UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

April 27, 2010

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

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2010002

Dear Mr. Franke:

On March 31, 2010, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Crystal River Unit 3. The enclosed integrated inspection report documents the inspection findings which were discussed on April 12, 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, one NRC identified finding of very low safety significance (Green) was identified. The finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance of the issue and because it was entered into your corrective action program, the NRC is treating the issue as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. Also, 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 05000302/2010002
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).

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Letter to J. Franke from Marvin D. Sykes dated April 27, 2010

SUBJECT: CRYSTAL RIVER UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000302/2010002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2010002
Licensee:	Progress Energy (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	Crystal River, FL
Dates:	January 1, 2010 – March 31, 2010
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector R. Aiello, Senior Operations Engineer R. Chou, Reactor Inspector P. Higgins, Project Engineer S. Ninh, Senior Project Engineer K. Schaaf, Operations Engineer N. Smith, Project Engineer
Approved by:	M. Sykes, Chief, Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2009002; 01/01/2010-03/31/2010; Crystal River Unit 3; Follow-up of Events and Notices of Enforcement Discretion; Other Activities.

The report covered a three month period of inspection by resident inspectors, two operations engineers, three project engineers, and one reactor engineer. One Green NCV was identified. The significance of most findings is identified by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process", Revision 4, dated December 2006.

A. NRC Identified & Self-Revealing Findings

Cornerstone: Mitigating Systems

Green: The inspectors identified a non-cited violation of Crystal River Unit 3 Operating License Condition 2.C.(9), for failure to take compensatory actions when a main control room (MCR) and cable spreading room (CSR) floor/ceiling interface access hatch was open rendering the CSR Halon fire extinguishing system inoperable. Once identified, the licensee initiated nuclear condition report (NCR) 266356 in the corrective action program to address this issue.

The finding is more than minor because it is associated with the protection against external factors attribute, i.e., fire, and degraded the Mitigating Systems cornerstone objective to ensure the availability of systems that respond to initiating events. Specifically, the finding adversely affected the suppression fire extinguishing system capability defense-in-depth element. The inspectors evaluated this finding under NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process (SDP). The inspectors determined that a Phase 2 SDP was required for this finding because the CSR Halon concentration was highly degraded; a fire could occur due to non-qualified cables or transient combustibles while the hatch between the MCR and CSR was open; a duration factor (exposure time) was between 3 and 30 days; and control room operators evacuated the MCR in response to the fire. However, Phase 2 SDP of IMC 0609 Appendix F does not currently include explicit treatment of fires leading to MCR abandonment, either due to fire in the MCR or due to fires in other fire areas. Therefore, a Phase 3 SDP evaluation for this type of finding was needed. A Regional Senior Reactor Analyst performed a Phase 3 SDP for this finding and concluded that the finding was of very low safety significance (Green). The major assumptions and the dominant accident sequence were discussed in the 4OA5 analysis section of this report. The inspectors did not identify a cross-cutting aspect associated with this finding because it does not reflect current licensee performance. (Section 4OA5)

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 3 began the inspection period with the full core off-loaded to the spent fuel pool. The unit remained in this condition for the remainder of the inspection period.

REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

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

- January 3 -14 with nightly outside temperatures near or below freezing

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment Walkdowns

a. Inspection Scope

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

- Emergency diesel generator EGDG-1A system using operating procedure OP-707, Operation of the ES Emergency Diesel Generators, while EGDG-1B was out of service for scheduled maintenance

- Emergency diesel generator EGDG-1B system using operating procedure OP-707, Operation of the ES Emergency Diesel Generators, while EGDG-1A was out of service for scheduled maintenance
- B train decay heat closed cycle cooling (DC), raw water (RW) pump RWP-2B, and RWP-3B systems using OP-404, Decay Heat Removal System and OP-408, Nuclear Services Cooling System, while A train RW systems (RWP-2A and RWP-3A) were out of service for planned maintenance

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors conducted a detailed walkdown/review of the alignment and condition of both trains of spent fuel pool cooling with the full core off loaded to the spent fuel pool. The inspectors used licensee operating procedure, OP-406, Spent Fuel Cooling System, as well as design documents, and reviewed the applicable portions of the Final Safety Analysis Report (FSAR) to verify proper system alignment. This completes one sample of a complete system alignment.

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

- Piping and pipe supports did not show evidence of water hammer;
- Oil reservoir levels indicated normal;
- Snubbers did not indicate any observable hydraulic fluid leakage;
- Component foundations were not degraded;
- No fire protection hazards; and
- Temporary scaffolding had been installed per station procedures.

A review of outstanding maintenance work orders was performed to verify that any deficiencies did not significantly affect the system function. In addition, the inspectors reviewed nuclear condition reports (NCRs) to verify that system problems were being identified and appropriately resolved.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Area Walkdowns

a. Inspection Scope

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

- EGDG-1A and associated control room area
- Main control room
- Auxiliary building 95' level service water pump area
- Emergency Feed Tank EFT-2 building
- B ES 4160V switchgear room

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

On February 17 and on March 3 the inspectors observed licensee fire brigade response to a simulated fire. Both drills involved a fire in the turbine building hydrogen seal oil unit. The inspectors checked the brigade's communications, ability to set up and execute fire operations, and their use of fire fighting equipment. The inspectors verified compensatory actions were in place to ensure that additional alarms which may be received during the drill were addressed. Additionally, the inspectors verified that the licensee considered the aspects as described below when the brigade conducted the firefighting activities and during the post drill critique. The inspectors attended each drill's post-drill critique to check that the licensee's drill acceptance criteria were met and that any discrepancies were discussed and resolved. Administrative Instruction AI-2205, Administration of CR-3 Fire Brigade, was reviewed to assure that acceptance criteria were evaluated and deficiencies were documented and corrected. In addition, the inspectors reviewed the storage, training, expectations for use and maintenance associated with the self-contained breathing apparatus (SCBA) program. This completed one sample representing observation of selected fire drills. Documents reviewed are listed in the attachment. The inspectors observed that:

- The brigade, including the fire team leader, had a minimum of five members;
- Members set out designated protective clothing and properly donned gear;
- SCBA were available and properly used;
- Control room personal verified fire location, dispatched fire brigade and sounded alarms. Emergency action levels were declared and notifications were completed.
- Fire brigade leader as well as the control room senior reactor operator had copies of the pre-fire plans.
- Brigade leader maintained control: Members were briefed, discussed plan of attack, received individual assignments, and completed communications checks. Plan of attack discussions were consistent with pre-fire plans

- Fire brigade arrived at the fire scene in a timely manner, taking the appropriate access route specified in the strategies and procedures.
- Control and command was set up near the fire scene and communications were established with the control room and the fire brigade members.
- Effectiveness of radio communication between the command post, control room, plant operators and fire brigade members;
- Fire hose lines reached all necessary fire hazard locations, were laid out without flow constrictions, and were simulated as being charged with water.
- The fire area was entered in a controlled manner following the two person rule .
- The fire brigade brought sufficient fire-fighting equipment to the scene to properly perform its fire-fighting duties.
- The fire brigade checked for fire victims and fire propagation into other areas.
- Effective smoke removal operations were simulated in accordance with the pre-fire plan.
- The fire-fighting plan strategies were utilized.
- The drill scenario was followed, and the drill acceptance criteria were met.
- All fire fighting equipment was retuned to a condition of readiness.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

Annual Review

a. Inspection Scope

The inspectors observed maintenance personnel perform heat exchanger inspections and cleaning for the two listed heat exchangers. The inspector observed as-found conditions when the heat exchangers were opened for inspection and tube cleaning to verify the heat exchangers were in an acceptable condition to perform their design function. The documents reviewed are listed in the attachment.

- Service water heat exchanger SWHE-1C
- Decay heat closed cycle cooling heat exchanger DCHE-1A

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On January 26 the inspectors observed and assessed licensed operator crew response and actions for the Crystal River Unit 3 licensed operator simulator evaluated session SES-53. Session SES-53 involved a loss of a condensate pump, a large leak in the service water (SW) system, seat leakage through turbine bypass valves and a reactor coolant system (RCS) leak in containment. The inspectors observed the operator's use of abnormal procedures AP-330, Loss of Nuclear Service Cooling; AP-510, Rapid Power Reduction; and AP-520, Loss of RCS Coolant or Pressure. Additionally, emergency operating procedures used during the scenario included EOP-02, Vital System Status Verification and EOP-05, Excessive Heat Transfer. The operator's actions were verified to be in accordance with the above procedures. Event classification and notifications were verified to be in accordance with emergency management procedure EM-202, Duties of the Emergency Coordinator. The simulator instrumentation and controls were verified to closely parallel those in the actual control room. The inspectors attended the management crew critique and evaluation to verify the licensee had entered any adverse conditions into the corrective action program. The inspectors evaluated the following attributes related to crew performance:

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On February 10, 2009, the licensee completed the comprehensive biennial requalification written examinations and annual requalification operating tests required to be administered to all licensed operators in accordance with 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of the written examinations, individual operating tests and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 0609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

.3 Biennial Review by Regional Inspector

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of February 1, 2010, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the facility licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5, 1998, "American National Standard for Nuclear Power Plant Simulators for Use in Operator Training and Examination." The inspectors observed one crew during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records, simulator performance test records, operator feedback records, licensed operator qualification records, remediation plans, watchstanding records, and medical records. The records were inspected using the criteria listed in Inspection Procedure 71111.11. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. The review included the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations, and the resolution, of historical equipment problems. For those systems, structures, and components within the scope of the Maintenance Rule (MR) per 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The documents reviewed are listed in the attachment. The inspectors conducted this inspection for the following two issues:

- NCR 364258, Repeat MR functional failure of diesel fuel pump DFP-1D
- NCR 380893, RWP-2A bearing failure

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

Inspection procedure IP 71111.13, Maintenance Risk Assessments and Emergent Work Control specifies verification of performance of risk assessments for planned or emergent maintenance activities during all modes of operation. Due to the extended no mode condition i.e., full core off loaded to the spent fuel pool, to support reactor building containment repair, there were no opportunities for inspection in this area during the inspection period. Outage related risk assessment monitoring was performed under section 1R20.

1R15 Operability Evaluations

The inspectors reviewed the following two NCRs to verify operability of systems important to safety was properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. The inspectors determined if operability of systems or components important to safety was consistent with ITS, the FSAR, 10 CFR Part 50 requirements, and when applicable, NRC Inspection Manual, part 9900, Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." The inspectors reviewed licensee NCRs, work schedules, and engineering documents to check if operability issues were being identified at an appropriate threshold and documented in the corrective action program, consistent with 10 CFR 50, Appendix B requirements and licensee procedure NGGC-CAP-200, Corrective Action Program. Due to the extended no mode condition i.e., full core off loaded to the spent fuel pool, to support reactor building containment repair, there were few opportunities for inspection in this area during the inspection period.

- NCR 385436, EGDG-1A Top Cover/Inspection Cover Minor Leak At Weld
- NCR 388509 Raw water spool piece (RW-4) below minimum wall thickness

1R18 Plant Modifications

Permanent Plant Modifications

a. Inspection Scope

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

- EC 72085, Rupture Disk Modification on Nuclear Service Water Heat Exchangers

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 six post-maintenance tests reviewed are listed below:

- SP- 375A, CHP-1A And Valve Surveillance, after performing maintenance per work order (WO) 1403095
- SP-344C, Containment Cooling System Fan And Valve Surveillance, after performing maintenance per WO 1638755
- SP-375B, CHP-1B And Valve Surveillance, after performing maintenance per WOs 1638759 and 1384581
- SP-354B, Monthly Functional Test of the Emergency Diesel Generator EGDG-1B, after performing 2-YR preventative maintenance per WOs 1444356, 1391181, and 1357712
- SP-354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A, after performing 2-YR preventive maintenance per WOs 1394794, 218756, 1064421, and 1567847
- SP-344A, RWP-2A, SWP-1A and Valve Surveillance after performing motor replacement of RWP-2A per WO 1698448

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 integrated inspection report 05000302/2009005 documented NRC inspection activities for the last quarter of 2009. During this 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 05000302/2010003. Documents reviewed are listed in the Attachment.

- Outage related risk assessment monitoring
- Controls associated with reactivity management, electrical power alignments, and spent fuel pool cooling
- Implementation of equipment clearance activities

b. Findings

No findings of significance were identified

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed four 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:

- Performance test PT- 450T, EFP-2 Performance Data (including quarterly IST)

Surveillance Test:

- PT-911, Portable Power Independent Pump PPIP-1 Performance Test
- SP-904A, Calibration of ES 4160 Volt ES "A" Bus Undervoltage and Bus Degraded Grid Relays
- SP-300, Operating Surveillance Log (Control Complex and Auxiliary Building Log Readings)

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

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

- January 26, license operator simulator evaluated session, SES-53, involving the loss of SW, plant runback, plant trip, and an RCS leak

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Initiating Events and Mitigating Systems Cornerstones

a. Inspection Scope

The inspectors checked licensee submittals for the PIs listed below for the period January 1, 2009 through December 31, 2009 to verify accuracy. Performance indicator definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 6, were used to check the reporting for each data element. The inspector checked licensee events reports (LERs), operator logs, and daily plant status reports to verify the licensee accurately reported the data including the number of critical hours reported. The inspectors checked that any deficiencies affecting the licensee’s performance indicator program were entered into the corrective action program (CAP) and appropriately resolved.

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Power Changes per 7000 Critical Hours
- Unplanned Scrams with Complications

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Review

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected priority 2A NCR 380670 for a detailed review and discussion with the licensee. The NCR was written to address foreign material (FM) that was identified in the reactor cooling system (RCS), Decay Heat (DH) system and Reactor Vessel (RV). Several NCRs had been written to address the FM located in these different areas as a result of the cutting and welding maintenance activities during the steam generator replacement. The inspectors checked that the issues had been completely and accurately identified in the licensee's corrective action program, safety concerns were properly classified and prioritized for resolution, 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 licensee identified several issues with foreign material exclusion (FME) control with the maintenance activities associated with cutting and welding of the steam generator RCS piping. The licensee found that the contractual vendor performing this maintenance did not adhere to FME procedural requirements as described in MNT-NGGC-007, FME Program. Additionally, licensee management failed to ensure the contracting organization followed the requirements for MNT-NGGC-007. However, with the exception of a couple of examples, most of the FME was identified by the inspections that were completed as part of the planned inspection requirements to complete the maintenance, which included foreign object search and retrieval (FOSAR) inspections. Upon finding FME, the licensee expanded the scope of inspections and was very aggressive in pursuing and retrieving any identified or potential FME. The licensee made a good effort in completing additional inspections and retrieving FME. The inspectors noted there were no identified consequences or system failures as a result of FME.

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

(Closed) LER 05000302/2009-004-00, -01 Main Steam Safety Valve Lift Setpoints OutsideTolerance Longer Than Allowed by Technical Specifications

On September 22, 2009, while performing surveillance procedure SP-650, ASME Code Safety Valves Test, on the 'A' OTSG in Mode 1, the as-found set point for main steam safety valve (MSSV) MSV-34 was found above its required tolerance. The acceptance criteria specified in Improved Technical Specifications (ITS) 3.7.1 is +/- 3 percent of the nominal set point. MSV-34 lifted at +4.3 percent of the nominal set point. The MSSV was declared inoperable and actions associated with ITS 3.7.1 became applicable. The valve was returned to operable status by adjusting its set point to within +/- 1 percent. Two other MSSVs (MSV-36 and MSV-43) scheduled for testing were found to be outside of their required tolerance. MSV-36 lifted at -4.3 percent of its nominal set point while MSV-43 lifted at -5.2 percent of its nominal set point. Both valves were declared inoperable and actions associated with ITS 3.7.1 became applicable. The valves were returned to operable status by adjusting their set points to within +/- 1 percent. The licensee concluded that multiple MSSV's were inoperable for a period longer than allowed by plant ITS. The licensee determined that this ITS violation was a result of a failure, in past years when MSSV's were found out of tolerance, to provide adequate instructions to the vendor refurbishing the valves to determine the root causes of the out of tolerance condition. This lack of adequate vendor instructions was the result of the licensee's failure to follow Corrective Action Program procedures which require that physical evidence and important information that is essential to identifying cause(s) be preserved. Corrective actions included revision of vendor specifications to better control vendor rebuild and testing activities. This performance deficiency was determined to be greater than minor because it was associated with the mitigating system cornerstone attribute of equipment performance, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because the inspectors responded "no" to all questions in the mitigating systems cornerstone column of Table 4a, Manual Chapter 0609, Attachment 0609.04. This licensee-identified finding involved a violation of ITS 3.7.1, Main Steam Safety Valves. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel Activities

a. Inspection Scope

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

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

b. Finding

No findings of significance were identified.

.2 (Closed) Unresolved item (URI) 05000302/2008006-03, Evaluate Opening Access Hatch to Cable Spread Room (CSR)

a. Inspection Scope

The inspectors completed an in-office review and characterization of URI 05000302/2008006-03. The inspectors reviewed the licensee's design and licensing basis for the Crystal River Unit 3 (CR3) main control room (MCR) fire barrier floor and CSR Halon fire extinguishing system, as well as, associated fire protection compensatory actions for these fire protection features when they are out-of-service, degraded, and/or inoperable. This review also included reviews of selected control room log entries; licensee's fire protection calculations for the Halon fire extinguishing system gas concentration, National Fire Protection Association (NFPA) 12A Code compliance documents, and; for the CSR, a review of additional licensee information, a review of fire brigade drill records, and qualitative evaluation of the risk significance. Documents reviewed by the inspectors are listed in the attachment.

b Findings

Introduction: A Green non-cited violation (NCV) of CR3 Operating License Condition 2.C.(9) was identified for failure to take compensatory actions when a MCR to CSR floor/ceiling interface access hatch was open rendering the CSR Halon fire extinguishing system inoperable.

Description: During the triennial fire protection inspection (TFPI) conducted January 28 – February 15, 2008, an URI was identified related to an observed open 30” x 36” access hatch from the MCR floor to the CSR (FZ CC-134-118A) which potentially degraded the CSR Halon fire extinguishing system with no compensatory measures in place. This item was unresolved pending further analysis by the licensee and NRC review. This URI was discussed in NRC Inspection Report No. 05000302/2008006.

The inspectors identified that the licensee failed to recognize that the Halon fire extinguishing system protecting the CSR (FZ CC-134-118A) was rendered inoperable when the MCR to CSR floor/ceiling interface access hatch was open to allow installation of a temporary power cable. The hatch had existed since the beginning of CR3 operating cycles, but had never been recognized as a gas containment barrier whose closure was necessary for the cable spreading room Halon fire extinguishing system to be effective. The open hatch violated the gas containment barrier integrity of the CSR and challenged the ability of the Halon system to achieve the required 5 percent concentration to suppress a fire in the CSR because the Halon gas would be allowed to escape through the open hatch.

Section 2.C.(9) of Crystal River Unit 3 Operating License states, in part, that Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR) for the facility. Table 9.18 of FSAR states in part that the Fire Protection Plan (FPP) has been developed to describe the operational elements of the Fire Protection Program for Crystal River Unit 3 in order to assure response to a fire emergency is timely and adequate. The Plan addresses the Fire Protection Program organization, responsibilities, authorities, administrative controls, automatic and manual fire detection and suppression equipment, procedures, training, operating requirements and limitations, surveillance requirements, and basic design change processes. The Fire Protection Plan is updated periodically, as necessary, in support of changes to the program.

Section 6.3.2, Operability Requirements of Crystal River 3 FPP, Revision 25, states in part that fire suppression systems are required to be operable at all times. However, circumstances do arise, either planned or unplanned, that affect the operability of these systems. In the event that fire suppression systems are inoperable, compensatory measures listed in the applicable table shall be implemented. Table 6.8a states in part that when the Halon System was inoperable, the licensee is required to establish a continuous fire watch with backup fire suppression equipment for the unprotected area within 1 hour. This requirement should have been implemented during each time the hatch was opened. However, the licensee stated that, since this issue was not recognized, compensatory measures were not implemented during periods the MCR to CSR floor/ceiling interface access hatch was open. This issue was in the licensee's corrective action program under CR3 nuclear condition report (NCR) 266356. The inspectors verified that the licensee revised Table 6.8.a of FPP, Revision 26, to include that the hatch to the control room located inside the CSR room, Control Complex 134' El., shall remain closed at all times to ensure the operability of the Halon System.

Analysis: Failure to take compensatory actions when a MCR to CSR floor/ceiling interface access hatch was open rendered the CSR Halon fire extinguishing system inoperable was a performance deficiency. The finding is more than minor because it is associated with the protection against external factors attribute, i.e., fire, and degraded the Mitigating Systems cornerstone objective to ensure the availability of systems that respond to initiating events.

Specifically, the finding adversely affected the suppression fire extinguishing system capability defense-in-depth element. The inspectors evaluated this finding under NRC Inspection Manual Chapter (IMC) 0609, Appendix F, Fire Protection Significance Determination Process (SDP). The inspectors determined that a Phase 2 SDP was required for this finding because the CSR Halon concentration was highly degraded; a fire could occur due to non-qualified cables or transient combustibles while the hatch between the MCR and CSR was open; a duration factor (exposure time) was between 3 and 30 days; and control room operators evacuated the MCR in response to the fire. However, the Phase 2 SDP of IMC 0609 Appendix F does not currently include explicit treatment of fires leading to MCR abandonment, either due to fire in the MCR or due to fires in other fire areas. Therefore, a Phase 3 SDP evaluation for this type of finding was needed.

A Regional Senior Reactor Analyst performed a Phase 3 SDP and concluded that the finding was of very low safety significance (Green). The major assumptions of the evaluation were: (1) The credible ignition sources in the CSR were transient combustibles and Thermo-plastic (self-ignited) cables; (2) There was a limited amount of Thermo-plastic cables in the CSR; (3) Suppression of any type was not credited; (4) Use of the Remote Shutdown Panel was a highly complex, high stress action for the operators for which they had low experience/training and; (5) An exposure time of approximately two weeks. The dominant accident sequence involved a postulated transient fire while the hatch between the MCR and CSR was removed. Control room operators evacuated the MCR in response to the fire. However, they failed to control the plant from the Remote Shutdown Panel and core damage ensued. The inspectors did not identify a cross-cutting aspect associated with this finding because it does not reflect current licensee performance.

Enforcement: Section 2.C.(9) of Crystal River Unit 3 Operating License states in part that Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the FSAR for the facility.

Table 9.18 of FSAR states in part that the FPP has been developed to describe the operational elements of the Fire Protection Program for Crystal River Unit 3 in order to assure response to a fire emergency is timely and adequate.

Section 6.3.2, Operability Requirements of Crystal River 3 FPP, Revision 25, states in part that fire suppression systems are required to be operable at all times. However, circumstances do arise, either planned or unplanned, that affect the operability of these systems. In the event that fire suppression systems are inoperable, compensatory measures listed in the applicable table shall be implemented. Table 6.8a states in part that when the Halon System was inoperable, the licensee is required to establish a continuous fire watch with backup fire suppression equipment for the unprotected area within 1 hour.

Contrary to the above, on January 31, 2008, the licensee failed to recognize that the Halon fire extinguishing system protecting the CSR (FZ CC-134-118A) was rendered inoperable when the MCR to CSR floor/ceiling interface access hatch was open. No compensatory measures were established in accordance with Table 6.8.a of the approved FPP described in Section 2.C.(9) of CR3 Operating License. This condition was not recognized by the licensee until the inspectors' inquiry. The licensee initiated NCR 266356 in the corrective action program to address this issue and revised the FPP. Because this finding is of very low safety significance and was entered into the licensee's corrective action program, this finding is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding is identified as NCV 5000302/2010002-01, Failure to take compensatory actions when a MCR to CSR floor/ceiling interface access hatch was open.

.3 Steam Generator Replacement Project and Containment Wall Repair (IP 50001)

a. Inspection Scope

The inspectors conducted a review of the licensee's Phase 2 preparation and detensioning activities for the repair of the containment wall delamination and reinstallation of the containment wall opening created during the Steam Generator Replacement Project (SGRP) in the last quarter of 2009. The inspectors reviewed Engineering Change (EC) 75218, Reactor Building Repair Phase 2 – Detensioning. The inspectors reviewed Work Package (WP) 1710B, Tendon Detensioning. The inspectors observed that the licensee performed the liftoff testing to check and record 10 horizontal tendon tensile strength in order to know the current tendon strength and prepare for the retensioning later. The inspectors also observed vertical and horizontal tendon detensioning. The inspectors observed and reviewed the records of the acoustic and strain gauges located in the tendon command center to detect the sound volumes and concrete strain changes potentially due to the new cracks or the compressive or tensile stress changes in the concrete during the detensioning. These two detecting devices are used to detect any expansion or additional cracking during the detensioning. The inspectors reviewed the procedures, drawings, calibrations, equipment and personnel qualification, and tendon detensioning communication plan associated with the detensioning. The reviews or observations were conducted in order to verify that the licensee performed activities in accordance with the approved procedures.

The wall delamination repair efforts remained in progress at the end of this inspection period.

b. Findings

No findings of significance were identified.

4OA6 Exit

Exit Meeting Summary

On April 12, 2009, 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.

Improved Technical Specification (ITS) 3.7.1 states that MSSV's shall be operable as specified in ITS Table 3.7.1-1 in Modes 1, 2 and 3. Contrary to the above, on September 22, 2009, while performing SP-650, ASME Code Safety Valves Test, on the 'A' OTSG in Mode 1, the as-found set points of three MSSVs were found outside the ITS 3.7.1 acceptance criteria of +/- 3 percent of the nominal set point. The valves were returned to operable status by adjusting their set point to within +/- 1 percent. The licensee concluded that the three MSSV's were inoperable for a period longer than allowed by plant ITS. The licensee determined that this ITS violation was a result of a failure, in past years when MSSV's were found out of tolerance, to provide adequate instructions to the vendor refurbishing the valves to determine the root causes of the out of tolerance condition. This lack of adequate vendor instructions was the result of the licensee's failure to follow Corrective Action Program procedures which require that physical evidence and important information that is essential to identifying cause(s) be preserved. The finding was evaluated using Phase 1 of the At-Power SDP, and was determined to be of very low safety significance (Green) because the inspectors responded "no" to all questions in the mitigating systems cornerstone column of Table 4a, Manual Chapter 0609, Attachment 0609.04. This issue was documented in the licensee's corrective action program as NCR 356521.

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 Oversight
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
P. Fagan, Repair Design Engineering Supervisor
E. Avella, Manager, Containment Repair

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

LIST OF ITEMS OPENED, CLOSED

Opened and Closed

05000302/2010002-01	NCV	Failure to Take Compensatory Actions When a MCR to CSR Floor/Ceiling Interface Access Hatch Was Open. (Section 4OA5.2)

Closed

05000302/2009004-00, -01	LER	Main Steam Safety Valve Setpoints Outside Required Tolerance Longer Than Allowed By Technical Specifications (Section 4OA3)
05000302/2008006-03	URI	Evaluate Opening Access Hatch to Cable Spread Room (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection

Procedures

AI-2205A, Pre Fire Plan – Control Complex
AI-2205B, Pre Fire Plan – Turbine Building
AI-2205C, Pre Fire Plan – Auxiliary Building
AI -2205F, Pre Fire Plan – Miscellaneous buildings and Components
SP-804, Surveillance of Plant Fire Brigade Equipment
HPP-500, Respiratory Protection Program
HPP-502, Respiratory Equipment Inspection and Maintenance
AP-880, Fire Protection
TRN-NGGC-0010, Fitness-for-Duty, Plant Access, Radiation Worker, and Respiratory Protection Training
WO 1631181, SP-804 completion dated January 17, 2010
HP-502, Monthly completion dated December 11, 2010

Section 1R07: Heat Sink Performance

Procedures

PM-275, General Preventative Maintenance

Work Orders

1631132 SWHE-1C Pick/Shoot and Clean
1648212 DCHE, Shoot and Clean Heat Exchanger

Section 1R12: Maintenance Effectiveness

Nuclear Condition Reports

NCR 346530, DFP-1D did not start during SP-354B
NCR 317302, DFP-1D failed to start during SP-354B

Section 1R20: Refueling and Outage Activities

Procedures

AI-504, Guidelines for Cold Shutdown and Refueling
WCP-102, Outage Risk Management

Section 4OA2: Problem Identification and Resolution

NCR 378182, Foreign Material in RCS near Vessel
NCR 376157, FME Material in DH Drop Line

NCR 376613, RCS Hot Leg FME NCRs Not Timely Issued
NCR 373986, SGR FME Issue Created While Performing FOSAR
NCR 376500, Two Foreign Objects Identified In “A” Hot Leg
NCR 376082, Debris in “A” Hot Leg
NCR 374649, Foreign Material Identified in the RCS
NCR 360111, FME Cavity FME Improvements

<u>Section 4OA5, Other Activities</u>

<u>Section 4OA5.2</u>

<u>Nuclear condition reports</u>

NCR 266356
NCR 264494

<u>Other</u>

Crystal River Unit 3 Operating License
Fire Protection Plan, Revision 25
Fire Protection Plan, Revision 26
Updated Final Safety Analysis Report, Chapter 9
IMC 0609, App F, Fire Protection Significance Determination Process, dated 02/28/05,

<u>Section 4OA5.3</u>

<u>Procedures</u>

PT-0407C, Reactor Building Containment Tendon Detensioning, Retensioning, Replacements, Examinations. and Testing
Precision Surveillance Corporation (PSC) Procedure F & Q 8.1, RAM Tendon Detensioning, Revision 2
PSC Procedure 9.0, Monitor Tendon Force (Liftoff), Revision 0
PSC Procedure 10.0, Calibration of Measuring and Test Equipment, Revision 0
PSC Procedure 10.1, Verification of Calibration Status of Hydraulic Pressure Gauges, Revision 0
Mistras Project R10-215, CR3 Tendon Detensioning Monitoring Procedure, Revision 0

<u>Other</u>

Engineering Change (EC) 75218, Reactor Building Repair Phase 2 – Detensioning
Work Package (WP) 1710B, Tendon Detensioning
Tendon Detensioning Monitoring Communication Plan
CR3 Detension Strain Gauge Graphs and Logs
Acoustic Sound System Graphs and Records