CDP-1B had to be secured and control room operators rapidly reduced reactor power to 61 percent to prevent a reactor trip.

Description: On October 29, 2007, with the unit at 100 percent power, the control room was notified that smoke was observed coming out of the condensate control cabinet. The cabinet is located in the area near the condensate pump. Control room operators noticed erratic pump and motor control indications for the 1B condensate pump. The pump was tripped and a rapid power decrease to 61 percent power was performed. The licensee's investigation found the condensate pump control switch in the mid-position. The most likely cause was inadvertently bumping as a result of pre-outage maintenance activities. Those maintenance activities included pulling cables through a conduit in the overhead above the condensate control cabinet. Review of the circuit showed that the control circuitry would heat up if the switch was taken to a mid-position. The circuitry was not designed to operate correctly with the switch in a mid-position. Consequently, when the switch was bumped, circuitry in the control cabinet overheated resulting in loss of control of the 1B condensate pump.

Analysis: The performance deficiency associated with this finding was for failure to prevent inadvertent bumping of the condensate pump control switch. The finding was more than minor since it affected the equipment performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenged critical safety functions. Using the NRC manual Chapter 0609.04, “Significant 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 Human Performance, with a work control component (H.3.a). Specifically the licensee did not appropriately plan work activities and adequate protection was not taken around the condensate pump control switch, leaving the switch handle exposed to being bumped during maintenance activities. This caused a loss of operational control of the CDP-1B condensate pump, and subsequently required the pump to be secured, and directly led to a reactor rapid power reduction. Corrective actions included removing the switch handle from the control cabinet to prevent the switch handle from being bumped. Additionally, the licensee conducted additional walk downs to look for any plant equipment that may be vulnerable to inadvertent contact.

Enforcement: The failure to prevent inadvertent bumping of the condensate pump control switch during pre-outage maintenance activities 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 252450. This finding is identified as FIN 05000302/2008002-02, Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient.

.3 Annual Sample Review

a. Inspection Scope

The inspectors selected NCRs 261472 and 262631 for a detailed review. The NCRs were initiated because operations had two reactivity management events where verification requirements were not performed to the standards of AI-500, Conduct of Operations Department Organization and Administration, Rev 140. The first event, which occurred on January 10, 2008, resulted in a rod index of approximately 3 percent lower than expected after an addition to the Makeup Tank (MUT) prior to the performance of SP-317, Reactor Coolant System Leak Rate Determination. An incorrect final value of Xenon was utilized in the calculation supporting the addition. The individual performing the independent verification of the calculation failed to identify the error prior to the addition. The second event, which occurred on January 18, 2008, involved an incorrect addition of Demineralized Water (DW) to the MUT even though the OP-304, Soluble Poison Concentration Control, calculation required the use of water from the "C" Reactor Coolant Bleed Tank (RCBT). This error resulted in a projected rod index of approximately 5 percent lower than expected for the calculation. This second event occurred because the operator positioned the Feed Selector Switch to the “DW” position instead of the “C RCBT” position. The operator, who performed the concurrent verification of the Feed Selector Switch position prior to completing the addition, did not detect the error. The inspectors checked that the issue had been accurately identified in the licensee’s corrective action program, and that safety concerns were properly classified, prioritized for resolution, and apparent cause determinations were sufficiently thorough.

The inspectors also evaluated the NCRs using the requirements corrective action procedure CAP-NGGC-200, Corrective Action Program.

b. Findings and Observations

Appropriate corrective actions were actively being implemented in a manner consistent with safety and compliance with plant Technical Specifications (TS) and 10 CFR 50. These corrective actions had not been completed at the time of this report.

.4 Annual Sample Review

a. Inspection Scope

The inspectors conducted a detail review of NCR 268933, (Identification of Unacceptable Indication on DHR Dissimilar Metal Weld to reactor coolant hot leg), NCR 261078, (Investigation into not meeting Material Reliability Program (MRP)-139), and NCR 256333, (Unacceptable indications on Weld overlay installed on PZR surge line to Hot Leg dissimilar metal weld). The licensee had to repeat UT examinations on the decay heat removal hot leg nozzle during a forced outage because they had not achieved greater than 90 percent coverage during RFO 15.

As a result of this re-examination, the licensee identified an indication in the dissimilar metal weld that required a full structural weld overlay repair. Included in the review was the safety evaluation and management approval process for approving the installation of the weld overlay repair on the DHR dissimilar metal weld with the DHR system in service. The inspectors checked that the issues had been completely and accurately identified in the licensee's corrective action program, and that safety concerns were properly classified and prioritized for resolution, apparent cause determination was sufficiently through, and appropriate corrective actions were implemented in a manner consistent with safety and compliance with plant technical specifications, 10CFR50 and repairs conducted in accordance with the requirements of 10CFR50.55a and the approved relief request for repairing the DHR dissimilar metal weld.

b. Findings and Observations

The inspectors determined that the licensee's review of issues and corrective actions associated with the examinations of the dissimilar metal welds on the PZR surge line hot leg nozzle and the decay heat removal hot leg nozzle were properly identified and corrected to preclude repetition and were reviewed by appropriate levels of management. The safety evaluation and subsequent review process for approving the installation of the weld overlay repair on the DHR dissimilar metal weld with the DHR system in-service was thorough, included training of personnel on contingency plans, and emphasized industry operating experience on similar repairs. One licensee identified violation of very low safety significance was identified in section 4OA7.

4OA3 Followup of Events and Notices of Enforcement Discretion

.1 Licensee Event Report (LER)

(Closed) LER 05000302/2008-001-00: Software Change Causes Inoperability of Redundant Core Subcooling Monitors for Longer than TS Allowable

The LER documented that due to a software change, both channels of the core subcooling monitors were inoperable for longer than the Improved Technical Specification allowed outage time. The inspectors reviewed the LER and NCR 263310 documenting the event. The inspectors checked the accuracy and completeness of the LER and the appropriateness of the licensee's corrective actions. Since this violation of Technical Specification 3.3.17, function 21, was identified during a surveillance test; was of very low safety significance; and was entered into the corrective action program, the finding was treated as a licensee identified violation as documented in Section 4OA7.

.2 Operator performance during non-routine event

a. Inspection Scope

For the six 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 9, Rapid power reduction to 65 percent in accordance with AP-510, Rapid Power Reduction

- January 26/27, Reactor power decrease to 63 percent and subsequent increase to 100 percent in accordance with OP-204, Power Operation

- January 31, Increase in power to new 100 percent RTP in accordance with OP-204-02, Power Level Upgrade from 2568 MWt to 2609 MWt

- February 28, Electrical grid perturbation due to loss of substation in South Florida

- March 1, Reactor shutdown in accordance with OP-209A, Plant Shut Down And Cool Down

- March 20 Reactor startup and power ascension to Mode 1 in accordance with OP-202A, Refueling Outage Plant Heatup and Startup

b. Findings

No findings of significance were identified.

4OA5 Other Activities

NRC Temporary Instruction (TI) 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs)

a. Inspection Scope

From March 10 to 19, 2008, the inspectors reviewed the licensee's activities related to the inspection and mitigation of dissimilar metal butt welds in the Reactor Coolant System (RCS) to ensure that the licensee activities were consistent with the industry requirements established in the Materials and Reliability Program (MRP) document MRP-139, Primary System Piping Butt Weld Inspection and Evaluation Guidelines, July 2005. The inspectors' activities took place during an unplanned outage and covered the following: a) implementation of baseline inspections for the Pressurizer (PZR) surge line-to-hot leg DMBW and the decay heat removal suction line-to-hot leg DMBW, b) documentation and direct observation of the weld overlay process on the decay heat removal suction line-to-hot leg DMBW, and c) documentation and direct observation of the volumetric examination of the decay heat removal suction line-to-hot leg DMBW after completion of the full structural weld overlay (FSWOL). The inspectors only implemented portions of TI-172 that corresponded to the available activities during the aforementioned outage. The remaining inspection activities will be completed prior to the end of 2008.

b. Findings and Observations

No findings of significance were identified

Licensees' Implementation of MRP-139 Baseline Inspections

1) Have baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance?

The inspectors' review of MRP-139 baseline inspections was limited to the following areas. The NRC's complete review of baseline examinations on the remaining welds in the scope of the MRP-139 program is scheduled to be performed by the end of 2008.

- Pressurizer – The licensee installed weld overlays on all PZR DMBWs within the scope of the MRP-139 program during the fall 2007 refueling outage. There were no ultrasonic (UT) examinations performed prior to the installation of weld overlays.

- PZR surge line-to-hot leg DMBW – The licensee installed a weld overlay during the fall 2007 refueling outage and subsequently removed it due to indication of lack of fusion identified during ultrasonic examinations. After the weld overlay removal, the licensee performed a UT examination in accordance with MRP-139 requirements. This UT examination served as a baseline examination until the installation of a weld overlay, which is planned during the next refueling outage.

- Decay heat removal system suction line-to-hot leg DMBW – The licensee performed a UT examination on the suction line-to-hot leg DMBW during the fall 2007 refueling outage. However, the examination coverage did not meet the coverage requirements of MRP-139. After additional surface preparation, the licensee performed a qualified phased array UT examination during a forced outage in March 2008. The phased array examination identified an unacceptable circumferential indication 15 inches long and 65 percent through wall. For information purposes, the licensee confirmed this indication by conventional qualified UT techniques which also identified the indication and, as added assurance, the results of the phased array examinations were also reviewed by the Electric Power Research Institute (EPRI). The licensee repaired the weld by installing a FSWOL. The UT performed on the weld overlay is presented below under the "Volumetric Examinations" section.

2) Is the licensee planning to take any deviations from MRP-139 requirements?

No, the licensee has not submitted any requests for deviation from MRP-139 requirements.

<u>Volumetric Examinations</u>

1) For each examination inspected, was the activity performed in accordance with the examination guidelines in MRP-139, Section 5.1 for unmitigated welds or mechanical stress improved welds and consistent with NRC staff relief request authorization for weld overlaid welds?

<u>Pressurizer Surge Line-to-Hot Leg DMBW (Baseline Inspection)</u>

Yes, the volumetric examination on the pressurizer surge line-to-hot leg line was performed in accordance with qualified procedures for UT examination in accordance with MRP-139 requirements. Procedures were qualified in accordance with ASME Section XI, Appendix VIII, as implemented through the EPRI Performance Demonstration Initiative (PDI) Program. Subsequent to the removal of the FSWOL, and prior to the conventional UT examination conducted during the fall 2007 refueling outage, the licensee verified surface contour to ensure it permitted volumetric examination as well as the surface finish to ensure it was 250 μ-inches RMS or better. Additional qualified phased array UT examinations were conducted during the forced outage in March 2008. Prior to this phased array examination, the licensee performed additional surface preparation to ensure the surface finish provided optimal phased array results.

<u>Decay Heat Removal Suction Line-to-Hot Leg DMBW After Mitigation by FSWOL</u>

Yes, the volumetric examination on the decay heat removal suction line FSWOL was performed in accordance with a qualified procedure for UT examination in accordance with MRP-139 requirements and consistent with the relief request submitted for NRC approval (Relief Request 08-001-RR, Revision 1, ADAMS Accession Number ML080740282).

The procedure was qualified in accordance with ASME Section XI, Appendix VIII, as implemented through the EPRI PDI Program. Prior to the examination, the licensee verified the FSWOL surface contour to ensure it permitted volumetric examination as well as the surface finish to ensure it was 250 μ-inches RMS or better. The licensee conducted the examination 48 hours after the third weld layer was completed. The licensee utilized UT phase array technology to perform the examination using 0° through 83° examination angles for the axial direction and 0° through 69° for the circumferential direction. The UT examiners scanned the FSWOL to the maximum extent practicable in two axial and two circumferential directions. The licensee was able to obtain adequate coverage in the WOL material volume for the detection of weld fabrication flaws. This part of the examination resulted in 99 percent coverage in the circumferential direction and 100 percent coverage in the axial direction. In addition, the licensee obtained 100 percent of the pre-service (PSI) required coverage in the circumferential and axial directions, as established in the relief request.

2) For each examination inspected, was the activity performed by qualified personnel?

Yes, the personnel involved in the UT examinations of the PZR surge line-to-hot leg nozzle and the decay heat removal suction line FSWOL were qualified in accordance with MRP-139 requirements and the licensee's relief request. The examiners were qualified Level II in the UT method as required by the vendor's UT phase array procedure and in accordance with the vendor's written practice for non-destructive examination (NDE) personnel. The UT examiners were also PDI qualified for the specific UT procedure they implemented. In addition to the Level II and PDI qualification, the UT procedure required additional training on the operation of the UT phase array instrument. The final examination report, including calibration data sheets, was reviewed by a vendor's Level III inspector in the UT method and a licensee's Level III in the UT method.

3) For each examination inspected, was the activity performed such that deficiencies were identified, dispositioned, and resolved?

Pressurizer Surge Line-to-Hot Leg DMBW (Baseline Inspection)

Yes, the inspectors reviewed documentation to verify deficiencies were identified, dispositioned, and resolved. The inspectors reviewed deficiency reports associated with the unacceptable indications in the weld overlay that was removed, and the weld re-examined by UT. The cause for the unacceptable (lack of fusion) weld overlay is still being evaluated. However, preliminary evaluation identified the cause to be welding on the downward motion instead of on the upward motion. There were no deficiency reports reviewed for the UT performed.

Decay Heat Removal Suction Line-to-Hot Leg DMBW after Mitigation by FSWOL

Yes, the inspectors reviewed documentation and directly observed field work to verify that deficiencies was identified, dispositioned, and resolved. The inspectors' review of documents for this volumetric examination covered the following: UT examination procedure and documentation to support its qualification for the intended use, assessment of personnel training and qualification, equipment certification and calibration records, and two corrective action documents which were generated to address minor deficiencies related to the UT couplant gel and the UT transducer wedges. Furthermore, the inspectors directly observed the calibration of the UT phase array equipment prior to the examination, the actual examination of the FSWOL, and the calibration check at the end of the examination to verify that the examination was performed in accordance with the relief request and the applicable procedures. The inspectors also discussed the examination requirements with the licensee and vendor personnel. Based on the aforementioned inspection activities, the inspectors considered that the examination was conducted in a manner such that deficiencies would be identified, dispositioned, and resolved.

<u>Weld Overlays</u>

1) For each weld overlay inspected, was the activity performed in accordance with ASME Code welding requirements and consistent with NRC staff requests authorizations? Has the licensee submitted a relief request and obtained NRR staff authorizations to install weld overlays?

Yes, the licensee installed the decay heat removal FSWOL in accordance with NRC approved Relief Request 08-001-RR, Revision 1, ADAMS Accession Number ML080740282. The inspectors reviewed the welding package and identified that the FSWOL was classified as a repair weld overlay, the first layer of weld metal deposited was not credited toward the required thickness, the weld overlay was not being installed over an existing weld overlay and the tempered bead method was incorporated in accordance with Attachment 2 of the relief request.

The inspectors reviewed the welding procedures, applicable procedure qualification records, welder performance qualification test records, and the in-process welding process control sheets for compliance to ASME Section IX requirements.

2) For each weld overlay inspected, was the activity performed by qualified personnel?

Yes, welding activities and personnel were qualified in accordance with the requirements identified in ASME Code Section IX. The inspectors reviewed the welding procedures, applicable procedure qualification records, welder performance qualification test records, and the in-process welding process control sheets for compliance with the relief request and ASME Section IX requirements.

3) For each weld overlay inspected, was the activity performed such that deficiencies were identified, dispositioned, and resolved?

Yes, the inspectors reviewed documentation and directly observed field work to verify that deficiencies were identified, dispositioned, and resolved. The inspectors' reviewed documentation of two corrective action documents which were generated to address minor deficiencies related to acceptance of welding material certified mill test reports and control of weld wire. The inspectors also discussed the requirements with the licensee and vendor personnel. Based on the aforementioned inspection activities, the inspectors considered that the installation of the FSWOL was conducted in a manner such that deficiencies would be identified, dispositioned, and resolved.

Mechanical Stress Improvement (Not Applicable)

The licensee has not implemented Mechanical Stress Improvement as a mitigation method for DMBWs.

In-service Inspection Program

1) Has licensee prepared an MRP-139 in-service inspection program?

No, the licensee does not have an MRP-139 in-service inspection program document. The licensee's MRP -139 inspection program consists of the documents listed below, which were previously prepared documents, and the inclusion of MRP-139 requirements as augmented inspections in the ASME Section XI In-service Inspection (ISI) Program. The inspectors reviewed the following documents and held discussions with licensee representatives.

- Progress Energy Nuclear Generation Group Alloy 600 Strategic Plan, Revision 0
- Nuclear Generation Group Standard Procedure ADM-NGGC-0112, Reactor Coolant System Material Integrity management Program, Revision 0
- Progress Energy Letter dated January 29, 2007, Inspection and Mitigation of Alloy 600/82/182 Pressurizer Butt Welds
- NRC Letter dated March 27, 2007, Confirmatory Action Letter Crystal River Unit 3
- Crystal River Action Request 00223348, CR3 PZR Weld and RCS Leakage Monitoring
- Progress Energy letter dated January 23, 2008, CR Summary of Ultrasonic Examination results of Structural Weld Overlays
- Action Request 00170903, Action Plan to Implement MRP-139
- ISI drawings and list of welds in the Scope of MRP-139.

2) Are welds appropriately categorized?

This was not reviewed during this inspection and is scheduled to be reviewed prior to the end of 2008.

3) Are inspection frequencies consistent with the requirements of MRP-139?

This was not reviewed during this inspection and is scheduled to be reviewed prior to the end of 2008.

4) What is the licensees' basis for categorizing welds as H or I and plans for addressing potential PWSCC?

This was not reviewed during this inspection and is scheduled to be reviewed prior to the end of 2008.

5) What deviations has the licensee incorporated and what approval process was used?

This was not reviewed during this inspection and is scheduled to be reviewed prior to the end of 2008.

4OA6 Exit

.1 Exit Meeting Summary

On April 7, 2008, 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 8, 2008, the NRC's Chief of Reactor Projects Branch 3, Region II Public Affairs Officer, and the Resident staff assigned to the Crystal River Nuclear Plant met with Progress Energy – Florida Power Corporation (FPC) to discuss the NRC's Reactor Oversight Process (ROP) and the Crystal River annual assessment of safety performance for the period of January 1, 2007 – December 31, 2007. The major topics addressed were: the NRC's assessment program, the results of the Crystal River Unit 3 assessment, and future NRC inspection activities. Attendees included FPC management, FPC site staff, three members of the Citrus County Sheriff's Department, three members of the public, and one newspaper reporter.

This meeting was open to the public. The NRC's presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML081010395 The licensee's handout presented at the meeting is also available from the NRC's document system (ADAMS) as accession number ML081010392 s accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

4OA7 Licensee Identified Violations

The following issues of very low safety significance (Green) were identified by the licensee and were violations of NRC requirements. These issues met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations.

- Improved Technical Specification (ITS) 3.3.17, Post Accident monitoring (PAM) Instrumentation, requires, in part, that both channels of the function, Degrees of Subcooling, shall be operable in MODES 1, 2, and 3. ITS 3.3.17, Condition C, states that with one or more functions with two required channels inoperable, restore one channel to operable within 7 days. Contrary to the above, on January 25, 2008, during surveillance testing, the licensee determined that both channels of the function, Degrees of Subcooling, had been inoperable since a software change on August 13, 2007. The inspectors determined that the failure to comply with ITS was of very low safety significance since the Degrees of Subcooling function would have remained available during the most limiting accident conditions (incore temperatures less than 1250°F). The software change only affected the Degrees of Subcooling function above incore temperatures of 1250°F. This issue is documented in the licensee's corrective action program as NCR 263310.

- 10 CFR 55.33 (b) states that if an applicant's general medical condition does not meet the minimum standards under §55.33(a)(1), the Commission may approve the application and include conditions to accommodate the medical defect. Contrary to the above, one licensed operator stood watch in a TS position as Operator at the Controls on 19 different occasions between July 9 and August 30, 2007, without complying with a newly issued license condition to take prescribed medication while performing licensed duties. Because of the extenuating circumstances that resulted in the operator not being properly informed of the new restriction, compliance with his license was reasonably beyond his control. This finding is of very low safety significance because other licensed operators were available to man the controls and the restricted operator was under supervision at all times. This event is documented in the licensee's corrective action program as NCR 244615.

- 10 CFR 50, Appendix B, Criterion V, "Instructions Procedures and Drawings", requires that activities affecting quality shall be prescribed by documented instructions, procedures or drawings of a type appropriate to the circumstances and these instructions, procedures and drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that the important activities have been satisfactorily accomplished. Contrary to these requirements, there were no written instructions to inform personnel implementing dissimilar metal weld inspections on what to do if the coverage of greater than 90 percent required by MRP-139 is not obtained. This resulted in the plant returning to power from RFO 15 without the ultrasonic examinations being conducted in accordance with the requirements of MRP-139. This finding is determined to be of very low safety significance because the deficiency was identified and examinations that met the requirements of MRP-139 were performed during a forced outage prior to the due date in MRP-139. The licensee entered the finding into their corrective action program as NCR 270077.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
M. Annacone, Plant General Manager
W. Brewer, Manager, Maintenance
S. Cahill, Manager, Engineering
P. Dixon, Manager, Nuclear Assessment
B. Foster, Acting Manager, Engineering
J. Franke, Director of Site Operations
R. Hons, Manager Training
J. Holt, Manager, Operations
D. Herrin, Acting, Supervisor, Licensing
M. Rigsby, Superintendent, Radiation Protection
J. Stephenson, Supervisor, Emergency Preparedness
I. Wilson, Manager Outage and Scheduling
D. Young, Vice President, Crystal River Nuclear Plant

NRC personnel:
M. Sykes, Chief, Branch 3, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000302/2008002-01	NCV	Inoperable Fire Penetration Seal (Section 1R05.1)
05000302/2008002-02	FIN	Failure to Implement Adequate Equipment Protection Resulted in a Plant Transient (Section 4OA2.2)

Closed

0500302/2008-001-00	LER	Software Change Causes Inoperability of Redundant Core Subcooling monitors for Longer than TS Allowable (Section 4OA3.1)

Discussed

NRC TI 2515/172		Reactor Coolant System Dissimilar Metal Butt Welds (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
SP-804, Surveillance of Plant Fire Brigade Equipment
HPP-502, Respiratory Equipment Inspection and Maintenance

Section 1R11: Licensed Operator Requalification Program

Procedures:
TAP-001, Training Conduct and Expectations, Rev 0
TAP-100, Analysis Phase, Rev 4
TAP-200, Design Phase, Rev 4
TAP-300, Development Phase, Rev 8
TAP-403, Conduct of Written Examinations, Rev 10
TAP-409, Conduct of Simulator Training and Evaluation, Rev 22
TAP-410, NRC License Examination Security Program, Rev 11
TAP-425, NRC Initial Licensed Operator Exam Development Guideline, Rev 3
TAP-426, Licensed Operator Continuing Training Biennial Written Examinations, Rev 2
TAP-427, Licensed Operator Continuing Training Annual Operational Examinations, Rev 1
TAP-500, Evaluation Phase, Rev 4
TPP-200, Licensed Operator Continuing Training Program, Rev 7
AR-401, PSA F Annunciator Response, Rev 37
OP-409, Plant Ventilation System, Rev 72
TRN-NGGC-0002, Performance Review and Remedial Training, Rev 0
TRN-NGGC-0008, Conduct of On-the-Job Training and Task Performance Evaluation, Rev 4
TRN-NGGC-0009, Training Exemption Requirements, Rev 0

Drawings
302-756, Excerpt of SH-001
208-078, CH-09

Job Performance Measures
In-plant JPM 086, Cross connect EDG fuel oil supplies
Administrative JPM 423, Perform a QPTR Calculation
Simulator JPM 433: Perform immediate actions of EOP-2, Vital System Status Verification

Simulator Conformity Documents Reviewed
TAP-412, Simulator Operation, Rev 4
TAP-422, Simulator Maintenance, Rev 3
TAP-428, Simulator Scenario - Based Testing, Rev 0
TPP-206, Simulator Program, Rev 6

<u>CR Simulator Transient Tests</u>
PTT6 (2006-2007), main turbine trip
PTT8 (2006-2007), Large Break LOCA with LOOP
Cycle 15 and 16 Low power physics test results

<u>Simulator Exercise Guide</u>
Simulator Examination Scenario 43, Rev 0
Simulator Examination Scenario 48, Rev 0
Simulator Examination Scenario 23, Rev 5 (Practice Scenario)

<u>Records:</u>
Badge access transaction reports for reactivation of licenses (4)
Licensed Operator medical records (15)
2 years of feedback summaries
Immediate course feedback summary (Form 500.2)
Remedial training records (6)
Quarterly reports
Examination pass/fail records for 2007
Open and closed simulator service request

<u>Written Examinations Reviewed:</u>
Five written examinations that were administered for the 2007 biennial requalification Examinations, Crews A-E, RO and SRO.

<u>Other:</u>
Enhanced design basis document for the chilled water system, issue date 07/03/91

<u>Section 1R12: Maintenance Effectiveness</u>

<u>Nuclear Condition Reports</u>
NCR 254518, A EGDG Oil Leak
NCR 262398, C EGDG Inlet Piping Coupling Leak

<u>Miscellaneous</u>
Equipment Performance Priority List dated 2/12/08

<u>Section 1R20: Refueling and Other Outage Activities</u>

<u>Procedures</u>
AI-504, Guidelines For Cold Shutdown and Refueling
OP-103B, Plant Operating Curves
OP-103H, Reactor Coolant System and Spent Fuel Pool Decay Heat Tables and Figures
OP-202A, Refueling Outage Plant Heatup and Startup
SP-324, Containment Inspection
WCP-102, Outage Risk Assessment
WCP-103, Station Readiness for Reduced Inventory, Mode 4/3 Entry and Mode 2/1 Entry

Sections 4OA2.4: Problem Identification and Resolution and 40A5: Other Activities

Procedures
SI-UT-130, Procedure for Phased Array Ultrasonic Examination OF Dissimilar Metal Welds, Rev. 0
SI-NDE-08, Qualification and Certification of NDE Personnel for Nuclear Applications, Rev 1
SI-UT-126, Procedure for Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds, Revision 3
PDI Protocol SI-UT-126, Table 1, Revision 0
Performance Demonstration Qualification Sheet No.: 535, Procedure SI-UT-126, Rev 3, Addenda 0
AREVA 54-ISI-829-08, Dated 8/07/07, Manual Ultrasonic Examination of Dissimilar Metal Welds
AREVA 54-ISI-600-02, Dated 7/16/06, Manual Ultrasonic Examination of Overlays for Thickness and Profile Measurements
AREVA 54-ISI-838-09, Dated 8/20/07, Manual Ultrasonic Examination of Weld Overlay Similar and Dissimilar Metal Welds
PDI Generic Procedure PDI-UT-8

Corrective Action Documents
NCR 00268933, Flaws identified During UT Exam of DH Nozzle Weld
NCR 00261079, R15 DH Nozzle Inspection Deficiency Decision Making Issues
NCR 00261078, DH UT Inspection did not Meet ASME or MRP-139 Requirements
Action Request 00170903, Action Plan to Implement MRP-139,
Action Request, "CR3 PZR Weld and RCS Leakage Monitoring"
NCR 00270074, Procedure Enhancement for MRP-139
NCR 00270077, Inspection Reports not documented in accordance with AREVA NDE Procedure
NCR 00271099, Phased Array Wedges not Field Verified
NCR 00271098, Coupling Gel Used for UT Inspection was Incorrect

Other Records
Progress Energy Nuclear Generation Group "Alloy 600 Strategic Plan, Rev. 0,"
Nuclear Generation Group Standard Procedure ADM-NGGC-0112, Rev. 0, "Reactor Coolant System Material Integrity management Program,"
Progress Energy Letter dated September 13, 2007, "Crystal River Unit 3 – Relief Request #07-003-RR, Rev. 1, and Response to Request for Additional Information
Progress Energy Letter dated March 7, 2007, "Crystal River Unit 3 Relief Request #08-001-RR, Rev. 0.
Progress Energy Letter dated January 29, 2007, "Inspection and Mitigation of Alloy 600/82/182 Pressurizer Butt Welds,"
NRC Letter dated March 27, 2007, "Confirmatory Action Letter Crystal River Unit 3,"
Progress Energy letter dated January 23, 2008, "CR Summary of Ultrasonic Examination results of Structural Weld Overlays
ISI drawings and list of welds in the Scope of MRP-139
Structural Integrity Calculation File 0800025.303 – Evaluation of Allowable Flaw Size in Decay Heat Nozzle to Pipe Weld during Weld Overlay Application
Certification of Chemical Analysis for Ultrasonic Couplant Batch No: 02220
Examination Data Sheet No.: CR3-HL Decay Heat-WOL-08-01
Calibration Certificate of Conformity No. 155 for Omniscan Instrument OMNI-1983
Calibration Certificate No. 2189, Equipment OMNI-MX, Serial OMNI-1983

Certification Records for Ultrasonic Transducer Models 115-000-405 (Serial 01MLLN) and 115-000-404 (Serial 01ML7V)
Qualification Records for Ultrasonic Phase Array Examiners
Traveler 105500-TR-006, Decay Heat Nozzle, Crystal River Construction Drawing, Rev. 3

Section 40A3: Followup of Events and Notices of Enforcement Discretion

OP-209A, Plant Shut Down And Cool Down
OP-103B, Plant Operating Curves