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

January 25, 2010

Mr. Jon A. Franke, Vice President
Crystal River Nuclear Plant (NA1B)
Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2009005

Dear Mr. Franke:

On December 31, 2009, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Crystal River Unit 3. The enclosed inspection report documents the inspection findings, which were discussed on January 11, 2010, 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, two self-revealing findings of very low safety significance (Green) were identified. The findings were both determined to involve a violation of NRC requirements. However, because of the very low safety significance of the issues and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy. Also, one licensee identified violation which was of very low safety significance is listed in Section 4OA7 of the report. If you contest the non-cited violations, 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 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, 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 05000302/2009005
w/Attachment: Supplemental Information

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
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Enclosure: Inspection Report 05000302/2009005
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Letter to Jon A. Franke from Marvin D. Sykes dated January 25, 2010

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2009005

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2009005
Licensee:	Progress Energy (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	Crystal River, FL
Dates:	October 1, 2009 – December 31, 2009
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector R. Chou, Reactor Inspector (Section 4OA5) A. Alen, NSPDP Intern B. Adkins, Construction Project Inspector P. Higgins, Project Engineer M. Barillas, Resident Inspector, Turkey Point R. Hamilton, Senior Health Physicist (Sections 2OS2, 4OA5) G. Kuzo, Senior Health Physicist (Sections 2PS2, 4OA1, 4OA5) W. Loo, Senior Health Physicist (Sections 2OS1, 4OA5) L. Mahlahla, Health Physicist (Sections 2OS2, 4OA1, 4OA5) S. Ninh, Senior Project Engineer E. Michel, Senior Reactor Inspector (Section 4OA5) B. Collins, Reactor Inspector (Sections 1R08 and 4OA5) R. Carrion, Senior Reactor Inspector (Section 4OA5)
Approved by:	M. Sykes, Chief, Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2009005; 10/01/2009 -12/31/2009; Crystal River Unit 3; Followup of Events and Notices of Enforcement Discretion

The report covered a three month period of inspection by resident inspectors, two regional project engineers, four regional reactor inspector, one regional construction project inspector, one regional intern and four regional health physicists. Two self-revealing NCVs were identified. The significance of most findings is identified by the 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

Cornerstone: Mitigating Systems

Green. A self-revealing Non-Cited Violation (NCV) of Improved Technical Specification (ITS) 5.6.1.1.a was identified for the failure to follow a plant procedure which resulted in a loss of a 480 volt engineered safeguards motor control center (ES MCC)-3B1. Concurrent with pre-existing conditions, the high pressure injection (HPI) system was declared inoperable and ITS 3.0.3 was entered for a period of one hour and 24 minutes. The licensee entered this issue into the corrective action program as nuclear condition report (NCR) 333515.

The finding was more than minor since it affected the equipment availability attribute of the mitigating system cornerstone and resulted in ITS 3.0.3 entry for the HPI system being inoperable. The finding was evaluated against NRC Phase 1 Significance Determination Process (SDP) and Phase 2 SDP was required due to a loss safety function of the HPI system. A Regional Senior Reactor Analyst performed a Phase 3 SDP evaluation and concluded this finding was of very low safety significance (Green). The major assumptions of the evaluation were that the HPI function was out of service for exposure period (1 .5 hours) and there would be no recovery of the de-energized motor control center. The dominant accident sequence involved a support system failure of the Emergency Feedwater (EF) Indication and Control System rendering Main Feedwater and automatic control of EF unavailable, operators were unable to manually control EF flow causing its failure and with the HPI function lost due to the performance deficiency, core damage ensued. The inspectors determined the cause of the finding is related to the cross-cutting area of Human performance with a work practices aspect H.4 (c)). Specifically, work scope changes involving safety-related equipment did not receive the appropriate level management oversight resulted in a plant procedural violation. (Section 4OA3.2)

Cornerstone: Initiating Events

Green. A self-revealing NCV of Improved Technical Specification (ITS) 5.6.1.1.a was identified for the failure to follow the provisions of preventative maintenance procedure PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train. Failure to follow

PM-126 caused the failure of the Group 7 control rod programmer during maintenance and resulted in the unexpected insertion of the Group 7 control rods fully into the core. This unexpected insertion of these control rods into the core caused control room operations personnel to manually trip the reactor from 100 percent power. The licensee entered this issue into the corrective action program as NCR 351705.

This finding was determined to be more than minor because it was associated with the initiating events cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available. The cause of this finding was directly related to the cross-cutting area of Human Performance with a work practices aspect (H.4 (b)). Specifically, the workers failed to follow the preventative maintenance procedure. (Section 4OA3.3)

B. Licensee Identified Violations

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violation and corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

Crystal River unit 3 began the inspection period shutdown in Mode 5 (< 200°F). On October 9 the movement of all the reactor fuel to the spent fuel pool was completed. The unit remained in this condition for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors evaluated the licensee's readiness for mitigating cold weather to assure that vital systems and components were protected from freezing in accordance with the licensee's administrative instruction AI-513, Seasonal Weather Preparations, Section 4.1, Cold Weather Preparations. The inspectors walked down portions of the systems/areas listed below to check for any unidentified susceptibilities. Operability of heat trace circuits and set points of temperature controls was verified. Nuclear condition reports (NCRs) were reviewed to check that the licensee was identifying and correcting cold weather protection issues.

- Borated water storage tank heaters
- Intermediate building cooling system instrument lines heat tracing
- Emergency feed water pump EFP-3 battery room temperature controls
- Emergency feed water pump EFP-3 heat tracing

There were no sustained periods of freezing weather during the inspection period.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

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

- A train decay heat removal (DHR) system which was providing cooling to the spent fuel pool using operating procedure OP-406, Spent Fuel Cooling System, while the nuclear service water (SW) system was out of service for planned maintenance
- B train spent fuel cooling, raw water (RW), and SW systems using OP-408, Nuclear Services Cooling System and OP-406 with the A train electrical systems out of service for planned maintenance

b. Findings

No findings of significance were identified.

1R05 Fire Protection

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 Final Safety Analysis Report (FSAR) Section 9.8, Plant Fire Protection Program, were met. Documents reviewed are listed in the attachment. The inspectors toured the following four areas important to reactor safety.

- A and B train 4160 volt Engineering Safeguards (ES) switch gear rooms
- Spent fuel pool floor
- Emergency feed pump EFP-1 and EFP-2 area
- Spent fuel pump and heat exchanger area

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

Internal Flood Protection

a. Inspection Scope

The Inspectors reviewed the Crystal River Unit 3, FSAR, Chapter 2.4.2.4, Facilities Required for Flood Protection, surveillance procedure SP-407, Fire and Flood Barrier Penetration Seals Inspection, and the Crystal River Unit 3 Design Basis Documents that depicted protection for areas containing safety-related equipment to identify areas that may be affected by internal flooding. A walk down of the auxiliary building A train DHR vault was conducted to ensure that flood protection measures were in accordance with design specifications. Specific plant attributes that were checked included structural integrity, sealing of penetrations, and operability of sump systems.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

From October 12 – 16, 2009, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator tubes, emergency feedwater systems, risk-significant piping and components and containment systems. Documents reviewed are listed in the attachment.

The inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4 and 1R08.5 below constituted one inservice inspection sample as defined in Inspection Procedure 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors evaluated the following non-destructive examinations mandated by the ASME Code Section XI to verify compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Ultrasonic Testing (UT) of MS-6A pipe-to-elbow weld, ASME Class 2, Main Steam system, 24" diameter – Direct Observation
- Magnetic Particle Testing (MT) of MS-6A pipe-to-elbow weld, ASME Class 2, Main Steam system, 24" diameter – Direct Observation

During non-destructive surface and volumetric examinations performed since the previous refueling outage, the licensee did not identify any recordable indications that were accepted for continued service. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed documentation for the following pressure boundary welds completed for risk-significant systems during the outage to evaluate if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the Construction Code. In addition, the inspectors reviewed the welding procedure specifications, welder qualifications, welding material certifications and supporting weld procedure qualification records to evaluate if the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Work Order 00754691, replacement of valve MUV-401
- Work Order 00998182, replacement of valve MUV-522

b. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 3 vessel head, a bare metal visual examination was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors reviewed records of the visual examination conducted on the Unit 3 reactor vessel head to evaluate if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, the inspectors reviewed the following documentation and/or observed the following activities:

- Evaluated if the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with the licensee procedures.
- Evaluated if the licensee's criteria for visual examination quality and instructions for resolving interference and masking issues were adequate.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage for Unit 3. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors performed an independent walkdown of portions of the containment building, which recently received a licensee boric acid walkdown and evaluated if the licensee's BACC visual examinations emphasized locations where boric acid leaks could cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of reactor coolant system components with boric acid deposits to evaluate if degraded components were documented in the corrective action system. The inspectors also evaluated the corrective actions for any degraded reactor coolant system components against the component Construction Code.

- NCR 269121, Boric Acid on RCV-177
- NCR 308537, Boric Acid on WDV-25
- NCR 245897, Boric Acid on MU-101-FC

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to evaluate if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- NCR 270753, RCP-1B Standpipe Boron Leak
- NCR 293115, RCV-164 Boron Leak
- NCR 350362, Boron Leaks on DHV-211 and DHV-32

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

The Crystal River Unit 3 Steam Generators were replaced during this inspection period, and there were no inspection activities associated with the generators which were replaced. The activities associated with the new generators were performed under IP 50001 and are described in Section 4OA5 of this report.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if;

- the licensee had established an appropriate threshold for identifying ISI/SG related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

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 an assessment of the licensee’s practices associated with 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, the inspectors verified that reliability and unavailability were properly monitored and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors conducted this inspection for the following one equipment issue:

- NCR 358949, Makeup valve MUV-36 stem/disc separation

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 were adequate. 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 administrative procedure, AI-504, Guidelines for Cold Shutdown and Refueling, was verified for the following one work week assessment.

- Work Week 09WW39, Risk condition yellow with the reactor coolant system drained to the reactor vessel flange level

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following four 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 corrective action procedure CAP-NGGC-200, Corrective Action Program.

- NCR 360381, Corrosion on the containment liner
- NCR 362109, Cracked/Broken hold down springs on three fuel assemblies
- NCR 364588, Unusual crud indications on CR3C16 Mark-B HTP fuel
- NCR 359930, Leakage past containment isolation valve FWV-46

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed one 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 engineering procedure EGR NGGC-00005, Engineering Change, to assess if the modification was properly developed and implemented.

- Engineering change EC 74743, Temporary Power for spent fuel cooling pump SFP-1A

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 four post-maintenance tests reviewed are listed below:

- SP-354A, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1A (fast start and full load Test portion only), after performing maintenance on the diesel engine per work order (WO) 1552552
- SP-344A, RWP-2A, SWP-1A And Valve Surveillance (RWP-2A portion only), after replacing the RWP-2A pump motor per WO 1581469
- PM-105, Inspection, Testing And Maintenance Of Electrical Motors, after performing maintenance on containment air handling fans AHF-1A and AHF-1B, per WO 1171573 and 1171574 respectively
- SP-354B, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1B, after performing maintenance per WO 1670978

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

Steam Generator Replacement Refueling Outage (RFO16)

a. Inspection Scope

On September 26 the unit was shutdown for a planned steam generator replacement refueling outage. NRC inspection activities for the first five days of the outage are documented in NRC integrated inspection report 05000302/2009004. During the quarter, the inspectors observed and monitored licensee controls over the refueling outage activities listed below. Additional inspection results for RFO16 will be documented in next quarter's NRC integrated inspection report 05000302/2010002. 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
- Refueling activities
- Reactor mode changes

b. Findings

No findings of significance were identified. During the creation of temporary opening in the reactor containment building to support steam generator replacement, the licensee discovered an internal crack in the concrete containment. The circumstances associated with the crack in the concrete containment wall are being assessed by an NRC special inspection team. The results of that inspection will be documented in NRC special inspection report 05000302/2009007.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed five 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- 344B, RWP-2B, SWP-1B And Valve Surveillance

Containment Isolation Valve Test:

- SP-179C, Containment Leakage Test – Type C, make-up valves MUV-49 and MUV-567
- SP-179C, Containment Leakage Test – Type C, containment monitoring system penetration 356

Surveillance Test:

- SP-435, Valve Testing During Cold Shutdown
- SP-523, Station Batteries Service Test (batteries 3B1 and 3B2 only)

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control To Radiologically Significant Areas

The inspectors evaluated licensee guidance and its implementation for controlling worker access to radiologically significant areas and for monitoring jobs in-progress. The inspectors directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection (RP) requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation area and high radiation (HRA) locations established within radiologically controlled area (RCA) locations of the auxiliary building, reactor building, and radioactive waste (radwaste) processing and storage facilities. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were observed and evaluated for selected Locked High Radiation Area (LHRA) locations. Changes to procedural guidance for LHRA and Very High Radiation Area controls were discussed in detail with select health physics (HP) supervisors. Physical controls for storage of irradiated material within the spent fuel pool (SFP) were observed and discussed with cognizant licensee representatives. In addition, licensee controls for areas where dose rates could change significantly as a result of Refueling Outage 16 (RFO 16) tasks or radwaste processing activities were reviewed and discussed.

The inspectors attended selected pre-job briefings and remotely observed selected activities associated with the steam generator (SG) replacement activities during RFO 16. The observations afforded the inspectors opportunity to assess various aspects of the RP program including communications between HP and various workgroups, internal communications, RWP controls, contamination control, surveys, radiation worker adherence to RWP and other HP guidance, HPT proficiency in providing job coverage and supervisory willingness to intervene when conditions deviated from expected.

The inspectors observed various other work activities including radiography in the radiography testing bunker, "A" SG upper manway diaphragm removal, and shielding of various pipes and scaffold building in the reactor building. The inspectors evaluated electronic dosimeter (ED) alarm setpoints with area radiation survey results and discussed ED alarm response actions with radiation workers, HPTs, and RP supervisors.

The inspectors evaluated the effectiveness of radiation exposure controls and monitoring, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Licensee evaluations of skin dose resulting from discrete radioactive particle or dispersed skin contamination events were reviewed and assessed. For HRA tasks involving significant dose rate gradients, the inspectors evaluated procedural guidance for the use and placement of whole body and extremity dosimetry to monitor worker exposure.

RP activities were evaluated against the requirements of Final Safety Analysis Report (FSAR) Chapter 11; Improved Technical Specifications (ITS) Section 5.0; 10 Code of Federal Regulations (CFR) Part 20; and approved licensee procedures. Records reviewed are listed in Sections 2OS1, 2OS2, 2PS2, 4OA1, and 4OA5 of the attachment.

Problem Identification and Resolution: Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. This included the review of a licensee self-assessment and selected Nuclear Condition Report (NCR) documents related to radworker and HPT performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Revision (Rev.) 28. Licensee CAP documents reviewed are listed in Sections 2OS1 and 4OA1 of the attachment.

The inspectors completed 21 of the specified line-item samples detailed in Inspection Procedure (IP) 71121.01. In addition, the inspectors evaluated radiation protection activities associated with steam generator replacement activities as specified in IP 50001.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope

ALARA. The inspectors reviewed ALARA program guidance and its implementation for select RFO 16 refueling and maintenance tasks. The inspectors evaluated the accuracy of ALARA work planning and dose budgeting, observed implementation of ALARA initiatives and radiation controls for selected jobs in-progress, assessed the effectiveness of source-term reduction efforts, and reviewed historical dose information.

ALARA planning documents and procedural guidance were reviewed and projected dose estimates were compared to actual dose expenditures for the high dose jobs associated with the SG replacement and other refueling outage tasks. Differences between budgeted dose and actual exposure received were discussed with plant ALARA staff. Changes to dose budgets relative to changes in radiation source term and/or job scope were also discussed. The inspectors reviewed select interdepartmental initiatives instituted to reduce exposure to plant personnel during the SG replacement activities. The inspectors questioned the consistent overestimates for work associated with the SG replacement activities. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel. The inspectors also attended an ALARA review committee meeting and observed the interface between plant management and ALARA planning staff.

The inspectors made direct field or closed-circuit-video observations of outage job tasks dose reduction initiatives for tasks such as pipe cutting and welding for replacement of the SG equipment and piping. For the selected tasks, the inspectors evaluated radiation worker and HPT job performance; individual and collective dose expenditure versus percentage of job completion; surveys of the work areas, appropriateness of RWP requirements; and adequacy of implemented engineering controls. The inspectors interviewed radiation workers and job sponsors regarding understanding of dose reduction initiatives and their current and expected accumulated doses at completion of the job tasks.

Implementation and effectiveness of selected program source-term reduction initiatives including source term mitigation and shielding were evaluated. Chemistry program ALARA initiatives including zinc injection, maintenance of elevated pH, use of macroporus resin, and increased cleanup times and their effect on reactor building dose rates were reviewed and discussed in detail. Operations initiatives such as maintaining some pressure in the pressurizer while flushing the steam generator drain lines and reverse flushing the feedwater lines also were discussed. The inspectors reviewed design changes incorporated into the replacement SG to reduce collection of corrosion products, facilitated worker ingress and egress, and improved hangers for shielding installation.

Plant exposure history for the current and previous calendar year and data reported to the NRC pursuant to 10 CFR 20.2206 were reviewed, as were established goals for reducing collective exposure during the current RFO 16 outage. The inspectors reviewed procedural guidance for dosimetry issuance and exposure tracking. The inspectors examined dose records of two declared pregnant workers from CY 2007 to November 2009 to evaluate assignment of gestation dose. ALARA program activities and their implementation were reviewed against 10 CFR Part 20, and approved licensee procedures. In addition, licensee performance was evaluated against guidance contained in Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the attachment.

Problem Identification and Resolution: The inspectors reviewed selected NCRs in the area of exposure control. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Rev. 28. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Documents reviewed for problem identification and resolution are listed in Sections 2OS1 and 2OS2 of the attachment.

The inspectors completed 18 of the specified line-item samples detailed in IP 71121.02. In addition, the inspectors evaluated radiation protection activities associated with steam generator replacement activities as specified in IP 50001.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization: Selected liquid and solid radioactive waste (radwaste) processing system components and radioactive waste storage locations were inspected for material condition and for configuration compliance with the FSAR and Process Control Program (PCP) details. Inspected equipment included the hold-up tanks, waste processing tanks, resin dewatering equipment; radwaste storage locations. The inspectors discussed component function, equipment operability, and changes to radwaste processing systems with licensee staff.

Radioactive waste disposal data for CY 2007 and CY 2008 were reviewed and discussed. Data for select radioactive waste streams sampled for Part 61 radionuclide analysis and characterization from January 1, 2008, through January 31, 2009, were reviewed and discussed in detail with cognizant staff. Licensee guidance and processes for monitoring changes in waste stream isotopic mixtures were discussed with cognizant licensee representatives. For spent primary and secondary resins, spent filters, and dry active waste (DAW), the inspectors reviewed radionuclide determination analyses and evaluated determination of hard-to-detect nuclides. The subject reviews included evaluation of gamma spectroscopy data comparisons of licensee waste stream analyses with vendor laboratory data and verification of appropriate use of scaling factors for waste characterization.

Radwaste processing activities were reviewed for compliance with details in the applicable FSAR Sections and guidance provided in the Branch Technical Position on Waste Classification and Waste Form; details outlined in ITS, PCP, and applicable procedures; and requirements in 10 CFR Part 20, and10 CFR Part 61. Documents reviewed are listed in Section 2PS2 of the attachment.

Transportation: During the week of December 3, 2009, the inspectors directly observed preparation activities for shipment of radioactive material waste of low specific activity and radioactive material shipment of low specific activity material. The inspectors noted package bracing and conveyance placards, observed results of radiation and contamination surveys, evaluated shipping paper documentation for adequacy and completeness, and interviewed the shipping technicians and driver regarding knowledge of Department of Transportation (DOT) regulations. Dose rate and contamination survey data for the shipping packages and conveyance were verified and the results compared to DOT limits. In addition, training provided to individuals involved with the observed shipments was reviewed and evaluated.

In addition, additional shipping records and supporting documents for radioactive material and radioactive waste shipments conducted from January 1, 2007, through November 30, 2009, were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for use of Type B shipping casks were compared to recommended vendor protocols and Certificate of Compliance requirements. In addition, use and preparation of Type A containers for selected shipments were evaluated.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, and 49 CFR Parts 172-178. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the attachment.

Problem Identification and Resolution: The inspectors reviewed and discussed with HP supervision selected NCRs and audits associated with transportation and radioactive waste processing program activities. The inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure CAP-NGGC-0200, Corrective Action Program, Rev. 28.

The inspectors completed the six specified line-item samples detailed in IP 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Reactor Safety

a. Inspection Scope

The inspectors checked the mitigating system performance indicators (MSPI) listed below to verify the accuracy of the PI data reported. Performance indicator data submitted from October 2008 through September 2009 was compared for consistency to data obtained through review of monthly operating reports, nuclear condition reports, and control room logs. The inspections were conducted in accordance with NRC Inspection Procedure 71151, Performance Indicator Verification. The applicable planning standard, Nuclear Energy Institute (NEI) 99-02, Revision 5, Regulatory Assessment Performance Indicator Guidelines, and the licensee's calculation P06-0002, CR3 MSPI Basis Document for the CR3 Nuclear Plant, were used to check the reporting for each data element. The inspectors discussed the PI data with the licensee personnel associated with performance indicator data collection and evaluation.

- Emergency AC power
- Residual heat removal/decay heat system
- Heat removal system
- High pressure injection system
- Cooling water system

b. Findings

No findings of significance were identified.

.2 Radiation Safety

a. Inspection Scope

The inspectors sampled licensee records to verify the accuracy of reported PI data for the periods listed below. To verify the accuracy of the reported PI elements, the reviewed data were assessed against guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 5.

Occupational Radiation Safety Cornerstone: The inspectors reviewed PI data collected from October 1, 2008, through September 30, 2009, for the Occupational Exposure Control Effectiveness PI. For the reviewed period, the inspectors assessed CAP records to determine whether HRA, VHRA, or unplanned exposures, resulting in ITS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessment results, and ED alarm data for cumulative doses and/or dose rates exceeding established set-points. The reviewed documents relative to this PI are listed in Sections 2OS1, 2OS2, and 4OA1 of the attachment.

Public Radiation Safety Cornerstone: The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the period of October 1, 2008, through September 30, 2009. For the assessment period, the inspectors reviewed cumulative and projected doses to the public and selected condition reports related to radiological effluent control. Documents reviewed are listed in Section 4OA1 of the attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.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 corrective action program (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 Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in section 4OA2.1 above, plant status reviews, plant tours, and licensee trending efforts. The inspectors' review nominally considered the six month period of July 2009 through December 2009. The review also included issues documented in the licensee's Equipment Performance Priority List dated December 16, 2009, and various 3rd quarter 2009 departmental CAP Rollup & Trend Analysis reports, nuclear oversite reports and maintenance rule (MR) reports. Corrective actions associated with a sample of the issues identified in the licensee's corrective action program were reviewed for adequacy.

b. Findings and Observations

No findings of significance were identified. The inspectors evaluated the licensee's trend methodology and determined that the licensee had performed a detailed review.

The inspector's review of licensee performance over the last six months noted the continuance of a negative trend associated with battery grounds. This negative trend was identified by the licensee and documented in NCR 200813 in 2006. The licensee has been effective in the identification and repair of equipment causing the grounds when the grounds are not intermittent. An upgrade of the ground detection system starting in 2010 should aid in the identification of equipment causing intermittent grounds.

.3 Annual Sample Review

a. Inspection Scope

The inspectors reviewed priority two NCR 357012 that addressed an adverse trend in the control of test equipment and procedure use and adherence. Improper control of test equipment and inadequate procedure use and adherence resulted in two plant trips in 2009 and a loss of an engineering safeguards motor control center (MCC 3B1). A common cause investigation of these three events was initiated to identify any work behaviors and any latent organizational weaknesses that needed to be addressed. The inspectors checked that the issue had been completely and accurately identified in the licensee's corrective action program; safety concerns were properly classified and prioritized for resolution, common cause determination was sufficiently thorough, and appropriate corrective actions were initiated. 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

No findings of significance were identified. The regulatory aspects associated with each plant event described above are documented in section 4OA3 of this report and NRC integrated inspection report 05000302/2009002. The inspectors found that the licensee's common cause investigation was both comprehensive and thorough. The use of Progress Energy personnel from another site on the investigational team personnel resulted in an unbiased look into the three events. Corrective actions that include the installation of engineered test points for routine connection of test equipment and the implementation of a site human performance excellence plan appear to be reasonable.

4OA3 Follow-up of Events and Notices of Enforcement Discretion

.1 Operator Performance During Non-Routine Event

a. Inspection Scope

For the one non-routine plant evolution described below, the inspectors reviewed the operating crew's performance, operator logs, and the overall plant response to the following one event.

- NCR 358724, Internal containment concrete cracking found during creation of a temporary containment opening to support steam generator replacement

b. Findings

No findings of significance were identified. During the creation of temporary opening in the reactor containment building to support steam generator replacement, the licensee discovered an internal crack in the concrete containment. The circumstances associated with the crack in the concrete containment wall are being assessed by an NRC special inspection team. The results of that inspection will be documented in NRC special inspection report 05000302/2009007.

.2 (Closed) LER 05000302/2009-002-00: In Technical Specification 3.0.3 for Greater Than One Hour due to a Loss of a Motor Control Center During Testing

a. Inspection Scope

The inspectors reviewed the root cause evaluation associated with LER 05000302/2009-002-00 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 Non-Cited Violation (NCV) of Improved Technical Specification (ITS) 5.6.1.1.a for failure to follow a plant procedure resulted in a loss of 480 volt engineered safeguards motor control center (ES MCC)-3B1. Concurrent with pre-existing conditions, the high pressure injection (HPI) system was declared inoperable and ITS 3.0.3 was entered for a period of one hour and 24 minutes.

Description: On April 30, 2009, the 480v engineered safeguards motor control center 3B1 (ES MCC-3B1) de-energized as a result of an arc flash generated when test equipment was connected to an ES MCC-3B1 circuit breaker. Prior to the event, test equipment was being connected to various circuit breakers per WO1496283 in order to collect data for the purpose of updating emergency diesel margin calculations. During the performance of the Work Order, electricians connected test leads to the “line” side of the “B” phase of circuit breaker MTMC-5-8AR vice the “load” side as directed by the WO instructions. The connection of the test equipment introduced a phase-to-ground arc flash that ultimately caused the de-energization of ES MCC-3B1.

Following the loss of the ES MCC-3B1, safety-related equipment makeup and purification system (MU) main lube oil pump MUP-2C was declared inoperable. Loss of main lube oil pump MUP-2C resulted in HPI system make-up pump MUP-1C being declared inoperable. Combined with a pre-existing clearance order (tagout) on HPI valve MUV-26, this condition resulted in one pump operation through two (2) injection legs. The HPI System was declared inoperable and ITS 3.0.3 was entered. Emergency diesel generator EGDG-1B and other mitigating systems associated with 480v ES MCC 3B1 was also declared inoperable. Due to the electrical transient, reactor building air handling fan and several control room complex loads were also tripped off. In addition, several extraction steam bypass valves and turbine drain system valves opened. The unit load demand was immediately taken to hand and reactor power was lowered due to the additional steam flow. The extraction steam bypass valves and turbine drain system valves were closed. The highest power level observed was approximately 2616 megawatt thermal (MWth) and the licensed power limit for Crystal River 3 is 2609 MWth.

ITS 3.0.3 was exited 1 hour and 24 minutes later when the clearance order on MUV-26 was lifted to restore the downstream cross-tie line and the downstream HPI line resulting in three (3) available injection legs. The event was reported to the NRC on June 22, 2009, pursuant to 10 CFR 50.73(a)(2)(i)(B) for being in ITS 3.0.3 for greater than one hour.

The licensee determined there were two root causes associated with this event.
The first root cause was for working on energized equipment with less than adequate controls in place. Specifically, the work order allowed work on energized equipment when it was possible to perform the work with the equipment out of service. The second root cause was attributed to procedure adherence / documentation culture not meeting expectations. Specifically, electricians improperly made the decision to deviate from work order instructions based on verbal directions from their supervisor, and without having the work instructions formally revised in accordance with the licensee's Work Management Process procedure.

Analysis: A failure to follow written work instructions in accordance with plant procedure ADM -NGGC-0104, Work Management Process, resulted in a loss of ES MCC-3B1 was a performance deficiency. Loss of ES MCC-3B1 and concurrent with pre-existing conditions caused the HPI system inoperability. The finding was more than minor since it affected the equipment availability attribute of the mitigating system cornerstone and resulted in ITS 3.0.3 entry for the HPI system being inoperable for approximately 1 hour and 24 minutes. The inspectors evaluated the finding against NRC SDP Phase 1 and determined that SDP Phase 2 was required because there was a loss of the HPI system safety function with no credit for the non-safety related direct current backup lube oil pump (MUP-3C). In addition to loss of 480 volt ES MCC-3B1 and reactor building air handling fan/control room complex loads tripped off and several extraction steam and turbine drain system valves opened during the electrical transient. Therefore, NRC Phase 3 evaluation was warranted to determine the level of safety significance. A Regional Senior Reactor Analyst performed a Phase 3 Significance Determination Process evaluation and concluded the performance deficiency was of very low safety significance (Green). The major assumptions of the evaluation were that the High Head Safety Injection function was out of service for the exposure period, the exposure period was 1.5 hours and there would be no recovery of the de-energized motor control center. The dominant accident sequence involved a support system failure of the Emergency Feedwater Indication and Control System rendering Main Feedwater and automatic control of Emergency Feedwater unavailable, operators were unable to manually control Emergency Feedwater flow causing its failure and with the High Head Safety Injection function lost due to the performance deficiency, core damage ensued. The inspectors determined the cause of the finding is related to the cross-cutting area of Human Performance with a work practices aspect (H.4 (c)). Specifically, the licensee failed to ensure that work scope changes involving safety-related equipment receive the appropriate level supervisor and management oversight which resulted in a plant procedural violation.

Enforcement: ITS 5.6.1.1.a states, in part, that written procedures shall be established, implemented, and maintained covering safety-related equipment maintenance activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 9.8 of plant procedure ADM-NGGC-0104, states that maintenance of safety-related equipment shall be pre-planed and performed in accordance with written procedures, documented instructions, or drawing appropriate to circumstances. Section 9.11.6 of plant procedure ADM-NGGC-0104 also states that the implementer review process provides an opportunity, prior to performing a given task, to examine the work package for adequacy, make any necessary changes, ensure specified parts are correct and available, and walk down the work to be performed.

Contrary to the above, on April 30, 2009, maintenance personnel failed to follow written work instructions in accordance with plant procedure ADM-NGGC-0104, Work Management Process, to ensure that work scope changes involving safety-related equipment receive the appropriate level of review and approval prior to proceeding with work. As a result, maintenance personnel improperly connected test equipment to an ES MCC-3B circuit breaker which resulting in the de-energization of ES MCC-3B1. Concurrent with pre-existing conditions, the HPI system was declared inoperable and ITS 3.0.3 was entered for a period of one hour and 24 minutes before the appropriate clearances were lifted to restore the HPI system to an operable status. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as NCR 333515, this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. This finding is identified as NCV 05000302/2009005-01, Failure to Follow a Plant Procedure Resulted in an Inoperable HPI System.

.3 (Closed) LER 05000302/2009-003-00, Manual Reactor Trip Due To Group 7 Control Rods Insertion Caused By Inadequately Protected Test Jumper

a. Inspection Scope

On August 24, 2009, while operating at 100 percent power, the Unit 3 Control Room operators received multiple alarms and observed the Group 7 control rods fully insert into the reactor core. The reactor was manually tripped prior to automatic actuation of the Reactor Protection System (RPS). This unexpected control rod insertion was caused by the failure of the Group 7 control rod programmer which resulted from inadvertent test jumper contact while using an improperly fused test jumper during maintenance on the system. The RPS responded as expected to the manual trip signal, control rods fully inserted. During and following the scram, all safety-related mitigating systems operated as designed, and all operator actions in response to the scram were deemed to be appropriate. This LER and its associated Nuclear Condition Report (NCR) 351705, including the Root Cause Analysis (RCA) for the event, as well as other NCR's associated with this event, were reviewed by the inspectors. The inspectors also reviewed previous NCRs related to improper use of jumpers and walked down the control rod programmer instrumentation. Furthermore, the inspectors interviewed the RCA team leader.

b. Findings

Introduction: A Green self-revealing NCV of Improved Technical Specification (ITS) 5.6.1.1.a was identified for the failure to follow the provisions of preventative maintenance procedure PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train. Failure to follow PM-126 caused the failure of the Group 7 control rod programmer during maintenance and resulted in the unexpected insertion of the Group 7 control rods fully into the core. This unexpected insertion of these control rods into the core caused control room operations personnel to manually trip the reactor from 100 percent power.

Description: On August 24, 2009, Unit 3 was operating at 100 percent power when the control room operators received multiple alarms and observed all eight of the Group 7 control rods fully insert into the reactor core. The reactor was manually tripped prior to automatic actuation of the reactor protection system (RPS).

The Group 7 control rods are the controlling group. They are expected to insert and withdraw in small increments corresponding to integrated control system (ICS) commands when the ICS is in Automatic. Movement of the rods is supervised by a programmer/controller (programmer) dedicated to Group 7. The unexpected drop of the Group 7 control rods was due to the failure of the programmer caused by inadvertent test jumper contact during performance of PM-126, using an improperly fused test jumper. These two conditions caused an over-current failure of the output driver within the Group 7 CRD programmer, which caused the unexpected insertion of the Group 7 control rods fully into the core. The licensee performed a root cause analysis of this event and concluded that the root cause was failure to follow the provisions of PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train. PM-126 directs use of a fused jumper, with a current limit of 0.1 amp. The jumper fuse was checked and found to be a 1.0 amp fuse. This is not consistent with the procedure, and is not adequate to protect the associated equipment which has a maximum current rating of 0.5 amp. During performance of PM-126, the jumper made inadvertent contact with a positive voltage/current source, causing an over-current failure of the output driver within the Group 7 CRD programmer. This failure resulted in the unexpected insertion of the Group 7 rods fully into the core, and the associated manual reactor trip by control room operations personnel.

The inspectors reviewed the licensee root cause analysis contained in NCR 351705, Significant Adverse Condition Investigation Report. This report concluded that the root cause of this event was poor performance of a task by the individual in that the worker did not verify that the proper fuse was in the jumper. The report also concluded that contributing causes of this event were insufficient level of detail in the procedure and a poor work environment. The inspectors determined that the identified root and contributing causes were accurate as stated.

The inspectors also reviewed NCR 351744 which was generated as a result of Rod 7-5 indicating a slower drop response than the other rods in Group 7, and NCR 351741 which was generated as a result of pressure pulsations occurring in various condensate system lines following the manual reactor trip. No significant issues were identified by the inspectors during these reviews.

Analysis: The licensee's failure to follow the provisions of PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train, was a performance deficiency which resulted in a manual reactor trip. This finding was determined to be more than minor because it was associated with the initiating events cornerstone attribute of Human Performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions were not available.

The cause of this finding was directly related to the cross-cutting area of Human Performance with a work practices aspect (H.4 (b)). Specifically, the workers failed to follow the preventative maintenance procedure.

Enforcement: Crystal River Unit 3 Nuclear Plant Improved Technical Specifications, Section 5.6.1.1.a, states in part, "Written procedures shall be established, implemented, and maintained covering the following activities: (a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 3(b) states "Instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation should be prepared, as appropriate, for the following systems: Control Rod Drive System (including part length rods)." Contrary to the above, the licensee failed to follow the provisions of PM-126, Electrical Checks of CRD [Control Rod Drive] Power Train. Failure to follow this procedure is a violation of Improved Technical Specifications. This failure resulted in a manual reactor trip. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as NCR 351705, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000302/2009005-02, Manual Reactor Trip Due To Group 7 Control Rods Insertion Caused by Inadequately Protected Test Jumper.

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. Findings

No findings of significance were identified.

.2 Unit 3 Steam Generator Replacement Inspection (IP 50001)

a. Inspection Scope

Design and Planning

The inspectors reviewed the following related to the licensee's steam generator replacement (SGR) project design and planning:

- scope and schedule
- engineering change (EC) packages
- 10 CFR Part 50.59 evaluation
- quality assurance program and corrective actions
- preparations for the creation of a temporary containment wall opening
- applicable engineering design, modification, and analysis associated with lifting, rigging, and transporting of the steam generators (SGs)
- radiation protection program controls, planning, and preparation
- security considerations associated with vital and protected area barriers affected by the SGR activities
- controls to minimize any adverse impact the activities may have had on other systems.

The licensee used ASME Boiler and Pressure Vessel Code (ASME Code) Section III, 1998 Edition through 2000 Addenda, and Section XI, 2001 Edition through 2003 Addenda, for the design, fabrication, replacement, and installation of the new replacement SGs (RSGs) and systems. The licensee used ASME Code Section III for the design, fabrication, installation, and nondestructive examination (NDE) as the construction code for the following major component modifications: reactor cooling system (RCS) primary piping connections; main steam, feedwater, and emergency feedwater piping connections; and the steel containment vessel. The inspectors reviewed and examined the SGR activities and compared them to the requirements of the ASME Code.

The inspectors reviewed EC 63038, Replacement Once Through Steam Generators (ROTSGs or RSGs), for Crystal River Unit 3. EC 63038 included the design changes, analyses, evaluations, safety analyses, 10 CFR Part 50.59 change evaluation, configuration, materials, implementation, and post modification testing acceptance. The inspectors reviewed the following other major modification packages: EC 62500, RCS Hot Leg Cutting and Welding; EC 63016, Containment Opening; EC 63025, Main Feedwater Flow Accelerated Corrosion (FAC) Pipe Replacement; EC 63026, RCS Cold Leg Cutting and Welding; EC 63027, Secondary Side Large Bore Pipe Cutting and Welding; EC 63034, Structural Interferences; and EC 63039 Replacement Steam Generator Anchorage. The inspectors also reviewed other miscellaneous and temporary modifications.

The inspectors reviewed design calculations and analyses for design methods, assumptions, loadings, computations, and accuracies. The inspectors also selected work order (WO) packages prepared for the construction and implementation of the ECs for review to determine whether appropriate work processes and quality control hold points were implemented.

The inspectors reviewed the licensee's screening of selected ECs to determine whether the modifications were evaluated in accordance with 10 CFR Part 50.59, "Changes, tests, and experiments."

Removal and Replacement

The inspectors periodically reviewed radiation controls. The inspectors verified that the tendons were removed after plant shutdown and prior to the hydro demolition of concrete for the containment building wall opening located above the equipment hatch in order to transport the SGs out from or into the containment. The inspectors observed the hydro demolition process. The licensee identified some concrete cracks/separation issues. The concrete separations could be seen within the entire perimeter of the opening. An NRC Special Inspection Team (SIT) was formed to inspect the separation issues. The SIT's charter included review of the evaluation, root cause, and corrective actions to be implemented to ensure an operable containment structure. The SIT remained ongoing at the end of this inspection period. Results of the SIT will be documented in NRC special inspection report 05000302/2009007.

The inspectors reviewed the cracks or delamination found on the exposed concrete around the containment building wall opening during and after the hydro demolition. The licensee evaluated the containment wall cracks to modify the horizontal transfer system (HTS) supporting structures. Prior to the removal of the original or old SGs (OSGs), the inspectors reviewed, observed, and evaluated the associated temporary and permanent modifications of the cutting, disconnecting, and the providing of temporary supports for the OSGs and cutoff piping. The inspectors observed: lifting, rigging, downending and upending, and transporting of the OSGs, RSGs, and associated equipment; machining and preparations of the existing piping for the connections to the RSGs; welding and NDE activities; and the radiological safety plan for the temporary storage and disposal building of the retired steam generators. The inspectors reviewed and observed the major structural modifications. The inspectors observed the licensee performance inspection of the steam generator hold down bolts to make sure that the bolts were acceptable to hold down the RSGs after the OSGs were moved out from the cubicles. During the SG removal and replacement, the inspectors observed licensee activities associated with controls for excluding foreign material, including the primary and secondary side of the steam generators and in the related RCS openings, and the establishment of operating conditions including defueling, RCS draindown and system isolation. The inspectors reviewed and inspected the status of security activities associated with vital and protected area plant and equipment changes required for the transportation of the OSGs and RSGs. The inspectors reviewed the installation, use, and removal of temporary services directly related to the steam generator replacement activities.

The inspectors observed and reviewed selected welding, NDE, preservice inspections, baseline inspections, and corrective action activities for the Class 1 and 2 piping and components of the RSGs.

The inspectors reviewed procedures, examination results, modification packages, and work order packages related to the modifications, including the construction hatch steel containment vessel (SCV) reinstallation, to ensure compliance with the requirements of the ASME Code.

This inspection remained ongoing at the end of this inspection period because the closure of the containment wall opening was not completed due to the cracks/delamination identified in the containment wall.

<u>RSG Fabrication, Preservice Inspection, and Baseline Inspection</u>

The inspectors reviewed records associated with the materials, fabrication, examination, and testing for the RSGs manufactured by Babcock & Wilcox Canada (BWC), and replacement hot leg piping subassemblies (“Candy Canes”), to verify compliance with the ASME Code.

The inspectors reviewed records and conducted interviews with appropriate plant personnel associated with the following RSG welds:

- W-22 (ASME Class 1, Lower Head to Lower Tubesheet Weld)
- W-65 (ASME Class 2, Lower Head to Shell Can #1)
- W-415 (Seal Weld Tube to Tubesheet Cladding)

The inspectors reviewed records associated with: certified material tests; materials NDE, weld preparation, weld preheat treatment and post weld heat treatment, hydrostatic testing, and preservice inspection (post hydrostatic testing); ASME Certificates of Accreditation and Authorization; weld material qualification; weld control sheet and welder’s records; and nonconformance reports. Other documentation reviewed included ASME data reports, ASME design specification, ASME design reports, and ASME repair/replacement documentation reconciliation of items, and the associated engineering change packages. The inspectors also walked down one RSG while stored on site, prior to movement into containment.

The inspectors reviewed documentation and interviewed plant personnel regarding the preservice and baseline testing of RSG tubing. All preservice inspection (PSI) or baseline inspection (BI) using eddy current testing (ECT) for the RSG tubes was completed off-site by BWC in Cambridge, Ontario, Canada. Full length 100 percent bobbin inspections were performed, and 100 percent X-probe data was acquired, but only analyzed in areas of interest. Profilometry data was also collected in the tubesheet expansion region. The inspectors reviewed the licensee’s degradation assessment, BWC ECT PSI report, ECT acquisition procedures, examination technique specification sheets, and data analysis procedures. In addition, the inspectors reviewed documentation regarding the manufacture of the RSG tubing by Sumitomo in Japan including heat treatment records and nonconformance reports.

The inspectors reviewed records and conducted interviews with plant personnel associated with the fabrication of the replacement hot leg piping subassemblies. Documents reviewed included the ASME Data Report, corrective action documents, the vendor ASME Certificate of Authorization, a self-assessment report, and the engineering change package. The inspectors also walked down both replacement hot leg piping subassemblies while stored on site, prior to movement into containment.

Welding: The inspectors reviewed a sample of welding activities associated with the installation of the RSGs to evaluate compliance with licensee/contractor procedures and the applicable ASME Code. The inspectors reviewed joint configuration drawings, welding procedures, welding specifications, welding procedure qualifications, welder qualification records, weld data records, nuclear condition reports (NCRs), and post weld heat treatment procedures (where applicable) for the welds listed below.

- EF-00-041, SG A Emergency Feedwater Pipe-to-Elbow
- EF-00-042, SG A Emergency Feedwater Pipe-to-Elbow
- MS-00-024, SG A Main Steam East Elbow-to-Pipe
- MS-00-034, SG A Main Steam West Elbow-to-Pipe

In addition to the record review described above, the inspectors performed field observations either via video monitors located outside containment or by direct observation inside containment of the machine welding of the RCS hot leg and cold leg piping welds listed below.

- RC-00-189, SG A Hot Leg Flow Meter Pipe-to-Existing-Riser-Pipe
- RC-00-190, SG A Hot Leg Pipe-to-Nozzle
- RC-00-191, SG A Cold Leg A1 Elbow-to-Nozzle
- RC-00-192, SG A Cold Leg A2 Elbow-to-Nozzle
- RC-00-193, SG B Hot Leg Flow Meter Pipe-to-Existing-Riser-Pipe
- RC-00-194, SG B Hot Leg Pipe-to-Nozzle
- RC-00-195, SG B Cold Leg B1 Elbow-to-Nozzle
- RC-00-196, SG B Cold Leg B2 Elbow-to-Nozzle

The inspectors performed field observations by direct observation inside containment of the manual welding of the piping welds listed below.

- EF-00-044, SG B Emergency Feedwater Pipe-to-Pipe
- EF-00-047, SG B Emergency Feedwater Pipe-to-Pipe
- FW-00-069, SG B Feedwater Pipe-to-Elbow
- FW-00-072, SG B Feedwater Pipe-to-Pipe
- FW-00-076, SG B Feedwater Pipe-to-Elbow
- FW-00-079, SG B Feedwater Pipe-to-Elbow
- MS-00-053, SG B Main Steam West Nozzle-to-Elbow
- MS-00-055, SG B Main Steam West Pipe-to-Pipe

The inspectors also reviewed and verified a sample of welding machine settings for the weld equipment to verify that welding parameters were being maintained within the qualified procedure limits.

NDE

The inspectors reviewed the NDE procedures, calibration and examination reports, and NCRs, and observed in-process NDE examinations including liquid penetrant examinations (PTs), magnetic particle examinations (MTs), radiographic examinations (RTs), and ultrasonic examinations (UTs) for the following piping or component welds and compared them to the requirements of the procedures and the ASME Code for the construction, preservice, and baseline inspections:

PT – Construction

- RC-00-190, SG A Hot Leg Pipe-to-Nozzle
- RC-00-193, SG B Hot Leg Riser Pipe-to-Pipe
- RC-00-191, SG A Cold Leg A1 Pipe-to-Nozzle
- RC-00-196, SG B Cold Leg B2 Pipe-to-Nozzle
- RC-00-192, SG A Cold Leg A2 Pipe-to-Nozzle

MT - Construction

- MS-00-043, SG B Main Steam East Nozzle-to-Elbow
- MS-00-045, SG B Main Steam East Pipe-to-Pipe

RT – Construction

- FW-00-045, SG A Feedwater Header Pipe-to-Pipe Elbow
- FW-00-046, SG A Feedwater Pipe-to-Pipe Elbow
- FW-00-048, SG A Feedwater Pipe-to-SG Elbow
- MS-00-043, SG B Main Steam East Nozzle-to-Elbow
- MS-00-044, SG B Main Steam Pipe-to-Pipe
- MS-00-045, SG B Main Steam East Nozzle-to-Elbow
- EF-00-040R1, SG A Emergency Feedwater Pipe-to-Pipe
- EF-00-045C1, SG A Emergency Feedwater Pipe-to-Pipe
- EF-00-046C1, SG A Emergency Feedwater Pipe-to-Pipe

UT – Preservice and Baseline

- MS-00-043, SG B Main Steam East Nozzle-to-Elbow
- MS-00-045, SG B Main Steam East Pipe-to-Pipe

Containment Construction Hatch Opening and Closure - Steel and Concrete Containment

The inspectors reviewed the licensee's activities associated with the concrete removal and the removal and restoration of the steel containment liner plate (SCLP) for the containment construction hatch opening, as detailed in the EC 63016, Containment Opening.

The inspectors reviewed the plans for the cutting and restoration of the SCLP for the construction hatch opening and compared post testing requirements to the applicable ASME Code. The inspectors observed the hydraulic cutting, hydro demolition, of concrete for the containment construction hatch opening and reviewed the work order packages for the cutting to verify the steps had been completed and documented.

The inspectors reviewed the welding procedures, procedure qualification records, and welder qualification records to confirm that the Code-required essential and supplemental essential welding variables were met. The inspectors reviewed the work order package including welding electrode receipt inspection, vacuum box leak testing, MT records, material certification records, and qualification and certification records for NDE personnel, equipment, and consumables.

The inspectors reviewed the containment wall delamination that was found next to the Hoop or Horizontal Tendons about 10 inches from the outside surface of the containment wall. The delamination was identified on the exposed concrete wall surfaces around the opening of containment during and after the hydro demolition for the opening. The inspectors observed the licensee perform ultrasonic impulse response and impact echo examinations as NDE methods to detect and scope the delamination areas.

The licensee planned to address the closure of the containment wall opening as part of the repair of the containment wall delamination.

<u>Heavy Load, Rigging, Lifting, and Transporting Activities</u>

The inspectors reviewed Progress Energy maintenance procedures MNT-NGGC-0005, Control of Rigging and Temporary Loads, Revision 3 and MNT-NGGC-0021, Lifting and Rigging Practices and Equipment, Revision 0. The inspectors reviewed the SG lifting preparation activities as described in the following EC Packages and lifting equipment load test data to ensure that they were prepared in accordance with regulatory requirements, appropriate industrial codes and standards, and to verify that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures: EC 63020, Outside Erection Crane and Inside Auxiliary Crane, Revision 7; EC 63022, Steam Generator Rigging and Transport including the Outside Lift System (OLS), Hatch Transfer System (HTS), and Temporary Lifting Device (TLD), Revision 31; and EC 63023, OSG/RSG Haul Route, Revision 3.

The inspectors examined SGR project lifting, rigging, and transporting equipment including the polar crane, mobile crane, TLD, HTS including skid system, the down/up-ender device, OLS, and the self-propelled modular transporter (SPMT). The inspectors observed portions of rigging, lifting, transportation, and positioning of the original and replacement SGs.

The inspectors reviewed procedures, calculations, drawings, work packages, crane and equipment operator training and certificates, and load and function test records to determine whether they were in accordance with regulatory requirements and appropriate industrial codes and standards. The inspectors also reviewed polar crane and containerized winch system inspection and maintenance records.

The inspectors reviewed the licensee's analyses for buried piping located beneath the transport path as documented in Calculation S06-0019, Evaluation of Buried Utilities under Haul Route, Revision 0. The inspectors also reviewed calculations related to heavy load lifting systems, including: S06-0009, TLD Loads on Polar Crane Rail, Revision 0; S06-0012, Outside Lifting System East Foundation Design, Revision 3; and S06-0015, Seawater Room Wall Evaluation, Revision 1.

Quality Assurance (QA) Program and Corrective Actions

The inspectors conducted a review of the quality assurance program and its implementation for the SG replacement to assess compliance with the requirements of 10 CFR Part 50, Appendix B. The inspectors reviewed daily quality summaries, quarterly reports, QA audit procedures, audit reports, and QA personnel certifications, and conducted interviews with QA and quality control (QC) personnel.

The inspectors reviewed the surveillance reports and nonconformance reports issued for the root cause analyses, evaluations, repairs, or disposition during the manufacturing of the RSGs. The inspectors also selected the NCRs and Audit Reports for review during the implementation of the removal and reinstallation of the SGs. The review was to ensure that issues were being identified appropriately, entered into the corrective action program (CAP) correctly, and dispositioned adequately. The inspectors also reviewed procedures associated with the training of personnel for the identification, disposition, and documentation of the NCRs.

Post Installation Verification and Testing

The inspectors reviewed the post installation verification and testing program to verify that the required post installation verification and testing, procedural changes, and the adjustment of the instruments were properly identified. The inspectors verified that the modifications were completed in accordance with the design documents and reviewed the required post modification tests.

The inspectors reviewed the work packages to verify that the required NDE and preservice inspections were completed as designed and met code requirements for the major modifications of the SGs, structures, and piping.

The post modification or installation verification and testing related to RCS leakage testing, containment pressure leak testing, SG thermal and hydraulic performance testing, other instrument setting or testing were not completed during this quarter because the installation of the containment construction hatch opening closure and the repair of the containment wall delamination was delayed to the next year. Additionally, steam generator secondary side leakage testing had not been completed by the licensee at the conclusion of this inspection period.

All documents reviewed for IP 50001 are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.3 Steam Generator Replacement Inspection (Radiation Protection Activities)

a. Inspection Scope

The inspectors reviewed and evaluated ALARA planning and implementation of radiological controls for current SG removal/replacement, transport, and onsite storage activities

Radiation, contamination, and airborne radioactivity surveys were reviewed to evaluate radiological work conditions and the adequacy of prescribed postings and HP controls. The inspectors reviewed RWPs and evaluated ED alarm settings, worker and HPT instructions, special dosimetry needs, and protective clothing/equipment requirements. ALARA planning, dose reduction initiatives, and actual doses received were reviewed and discussed with the radiation protection and ALARA staff. The inspectors also reviewed plans and dose estimates for on-site storage of the old SGs and associated piping. Individual worker doses and project dose expenditures for the SG removal/replacement project activities were reviewed and discussed in detail with licensee representatives.

Initial administrative and physical controls associated with the SGs and associated piping prior to their removal from containment and subsequent to their placement in the onsite storage facility were observed and evaluated. The inspectors evaluated implementation of selected radiological controls including worker briefings, postings, shielding, and air monitoring associated with removal of the old SGs from containment. The inspectors also toured the onsite temporary storage facility and conducted independent radiological surveys of the onsite storage facility subsequent to placement of the SG components into the facility. In addition, the inspectors discussed on-going radiological controls and surveillances associated with the onsite storage facility.

The SG removal and replacement activities were evaluated against 10 CFR Parts 19 and 20, and approved licensee design documents and applicable procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Sections 2OS1, 2OS2, and 4OA1 of the attachment.

The inspectors completed the specified radiation protection line-items detailed in IP 50001.

b. Findings

No findings of significance were identified

.4 (Closed) NRC Temporary Instruction (TI) 2515/174, Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative

a. Inspection Scope

The inspectors reviewed elements of the licensee’s environmental monitoring program to evaluate compliance with the voluntary Groundwater Protection Initiative (GPI) as described in Nuclear Energy Institute (NEI) 07-07, Industry Ground Water Protection

Initiative – Final Guidance Document, August 2007 (ADAMS Accession Number ML072610036). Inspectors interviewed personnel, performed walk-downs of selected areas, and reviewed the following items:

- Site characterization of geology and hydrology as described in the licensee's groundwater flow study report
- valuations of systems, structures, or components (SSCs) that contain or could contain licensed material and evaluations of work practices that involved licensed material for which there is a credible mechanism for the licensed material to reach the groundwater
- Implementation of the onsite groundwater monitoring program to monitor for potential licensed radioactive leakage into groundwater
- Locations of groundwater monitoring wells installed as a result of implementation of the Groundwater Protection Initiative
- Procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long term decommissioning impacts
- Records of leaks and spills recorded in the licensee's decommissioning files in accordance with 10 CFR 50.75(g)
- Licensee briefings of local and state officials on the licensee's groundwater protection initiative
- Procedures for notification to the local and state officials and to the NRC regarding detection of leaks and spills
- Procedures for external notifications and reports if an onsite groundwater sample exceeds the criteria in the radiological environmental monitoring program
- Groundwater monitoring results as reported in the annual radiological environmental operating report
- Licensee and industry assessments of implementation of the groundwater protection initiative.

b. <u>Findings</u>

No findings of significance were identified with the licensee's implementation of NEI 07-07. This completes the NRC Region II inspection requirements.

.5 <u>(Discussed) Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172, Revision 1)</u>

a. <u>Inspection Scope</u>

The inspectors conducted a review of the licensee's activities regarding licensee dissimilar metal butt weld (DMBW) mitigation and inspection implemented in accordance with the industry self-imposed mandatory requirements of Materials Reliability Program (MRP-139), "Primary System Piping Butt Weld Inspection and Evaluation Guidelines." Temporary Instruction (TI) 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds," was issued February 21, 2008, to support the evaluation of the licensees' implementation of MRP-139. This inspection was limited to review of MRP-139 activities performed after March 2008.

TI 2515/172 was performed in March 2008 as documented in Inspection Report 2008002. During that time a complete program review (per TI 2515/172 paragraph 03.05) was performed.

The documents reviewed by the inspector for this inspection are listed in the attachment to this report.

From October 12 – 16, 2009, the inspectors performed a review in accordance with TI-172 as described in the Observation Section below:

b. Observations

In accordance with requirements of TI 2515/172, Revision 0, the inspectors evaluated and answered the following questions:

(1) Implementation of the MRP-139 Baseline Inspections

1. Have the baseline inspection been performed or are they scheduled to be performed in accordance with MRP-139 guidance?

   Yes, the licensee has performed all required baseline inspections at the time of this review.

   Overlays were installed on all applicable Pressurizer welds with the exception of the Surge Nozzle, which was attempted to be overlaid unsuccessfully in a previous refueling outage. Baseline UT exams for all these welds occurred at the time. Baseline UT exams for the Core Flood Nozzles (2) were performed at this time as well. All of these are described in Crystal River report 2008002.

   Baseline UT exams for the Cold Leg RCP Suction Nozzles (4), Cold Leg Discharge Nozzles (4), Cold leg HPI/MU Nozzle, and Cold Leg HPI Nozzles (3) were performed during this refueling outage, and were all performed in accordance with MRP-139 guidance.

2. Is the licensee planning to take any deviations from the MRP-139 baseline inspection requirements of MRP-139? If so, what deviations are planned, what is the general basis for the deviation, and was the NEI-03-08 process for filing a deviation followed?

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

(2) Volumetric Examinations

1. Were the examinations performed in accordance with the MRP-139, Section 5.1 guidelines and consistent with NRC staff relief request authorization for weld overlaid welds?

   Yes, all examinations were performed in accordance with applicable requirements.

2. Were examinations performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes, all personnel performing the examinations were qualified under the Performance Demonstration Initiative (PDI) program.

3. Were examinations performed such that deficiencies were identified, dispositioned, and resolved?

   Yes, examinations were performed in a manner where deficiencies were identified, dispositioned and resolved.

(3) Weld Overlays

1. Was the weld overlay repair performed in accordance with ASME Code welding requirements and consistent with NRC staff relief request authorizations? Has the licensee submitted a relief request and obtained NRR staff authorization to install the weld overlays?

   Yes, the weld overlay was performed in accordance with ASME requirements as well as the NRC-issued Safety Evaluation Report

2. Was the weld overlay repair performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes, the personnel performing the weld overlay were qualified in accordance with ASME Section IX requirements.

3. Was the weld overlay performed such that deficiencies were identified, dispositioned, and resolved?

   Yes, deficiencies were identified, dispositioned and resolved during the overlay welding process.

(4) Mechanical Stress Improvement (SI)

There were no stress improvement activities performed or planned by this licensee to comply with their MRP-139 commitments.

(5) Application of Weld Cladding and Inlays

This portion of the TI was not inspected during the period of this report.

(6) Inservice Inspection Program

1. Has the licensee prepared an MRP-139 inservice inspection program? If not, briefly summarize the licensee's basis for not having a documented program and when the licensee plans to complete preparation of the program.

Yes, the welds associated with MRP-139 have been entered into the Augmented Exam section of the licensee's ISI program.

2. In the MRP-139 inservice inspection program, are the welds appropriately categorized in accordance with MRP-139? If any welds are not appropriately categorized, briefly explain the discrepancies.

   Yes, the welds are appropriately categorized in accordance with MRP-139.

3. In the MRP-139 inservice inspection program, are the inservice inspection frequencies, which may differ between the first and second intervals after the MRP-139 baseline inspection, consistent with the inservice inspections frequencies called for by MRP-139?

   Yes, the inspection frequencies of the Augmented exams are consistent with the requirements of MRP-139.

4. If any welds are categorized as H or I, briefly explain the licensee's basis of the categorization and the licensee's plans for addressing potential PWSCC.

   Welds previously categorized as H or I have been inspected and have been recategorized.

5. If the licensee is planning to take deviations from the MRP-139 inservice inspection guidelines, what are the deviations and what are the general bases for the deviations? Was the NEI 03-08 process for filing deviations followed?

   No, the licensee is not planning to make any requests for deviation from MRP-139 requirements.

c. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 11, 2010, the resident inspectors presented the inspection results to Mr. J. Franke, Site Vice President, and other members of licensee management. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee Identified Violations

The following issue of very low safety significance (Green) was identified by the licensee and was a violation of NRC requirements. This issue met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

10 CFR 26.205(d) requires, in part, that individuals subject to work hour controls do not exceed 26 work hours in any 48-hour period and 72 work hours in any 7-day period; requires a 34-hour break in any 9-day period; and a 10-hour break between successive work periods. During the period of October 12 to October 19, 2009, one worker exceeded 26 hours in a 48-hour period; nine workers exceeded 72 hours in a 7-day period; five workers did not have a 34-hour break in a 9-day period; and two workers did not have the required 10-hour break between successive work periods. The violation was limited to one work group, Florida Transmission Personnel, who were on-site to support outage work. The licensee determined that the Transmission personnel did not have a firm understanding of the revised 10 CFR Part 26 requirements. The finding was more than minor because, if left uncorrected, it would become a more significant safety concern. Specifically, the excessive work hours would increase the likelihood of human performance errors during plant maintenance activities that could affect equipment performance. The finding is of very low safety significance because no significant events or human performance issues were directly linked to personnel fatigue as a result of the hours worked. This issue was documented in the licensee's corrective action program as NCR 361777.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
J. Holt, Plant General Manager
J. Dufner, Manager, Maintenance
S. Cahill, Manager, Engineering
J. Huegel, Manager, Nuclear Oversite
P. Dixon, Manager Training
C. Morris, Manager, Operations
D. Westcott, Supervisor, Licensing
B. Akins, Superintendent, Radiation Protection
C. Poliseno, Supervisor, Emergency Preparedness
I. Wilson, Manager Outage and Scheduling
J. Franke, Vice President, Crystal River Nuclear Plant
M. Bishara, SGR Design Engineering Manager
F. Dola, Nuclear Oversight Superintendent
J. Cravens, SGR Welding Engineer
R. Griffith, SGR Task Manager
K. Henshaw, SGR Rigging Supervisor
D. Herrin, Licensing Engineer
D. Jopling, SGR Civil Structural Supervisor
B. Kelley, RT Level III
D. Mayes, SGR Welding Engineer
W. Nielsen, SGR QC Supervisor
S. Powell, SGR Licensing engineer
J. Terry, SGR Project Manager
R. Vessley, SGR QC Supervisor

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Closed

05000302/2009002-00	LER	In Technical Specification 3.0.3 Greater Than One Hour Due to Loss of Motor Control Center During Testing (Section 4OA3.2)
05000302/2009005-01	NCV	Failure to Follow a Plant Procedure Resulted in an Inoperable HPI System (Section 4OA3.2)
05000302/2009003-00	LER	Manual Reactor Trip Due to Group 7 Control Rods Insertion Caused By Inadequately Protected Test Jumper (Section 4OA3.3)
05000302/2009005-02	NCV	Manual Reactor Trip Due to Group 7 Control Rods Insertion Caused by Inadequately Protected Test Jumper (Section 4OA3.3)

Closed		
05000302/2515/173	TI	Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative (Section 4OA5)
Discussed		
05000302/2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172, Revision 1) (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

AP-730, Grid Instability
AI-513, Seasonal Weather Preparations
NGGM-IA-0003, Transmission Interface Agreement for Operation, Maintenance, and Engineering Activities at Nuclear Plants
AP-1040, Aux Building Flooding
AP-1050, Turbine Building Flooding

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

Section 1R08 Inservice Inspection Activities

Procedures
ADM-NGGC-0112, Reactor Coolant System Material Integrity Management Program, Rev. 2
AI-608, Plant Leak Management, Rev. 7
EGR-NGGC-0207, Boric Acid Corrosion Control, Rev. 3
NDEP-0612, VT-2 Visual Examination of Nuclear Power Plant Components, Rev. 22
SP-204, Class 1 System, System Leakage Test for Inservice Inspection, Rev. 22

Corrective Action Documents
AR 255848, Magnetic Particle Inspection Materials not Traceable, dated 11/20/7
AR 256010, Various Areas of RB Concrete Containment are Degraded, dated 11/23/7
AR 352683, ISI Exam Limited due to Configuration, dated 8/28/09
AR 354158, ISI Exam Limited due to Configuration, dated 9/8/09

AR 354309, ISI Exam Limited due to Configuration, dated 9/9/09
AR 354316, ISI Exam Limited due to Configuration, dated 9/9/09
AR 354317, ISI Exam Limited due to Configuration, dated 9/9/09
AR 354323, ISI Exam Limited due to Configuration, dated 9/9/09
AR 354328, ISI Exam Limited due to Configuration, dated 9/9/09
NCR 269121, Boric Acid on RCV-177, dated 3/5/08
NCR 308537, Boric Acid on WDV-25, dated 11/27/08
NCR 346897, Boric Acid on MU-101-FC, dated 7/24/09
NCR 360809, Scaffolder with Improper Tie-Off in Reactor Building, dated 10/14/2009
NCR 360878, NRC Inspector Questions Validity of Incore Nozzles, dated 10/14/2009
NCR 360890, SGR T-Power Cable Ground Conductor Not Connected, dated 10/14/2009

<u>Other</u>
0010522722, Exelon PowerLabs Certificate of Calibration for Light Meter 105253, dated 12/1/2008
0010536393, Exelon PowerLabs Certificate of Calibration for IR Thermometer 105533, dated 4/9/2009
144400, Self-Assessment Report: Boric Acid Corrosion Control Program, dated 7/24-29/2005
3FO688-02, Letter: Crystal River 3 Response to GL 88-05, dated June 1, 1988
MAGNAFLUX Certificate of Compliance for Magnetic Particle Inspection Material Batch 04K075, dated 10/25/04
MT-09-087, Magnetic Particle Examination Report for MS-6A Pipe-to-Elbow Weld, dated 10/13/2009
PQR 6, Procedure Qualification Record, dated 7/10/76
PQR 6A, Procedure Qualification Record, dated 10/6/81
PQR 6B, Procedure Qualification Record, dated 8/6/82
PQR 6C, Procedure Qualification Record, Rev. 7
SII006-07-03-08852-1, Laboratory Testing Inc. Certified Test Report for Soundsafe 07120 Couplant, dated 3/29/2007
Sonic Systems International Certificate of Qualification, LII-PDI (Hancock), dated 7/27/09
Sonic Systems International Certificate of Qualification, LII-PDI-UT-1 (Fish), dated 7-21-09
Sonic Systems International Certificate of Qualification, LII-PDI-UT-1 (Hancock), dated 7/27/2009
Sonic Systems International Certificate of Qualification, MT (Fish), dated 7-21-09
Sonic Systems International Certificate of Qualification, MT (Hancock), dated 7/27/2009
Sonic Systems International Certificate of Qualification, VT & LII-MT (Fish), dated 7/21/09
Sonic Systems International Visual Acuity Examination Record (Fish), dated 7/19/2009
Sonic Systems International Visual Acuity Examination Record (Fish), dated 7/19/2009
Sonic Systems International Visual Acuity Examination Record (Hancock), dated 7/27/2009
Sonic Systems International Visual Acuity Examination Record (Hancock), dated 7/27/2009
UT-09-119, UT Calibration/Examination Report for MS-6A Pipe-to-Elbow Weld, dated 10/13/2009
VT-09-150, Visual Examination for Boric Acid Detection: Incore Nozzles, dated 10/1/2009
VT-09-183, Visual Examination for Boric Acid Detection: RPVH CRDM Nozzles, dated 10/15/2009
WCAL-013, Wesdyne Certificate of Calibration for 11-lb Weight WEM 04053, dated 12-04-03
Weld Procedure Qualification – Manual GTAW (Crowley), dated 02/09/94
Weld Procedure Qualification – Manual GTAW (Davis), dated March 3, 1997

Weld Procedure Qualification – Manual GTAW/SMAW (Davis), dated 3/10/99
WO 00754691, Replace Valve MUV-401, Rev. 2
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