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

April 15, 2004

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

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000302/2004003

Dear Mr. Young:

On March 27, 2004, 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, 2004, with you and members of your staff.

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

Based on the results of this inspection, there was one finding of very low safety significance (Green). Also, a licensee identified violation which was of very low safety significance is listed in Section 4OA7 of this report. If you wish to contest this NCV, 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 Crystal River Unit 3.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Joel T. Munday, Chief
Reactor Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000302/2004003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

cc w/encl:
Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Richard L. Warden
Manager Nuclear Assessment
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Michael J. Annacone
Engineering Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

R. Alexander Glenn
Associate General Counsel (MAC - BT15A)
Florida Power Corporation
Electronic Mail Distribution

Steven R. Carr
Associate General Counsel - Legal Dept.
Progress Energy Service Company, LLC
Electronic Mail Distribution

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, FL 32304

William A. Passetti
Bureau of Radiation Control
Department of Health
Electronic Mail Distribution

Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
Electronic Mail Distribution

Chairman
Board of County Commissioners
Citrus County
110 N. Apopka Avenue
Inverness, FL 36250

Jim Mallay
Framatome Technologies
Electronic Mail Distribution

Distribution w/encl: (See page 4)

Distribution w/encl:
B. Mozafari, NRR
C. Evans (Part 72 Only)
L. Slack, RII EICS
RIDSNRRDIPMLIPB
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2004003
Licensee:	Florida Power Corporation
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	December 28, 2003 to March 27, 2004
Inspectors:	S. Stewart, Senior Resident Inspector R. Reyes, Resident Inspector George T. Hopper, Senior Reactor Inspector (1R11.1) Larry Mellen, Senior Reactor Inspector (1R11.1, 1EP4) P. Van Doorn, Senior Reactor Inspector (1R07)
Approved by:	Joel T. Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2004-003; 12/28/2003 - 03/27/2004; Crystal River Unit 3; Event Followup

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

A. <u>NRC Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

<u>Green</u>: A self-revealing Green finding was identified for a loss of design control during maintenance on the integrated control system which later resulted in a reactor trip.

This finding is more than minor because it affected the configuration control attribute of the initiating event cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. The issue was of very low safety significance because although there was a reactor trip, mitigating systems remained available and were not affected. Because no safety systems were affected, the finding did not involve a violation of regulatory requirements. The cause of the finding involved the cross-cutting element of human performance. (Section 4OA3.1)

B. <u>Licensee Identified Violations</u>

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 operated at full power during the inspection period with the following exceptions: The plant was shutdown on March 3 to repair a small reactor coolant pump oil leak. The plant restarted and returned to power operation on March 4, 2004. On March 24, the reactor tripped from full power due to a feedwater transient. Repairs were made and the plant returned to power operation on March 25, 2004.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [Reactor-R]

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's plans 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. The inspectors walked down portions of the fire pump building and the boric acid storage tank area to check for any unidentified susceptibilities. Operability of heat trace circuits for boric acid storage tank piping was checked. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting cold weather protection issues. During periods when outdoor temperature fell below 40 degrees Fahrenheit (F), the inspectors verified that the licensee implemented the administrative instructions to mitigate any effects from cold weather. There were no periods of freezing weather during the inspection period.

The inspectors monitored the licensee's implementation of EM-220 on February 24, 2004, when the licensee was notified of a tornado watch for the local area. The licensee activities were checked to assure that vital systems and components were protected from severe weather in accordance with licensee Emergency Instruction EM-220, Violent Weather. The inspectors walked down portions of the following systems/areas to verify the licensee's mitigation strategy.

- 230 kV Switchyard
- Emergency Feedwater Pump 3 building

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment Walkdowns

a. Inspection Scope

The inspectors performed the following partial system walkdowns during this inspection period. The inspectors reviewed the alignment of the selected risk-significant systems to evaluate the readiness of the redundant trains while one train was out of service for maintenance. The inspectors checked switch and valve positions using the alignments specified in the listed operating procedures and checked electrical power alignment to critical components. The inspectors reviewed applicable sections of the Crystal River 3 Final Safety Analysis Report to obtain design and operating requirements. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting component alignment issues.

- On January 21, Plant Electrical Lineup in accordance with Operating Procedure OP-700A, 6900, 4160, and 480 Volt AC Buses when diesel generator MTDG-1 was removed from service for preventive maintenance per WO-364717 and WO-364718

- On February 9 and 10, the Emergency Diesel Generator EGDG-1A, using Operating Procedure OP-707, Operation Of The ES Diesel Generators, while the EGDG-1B was out of service for maintenance.

- On February 22, the Emergency Diesel Generator EGDG-1B, using Operating Procedure OP-707, Operation Of The ES Diesel Generators, while the EGDG-1A was out of service for maintenance

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown: On January 13 and 14, the inspectors conducted a detailed review of the alignment and condition of the operable low pressure injection system. Additionally, the reactor building sump was walked down on March 3, 2004. The inspectors used the following plant procedures, including the operating procedures (OP) and surveillance procedures (SP) listed below, as well as Final Safety Analysis Report (FSAR) Chapter 6.1.2.1.2, Low Pressure Injection to verify proper system alignment:

- OP-404 Decay Heat Removal System
- SP-412 ECCS and Containment Spray Leak Test (A and B Train)
- SP-435 Valve Testing During Cold Shutdown (LPI Valves Only)
- SP-603 Decay Heat Check Valve Leak Testing

The inspectors verified selected electrical power requirements, labeling, hangers and support installation, and associated support systems status. The walk downs also included evaluation of selected system piping and supports against the following considerations:

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

A review of outstanding maintenance work orders was performed to verify that any deficiencies did not significantly affect the system function. In addition, the inspectors reviewed the open nuclear condition reports (NCRs) to verify that system problems were being identified and appropriately resolved. The System Health Report July to December 2003, for the Decay Heat System was specifically reviewed as was pump and valve data from ASME Section XI testing.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down the following risk-significant plant areas to verify that control of transient combustibles and ignition sources were consistent with the licensee's Fire Protection Plan and 10 CFR Part 50, Appendix R. The inspectors also evaluated the material condition, operational lineup, and operational effectiveness of fire protection systems and assessed the condition of fire barriers used to contain fire damage. The inspections were completed using the standards of the Crystal River Fire Protection Plan; 10 CFR Part 50, Appendix R; the Florida Power Corporation Analysis of Safe Shutdown Equipment; and the Final Safety Analysis Report. The inspectors reviewed sections of OP-880, Fire Service System, and checked performance of SP-800, Monthly Fire Extinguisher Inspection, to monitor the operational condition of fire protection equipment. When applicable, the inspectors checked that compensatory measures for fire system problems were implemented. The inspectors observed performance of weekly fire alarm checks done in accordance with surveillance procedure SP-323, Evacuation and Fire Alarm Demonstration.

- Control Rod Drive Power Supply Area
- 95-Foot Penetration area (locked triangle room)
- 'A' and 'B' Emergency Diesel Generator Rooms
- A Decay Heat and Reactor Building Spray Vault
- B Decay Heat and Reactor Building Spray Vault
- EFP-3 Building

- Unit 3 Main Transformer, Auxiliary Transformer, Start-up Transformer, and Backup ES Transformer Area
- 95-Foot Intermediate Building
- Main Control Room
- Chill Water Pump and Control Complex Ventilation Area

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

On February 11, 2004, the inspectors observed the licensee fire brigade respond to a simulated fire on the “B” feedwater pump on the turbine deck. The inspectors checked the brigade’s communications, ability to set-up and execute fire operations, and their use of fire fighting equipment. The inspectors attended the post-drill critique to check that the licensee’s drill acceptance criteria were met and that any discrepancies were discussed and resolved. In addition to the drill observation, Administrative Instruction AI-2205, Fire Drill Planning and Evaluation Reports, dated January 2, February 2, February 4, February 11, February 16, and February 26, 2004, were checked to assure that acceptance criteria were evaluated and deficiencies were documented and remediated, as appropriate, as various fire response crews were evaluated.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed inspection records, work documents, preventive maintenance procedures, and other documentation to ensure that heat exchanger deficiencies that could mask or degrade performance were identified. Inspection records for risk significant heat exchangers were reviewed which included performance for the two heat exchangers on Decay Heat Closed Cycle Cooling (DC) and four heat exchangers on Nuclear Services Closed Cycle Cooling (SW). The inspectors also reviewed general health of the DC, SW, and Nuclear Services and Decay Heat Seawater (RW) systems via review of inspection/test results; review of chemistry activities; review of DC, RW and SW corrective maintenance history; review of current system health reports; and discussions with the system engineer. Selected Nuclear Condition Reports (NCRs) were reviewed for potential common cause problems and problems which could affect system performance to confirm the licensee was entering problems into the corrective action program and initiating appropriate corrective actions. In addition, the inspectors conducted a walk down of most of the heat sink systems and major components.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Biannual Inspection

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of January 26-30, 2004, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of simulator operating tests and job performance measures (JPMs) associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55, "Operators' Licenses." 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 reviewed and evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations. The inspectors observed three crews during the performance of the operating tests. Documentation reviewed included written examinations, JPMs, simulator scenarios, licensee procedures, on-shift records, licensed operator qualification records, watchstanding and medical records, simulator modification request records and performance test records, the feedback process, and remediation plans. The records were inspected against the criteria listed NRC Inspection Procedure 71111.11. Documents reviewed during the inspection are listed in the Attachment.

Following the completion of the annual operating examination testing cycle which ended on February 13, 2004, the inspectors reviewed the overall pass/fail results of the individual JPM operating tests, and the simulator operating tests administered by the licensee during the operator licensing requalification cycle. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

.2 Resident Inspector Quarterly Review

a. Inspection Scope

On January 15, the inspectors observed licensed operator response and actions on the simulator to Simulator Evaluated Session, SES-36. The session, which was part of the licensed operator's annual examination, required the crew to use plant procedures to establish high pressure injection system operations from a forced circulation - steam dump condition, shift decay heat removal system loops of operations, and a respond to a complete loss of decay heat removal. The inspectors specifically evaluated the following attributes related to operating crew performance.

- Clarity and formality of communication including crew briefings
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms including a loss of decay heat removal pump alarm
- Implementation of Emergency Operating Procedures, including EOP-2, Vital System Status Verification, EOP-4, Inadequate Heat Transfer
- Correct use and implementation of procedures AP-404, Loss of Decay Heat Removal, and OP-404, Decay Heat Removal System Operation, and Abnormal Procedure AP-770, Emergency Diesel Generator Actuation
- Control board operation and manipulation, including operator actions such as establishing decay heat removal system operation from a two reactor coolant pump operation configuration
- Oversight and direction provided by supervision, including ability to identify and implement appropriate technical specification actions, event classification, and notification of state authorities within the 15 minute requirement
- Effectiveness of the training oversight, evaluation, and critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the planned maintenance activities listed below to evaluate the licensee's implementation of the maintenance rule (10CFR50.65). The inspectors checked that licensee personnel monitored unavailability of equipment important to safety and trended key performance parameters. For the equipment issues described in the work orders (WO) listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10CFR50.65) with respect to the characterization of failures, the appropriateness of the associated a(1) or a(2) classifications, and the appropriateness of either the associated a(2) performance criteria or the associated a(1) goals and corrective actions. The inspectors checked if the licensee maintained safety functions when equipment important to safety was removed from service for maintenance. The inspectors also periodically reviewed the licensee's implementation of 10 CFR 50, Appendix B and technical specification requirements regarding safety system problems. The inspectors routinely checked that the licensee promptly entered problems with plant equipment into the corrective action program or the corrective maintenance program. The inspectors checked that the licensee monitored work practices and when appropriate, documented work problems in the corrective action program. The licensee's System Health Reports, July to December 2003, were routinely reviewed to check that problems were being documented and resolved and that industry information was being used in system assessments. Licensee electronic maintenance rule data and evaluation criteria were reviewed as part of these inspections.

- NCR 114453, Makeup Valve MUV-544 Failed To Stroke During Surveillance Test SP-370
- NCR 110023, Incorrect Parts Installed in Integrated Control System
- NCR 106443, Reactor Coolant System Pressure Boundary Leakage at the Upper Level Tap for Pressurizer Level Instrument RC-1-LT3 (Licensee Engineering Letter SE03-0110, RC System Maintenance Rule Status for 2003 was reviewed in this inspection)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following risk assessments to assess the effectiveness of licensee's risk evaluation for maintenance and testing. The inspectors reviewed daily maintenance schedules and observed work controls to check risk management while maintenance was conducted. The inspectors assessed operability of equipment using technical specifications, the Final Safety Analysis Report, licensee procedures, and regulatory information such as NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded And Nonconforming Conditions. The inspectors also reviewed maintenance schedules to check that overall risk was minimized through preservation of safety functions such as decay heat removal capability, reactor coolant system inventory control, electric power availability, reactivity control, and primary containment control. The inspectors checked if licensee personnel were managing risk by assuring that key safety functions were preserved and that upon identification of an unplanned situation, the resulting emergent work was evaluated by the licensee for risk and controlled as described in technical specifications, licensee Compliance Procedure CP-253, Power Operations Risk Assessment and Management, and licensee Administrative Instruction AI-500, Conduct of Operations. The inspectors checked that risk significant emergent work was documented in the corrective action program and that risk management actions were promptly initiated.

- Work Week 04W01, Risk assessment for planned preventive maintenance on Emergency Feedwater Pump 1 revised for an expanded clearance boundary for auxiliary steam valve ASV-144 that removed auxiliary steam supply from Units 1 and 2 (Work Order 474046)

- Work Week 04W02 for testing of emergency diesel generator 1B, revised when the cooler for instrument air pump IAP-3C was removed from service to unclog the heat exchanger evaporative cooler supply line (Clearance 64379)

- Work Week 04W05, Risk assessment for planned maintenance on Building Spray Pump BSP-1A (Work Order 479038) revised for an extended clearance boundary for Feedwater Pump FWP-7 when the pump remained out of service beyond the expected time due to foreign material concerns (NCR 116804)

- Work Week 04W06, Risk assessment for planned overhaul of emergency diesel generator EDG-1B, accomplished in accordance with NRC License Amendment Number 207, Dated June 13, 2003

- Work Week 04W08 Risk Assessment for overhaul of emergency diesel generator EDG-1A under clearance 65389 revised when auxiliary steam from Units 1 and 2 was removed from service for emergent maintenance on Unit 1.

b. <u>Findings</u>

No findings of significance were identified.

1R14 <u>Personnel Performance During Non-routine Plant Evolutions</u>

a. <u>Inspection Scope</u>

For the non-routine events described below, the inspectors either observed the activity or reviewed operator logs and computer data to determine that the evolution was conducted safely and in accordance with plant procedures. Specific checks were done to assess operator performance in coping with non-routine events and transients.

- Licensee Event Report (LER) 05000302/2003-005-01, Reactor Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Pump Control Problems. The LER identified human performance issues and the inspector verified completion of various corrective actions to address the performance problems. Further review of this event is discussed in Section 4OA3.1 of this report.

- Reactor startup and return to power operation on March 4, 2003 following shutdown to Mode 3 on March 3 to repair a small oil leak in a reactor coolant pump.

- Control room operations during entry into abnormal procedure AP-520, Loss Of RCS Coolant or Pressure, on March 12, 2004. The licensee entered the abnormal procedure as a result of increased reactor coolant leakage which occurred during valve alignments using operating procedure OP-402, Makeup And Purification System.

- Reactor trip on March 24, 2004.

- Reactor startup and return to power operation on March 25, 2004.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following degraded or nonconforming conditions to determine if operability of systems or components important to safety was consistent with technical specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, and when applicable, NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions. The inspectors monitored licensee nuclear condition reports (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-200, Corrective Action Program. The inspectors checked that when plant problems were identified, the resulting change in plant risk was identified and managed. The following issues, including the related nuclear condition reports (NCRs), were specifically checked:

- NCR 117674, EGDG-1B Degraded Water Jacket Cooling Thermostatic Valve Element

- NCR107952, Unqualified Coatings Found in the Reactor Building. Licensee Calculation S89-0050: Allowable Quantities of [Reactor Building] Coatings Failures and Surveillance Procedure SP-324, Containment Inspection, completed November 4, 2003, were reviewed by the inspector.

- NCR 119442 Service Water Heat Exchangers Swapped out of Sequence (Applicable portions of operating procedure OP-408, Nuclear Services Cooling System were reviewed)

- NCR 117598 Emergency Diesel Generator EGDG-1B Air Receiver Tank Internal Inspection Failed Acceptance Criteria. (Licensee Quality Assurance Report

2004-008, Digital Ultrasonic Thickness NDE Report: EGT-2A and EGT-2B, EGDG-1B Air Receiver Tank Lower Heads, was reviewed by the inspectors.)

- NCR 121257, Small Leak Observed at RWV-130 Weld on Outlet Side

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

On March 8, 2004, there were two operator workarounds (OWA) listed in the licensee's OWA list. The inspector reviewed the operations activity and the nuclear condition report associated with the OWAs. Corrective actions addressing the OWA were reviewed. The inspectors reviewed the degraded annunciator log, the degraded equipment list, and a recent operational communication relating to the RCP-1A oil leak, to determine if there were any operator workarounds which had not been identified by the licensee. Selected degraded equipment and lit annunciators were discussed with non-licensed operators, control room operators, and control room supervisors to check if issues should be classified as workarounds.

- Integrated Control System - Feedwater flow oscillates when controlling average reactor temperature

- Condensate System - Condensate valve CDV-39 will not control in automatic mode

Cumulative Effects

The inspectors performed a semi-annual evaluation of the potential cumulative effects of all outstanding OWAs. At the time of the inspection, there were two OWAs. The inspectors evaluated these OWAs along with issues on the degraded equipment log for their cumulative effects, and discussed these potential effects with control room supervisors and operators. The inspectors reviewed the equipment out-of-service logs and walked down the control room and plant areas to verify OWAs were being identified and properly entered into the corrective action program.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors observed or reviewed the following post-maintenance testing activities for risk significant systems to check the following (as applicable): (1) the effect of testing on the plant had been adequately addressed; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and demonstrated operational readiness; (4) test instrumentation was appropriate; (5) tests were performed as written; and (6) equipment was returned to its operational status following testing. The inspectors evaluated the licensee activities using the technical specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications. Final Safety Analysis Report Section 14.1.2.9 Station Blackout Accident, was specifically reviewed. The inspectors routinely checked that post maintenance testing issues were resolved in the licensee's corrective action program.

- Load Test of Battery Charger DPBC-1A per WO 414152-05 (Surveillance Procedure SP-522, Battery Charger Load Test) following preventive maintenance per WO 412342-01 (PM-141, Battery Charger Preventive Maintenance and Setpoint Adjustments)

- Surveillance Procedure SP-348A, Auxiliary Feedwater Pump AFW-7 Testing following Engineering Change 55555RO, Install Suction Side Screen on FWP-7

- Surveillance Procedure SP-348A Auxiliary Feedwater Pump FWP-7 and MTDG-1 Surveillance Testing following preventive maintenance on the backup power supply diesel (MTDG-1) using Work Orders WO-364717 and WO-364718, Preventive Maintenance Checks of MTDG-2A and 2B engines, respectively.

- Surveillance Procedure SP-354B, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1B, following diesel engine overhaul per WO 237895-01. A maintenance run of the engine only, conducted as part or the overhaul, was also reviewed.

- Operating Procedure OP-707, Operation of the Engineered Safeguards Emergency Diesel Generator and Surveillance Procedure SP-354A, Monthly Functional Test of the Emergency Diesel Generator EGDG-1A, after Overhaul under Clearance Number 65389. A maintenance run of the engine only, conducted as part or the overhaul, was also reviewed.

- Surveillance Procedure SP-300, Operating Daily Surveillance Log, (RCP section) after sealing the local oil cap and tightening fittings in the lube oil system on the Reactor Cooling Pump 1A, per WO number 490365.

- Surveillance Procedure SP-353, Control Room Emergency Ventilation System And RM-A5 Monthly Test, after trouble shooting RM-A5 and correcting a contact per WO number 529033.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors checked the following surveillance tests for risk-significant systems or components, to assess compliance with Technical Specifications, 10 CFR Part 50, Appendix B, and licensee surveillance procedure (SP) requirements. The testing was also checked for consistency with the Final Safety Analysis Report. The inspectors checked if the testing demonstrated that the systems were ready to perform their intended safety functions. During the inspections, the inspectors verified that licensee personnel were documenting surveillance problems in the corrective action program in accordance with 10 CFR Part 50, Appendix B, Criterion XVI, and licensee procedure CAP-NGGC-200, Corrective Action Program.

Inservice test (IST) activities were reviewed to ensure testing methods, acceptance criteria, and corrective actions were in accordance with the ASME Code, Section XI, and Florida Power Corporation ASME Section XI, Ten Year Inservice Testing Program, dated May 4, 1998.

- SP-370, Quarterly Testing of Valves, MUV-544 on January 3, 2004 (IST)
- SP-108, Reactor Trip Module and Control Rod Drive Trip Functional Test on Jan 12, 2004
- SP-349B, Emergency Feedwater Pump EFP-2 and Valve Surveillance Test on January 21, 2004 (IST)
- SP-365C, Diesel Fire Service Pump, FSP-2B Operability, on February 12, 2004
- SP-344B, RWP-2B, SWP-1B and Valve Surveillance, (RWP-2B only) on February 19, 2004
- SP-354B, Monthly Functional Test Of The Emergency Diesel Generator EDG-1B on March 11, 2004

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification listed below to ensure that it did not adversely affect the operation of the system. The inspectors screened temporary plant modifications for systems that were ranked high in risk for departures from design basis and for inadvertent changes that could challenge the systems to fulfill their safety function. The inspectors conducted plant tours and discussed system status with engineering and operations personnel to check for the existence of temporary modifications that had not been appropriately identified and evaluated.

- Engineering Change 55555RO, Feed Water Pump - 7 Suction Screen Addition

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed the changes made to Revision 24, dated February 27, 2004, of the Radiological Emergency Response Plan (RERP) against the requirements of 10 CFR 50.54(q) to determine whether any of the changes decreased the effectiveness of the RERP.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

On March 23, 2004, the inspectors observed the licensee in a simulator based emergency preparedness drill. Results of the drill are used by the licensee as inputs into the Drill/Exercise Performance and Emergency Response Organization Drill Participation Performance Indicators. During the scenario which included staffing of the technical support center and the emergency offsite facility, 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 checked 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.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Initiating Event and Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors checked licensee submittals for the performance indicators (PIs) listed below for the period January 1, through December 31, 2003,to verify the accuracy of the PI data reported during that period. Performance indicator definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 2, were used to check the reporting for each data element. The inspector checked licensee event reports (LERs), operator logs, daily plant status reports, nuclear condition reports (NCRs), and performance indicator data sheets to verify that the licensee had identified the cumulative safety system unavailability. The inspectors also checked the accuracy of the number of required hours reported. In addition, the inspectors interviewed licensee personnel associated with performance indicator data collection, evaluation, and distribution. The inspectors checked that deficiencies affecting the licensee’s performance indicator program were entered into the corrective action program and appropriately resolved.

Reactor Safety Cornerstone

- Safety System Unavailability, Residual Heat Removal System
- Safety System Unavailability, Heat Removal (AFW)

a. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Routine Problem Review

a. Inspection Scope

The inspectors selected the following nuclear condition report (NCR) for detailed review and discussion with the licensee. This report was examined to verify whether problem identification was timely, complete and accurate; safety concerns were properly classified and prioritized for resolution; technical issues were evaluated and dispositioned to address operability and reportability; root cause or apparent cause determinations were sufficiently thorough; extent of condition, generic implications, common causes, and previous history were adequately considered; and appropriate corrective actions were implemented or planned in a manner consistent with safety and technical specification compliance. The inspectors evaluated the report against the requirements of the licensee's corrective action program in Administrative Procedures CAP-NGGC-0200, "Corrective Action Program" and 10 CFR 50, Appendix B. Engineering Calculations M03-0006, Reactor Coolant System Leakage Calculations and M02-0004, SP-317 Reactor Coolant System Water Inventory Balance Correction Factors were reviewed. The July to December 2003 System Health Report regarding Reactor Coolant System, and the licensee's (10 CFR 50.65) maintenance rule event data base were also reviewed.

- Nuclear Condition Report 69214, Increasing Count Rate on Radiation Monitor RM-A6

b. Findings and Observations

There were no significant licensee performance issues or NRC violations identified by the inspectors regarding these condition reports. The inspectors verified that the apparent cause evaluation and corrective actions were appropriate and timely, commensurate with the safety significance of the problem.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000302/2003-005-00, Reactor Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Pump Control Problems

(Closed) Licensee Event Report (LER) 05000302/2003-005-01, Reactor Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Pump Control Problems

a. Inspection Scope

The inspectors reviewed both revisions of the LER to evaluate the licensee's assessment of the event and to identify any licensee performance deficiencies associated with the cause.

b. Findings

Introduction: A self-revealing Green finding was identified for a loss of design control of the integrated control system. An improper control card placed in the integrated control system caused a feedwater transient when the B main feedwater pump was placed in automatic control during power ascension. The transient resulted in a reactor trip on high reactor coolant system pressure.

Description: On November 5, 2003, the reactor tripped on high pressure after a feedwater transient occurred when the B main feedwater pump was placed in automatic control. The licensee identified the cause of the event was human error associated with not verifying that the correct circuit control cards were being installed in the integrated control system during preventive maintenance. Contributing causes included the identification of good circuit cards as defective during testing and not completing proper verifications when ordering replacement cards. The inspector reviewed the licensee's corrective action document including the root cause report and interviewed involved personnel. The inspector found that the personnel involved in the maintenance activity did not follow station procedures for verification of replacement parts, including verifying part numbers and doing a visual, like-for-like check. In three cases, wrong circuit cards were placed in the control system and when the B feedwater pump was placed in automatic operation during power ascension, a feedwater transient resulted that caused a reactor trip on high pressure. Safety systems responded as designed and feedwater remained available in manual operating mode to recover steam generator level and maintain safety. Station procedure ADM-NGGC-0106, Configuration Management Program Implementation, Step 9.2.12, required "Each replacement shall be identical (like-for-like) in material, function, and quality requirements unless the differences are reconciled and documented". The replacement cards were not checked like-for-like and a difference, the lack of a time delay feature was not reconciled prior to replacement. Failure of licensee personnel to follow configuration management procedures was a performance deficiency. Also, the failure to properly conduct the maintenance activity was a human performance error.

<u>Analysis:</u> The performance deficiency was more than minor because it affected the design control attribute of the Initiating Events cornerstone and resulted in an event that upset plant stability and challenged critical safety function. No mitigating systems were affected and they remained available. The maintenance was done while the plant was shutdown and the vulnerability for reactor trip existed with plant startup on November 5, 2003. Using the Significance Determination Process Phase 1 Screening Worksheet the performance deficiency was identified as a Transient initiator contributor in the Initiating Events Cornerstone. Because only the Initiating Event cornerstone was affected and no mitigating equipment was degraded, the inspectors found the occurrence to screen as Green. A Phase 2 assessment was not required. The degraded condition existed for one day and was not applicable when the feedwater system was shutdown.

<u>Enforcement:</u> Because the equipment affected by the performance deficiency was non-safety related, no violation of regulatory requirements occurred. The licensee replaced the improper cards with those of the correct design, did an extent of condition review, and initiated a high level root cause evaluation. The finding is identified as FIN 05000302/2004003-01, Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip. The LER is closed.

.2 <u>Automatic Reactor Trip</u>

a. <u>Inspection Scope</u>

On March 24, 2004, the Unit 3 reactor automatically tripped from 100% power due to an integrated control system transient. The problem was initiated by a malfunctioning reactor demand control circuit, which caused a feedwater transient. The AMSAC (Anticipated-Transient-Without-Scram Mitigating System Actuating Circuit) sensed the problem and responded as designed to trip the main turbine, which initiated a reactor trip. The inspectors responded to the control room and verified the unit was stable in Mode 3, and confirmed that all safety-related mitigating systems had operated properly. The inspectors examined operator and plant response by reviewing plant parameters, strip charts, operator logs, and discussed the event with operations personnel and members of the licensee's Event Review Team. The inspectors verified that appropriate notifications were made in accordance with 10 CFR 50.72. Furthermore, the inspectors reviewed the post-trip report and attended the Plant Nuclear Safety Committee meeting prior to restart.

4OA4 Cross Cutting Issues

Section 4OA3.1 describes a human performance error where the licensee failed to implement their procedures for design control during maintenance. As a result, a feedwater system transient occurred during power ascension which caused a reactor trip.

4OA5 Other Activities

.1 NRC Temporary Instruction (TI) 2515/153, "Reactor Containment Sump Blockage (NRC Bulletin 2003-01)"

a. Inspection Scope

The inspectors reviewed the Crystal River Nuclear Plant response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors." The inspectors verified that the compensatory measures described in the licensee's response had been implemented. The inspectors interviewed operations, engineering, and training personnel regarding sump design, site-specific risk implications of sump screen clogging, and verification that compensatory measures to minimize sump clogging vulnerabilities had been taken. The inspectors reviewed training records, procedures, and containment inspection documentation to assure that the licensee's schedule of mitigating actions was commensurate with risk. The inspectors reviewed Plant Nuclear Safety Committee Meeting number 2003-19 minutes which summarized the management review of the interim compensatory measures taken by the licensee. Finally, the inspectors completed a walkdown of the containment including the containment sump area to assess cleanliness and physical condition of equipment. The specific compensatory measures verified included:

- Licensed operators were trained in both classroom and simulator sessions on identification and mitigation of sump screen blockage. Crystal River 3 Lesson OPS 5-1028, Reactor Building Sump Clogging and Mitigation and Simulator Exercise LOR-2-07, ECCS Suction Transfer and Reactor Building Sump Screen Blockage Demonstration were used in the training.

- Emergency Management Procedures EM-225, Technical Support Guidance, and EM-225E, Guidelines for Long Term Cooling, were revised to provide guidance to emergency management personnel on mitigation of reactor building sump blockage events. Personnel assigned to technical support duties were trained on the revisions.

- Surveillance Procedure SP-324, Containment Inspection, was implemented at the end of refueling outage 13. The surveillance assured the absence of loose debris which could be carried to the containment sump and that post-LOCA flow paths were unobstructed. Discrepancies, such as scaffold material that had remained in the reactor building, were evaluated by engineering and determined to be acceptable prior to reactor operation.

- Surveillance Procedure SP-175, Containment Sump Level and Flood Monitoring System Calibration, was revised to include a specific check for sump screen gaps and breaches.

- Administrative Instruction AI-607, Pre-Job and Post-Job Briefings, was revised to add a precaution for foreign materials control when working in the reactor building.

- A debris source inventory was generated from detailed walkdowns inside the reactor building during refueling outage 13. The walkdown estimated the quantity of debris, including thermal insulation, fireproofing materials, and adhesive labels in the reactor building for input in a sump operability assessment.

b. Findings

There were no findings. The licensee had implemented compensatory measures as described in their response to NRC Bulletin 2003-01. The licensee initiated an engineering evaluation of the emergency core cooling recirculation function.

.2 NRC Temporary Instruction (TI) 2515/154, “Spent Fuel Material Control and Accounting at Nuclear Power Plants

Temporary Instruction 2515/154 Spent Fuel Material Control and Accounting at Nuclear Power Plant, Phase I and Phase II were completed during this inspection period. Appropriate documentation was provided to NRC management as required.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

The resident inspectors presented the inspection results to Mr. Young and other members of licensee management at the conclusion of the inspection on April 12, 2004. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. The licensee did not identify any proprietary information.

.2 Annual Assessment Meeting Summary

On April 13, 2004, the NRC's Chief of Reactor Projects Branch 3, Region II Public Affairs Officer, and Resident staff assigned to the Crystal River Nuclear Plant met with Progress Energy - Florida Power Corporation (FPC) to discuss the NRC's Reactor Oversight Process (ROP) and the Crystal River annual assessment of safety performance for the period of January 1, 2003 - December 31, 2003. The major topics addressed were: the NRC's assessment program and the results of the Crystal River assessment. Attendees included FPC management, FPC site staff, and one local TV reporter.

This meeting was open to the public. The NRC's presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML041050623. Licensee's handout was presented at the meeting is also available from the NRC's document system (ADAMS) as accession number ML041050617. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

40A7 Licensee-Identified Violation

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

10 CFR 50.74 requires in part that each licensee shall notify the commission in accordance with Section 50.4 within 30 days of the following in regard to a licensed operator or senior operator ... (c) permanent disability or illness as described in 10 CFR 55.25 of this chapter. Contrary to this, on March 10, 2003 a licensed operator had a change in medical condition as described in ANSI/ANS 3.4-1983, that was not reported to the commission within 30 days. This finding was identified by the licensee during an audit of medical records in July 2003. The NRC was notified of this finding in a letter dated August 27, 2003. The operator's license was amended on October 10, 2003 to require corrective lenses while performing licensed duties. The finding is of very low safety significance because there was no evidence that the licensee endangered plant operations as a result of impaired visual acuity while performing licensed duties after the change in medical condition occurred. This issue is documented in Nuclear Condition Report Number 100513.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

J. Huegel, Manager, Operations
W. Brewer, Manager, Maintenance
R. Davis, Manager, Training
J. Franke, Plant General Manager
J. Kreuhm, Manager, Work Controls and Outage
D. Roderick, Director Site Operations
D. Hanna, Supervisor, Self Evaluation and Emergency Preparedness
S. Powell, Supervisor, Licensing
M. Rigsby, Radiation Protection Manager
M. Annacone, Manager, Engineering
R. Warden, Manager, Nuclear Assessment
D. Young, Vice President, Crystal River Nuclear Plant
S. Young, Security Manager

NRC personnel:

J. Munday. Chief, Reactor Projects Branch 3, NRC Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-302/2004003-01	FIN	Loss of Design Control When an Improper Circuit Card Placed in the Integrated Control System Caused a Reactor Trip (Section 4OA3.1)

Closed

50-302/03-005-00	LER	Reactor Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Pump Control Problems (Section 4OA3.1)
50-302/03-005-01	LER	Reactor Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Pump Control Problems (Section 4OA3.1)

Discussed

TI 2515/153	Reactor Containment Sump Blockage (NRC Bulletin 2003-01) (Section 4OA5.1)
TI 2515/154	Spent Fuel Material Control and Accounting at Nuclear Power Plants (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

TPP-200, Licenced Operator Continuing Training Program, Rev 4
TTP-206, Simulator Program, Rev 3
TAP-409, Conduct of Simulator Training and Evaluation, Rev 7
TAP-410, NRC License Examination Security Program, Rev 3
TAP-412, Simulator Operation, Rev 1
TAP-422, Simulator Maintenance, Rev 0
NGGS-TRN-0002, Individual Performance Report, Revision 1
Form 422.1, Simulator Change Record, Revision 0
Form 206.3, Simulator Change Record, Revision 2
SOI-11, Attachment 1, Simulator Trouble Report
R02BIEN1 - 2002 RO Biennial Test #1
R02BIEN2 - 2002 RO Biennial Test #2
R02BIEN3 - 2002 RO Biennial Test #3
S02BIEN1 - 2002 SRO Biennial Test #1
S02BIEN2 - 2002 SRO Biennial Test #2
S02BIEN3 - 2002 SRO Biennial Test #3

simulator tests reviewed:
PTC1, Computer Real Time and Repeatability Test
PTS1, Steady State Performance Test
PTT1, Reactor Trip
PTT2, Total Loss of Feedwater
PTT3, Simultaneous Loss of all MSIVs
PTT4, Loss of Offsite Power from Full Power
PTT5, Reactor Coolant Pump Trip
PTT6, Main Turbine Trip
PTT7, Maximum Power Ramp
PTT8, Large Break LOCA (DBA)
PTT9, Main Steam Line Break Inside Containment
PTT10, PORV Failure With HPI Inoperable
PTT11, Maximum Design Load Rejection

<u>Maintenance Work Order Documents (1RO7)</u>

379841, SWHE-1A Cleaning dated 05/20/2003
399182, SWHE-1B Cleaning dated 06/30/2003
417944, SWHE01C Cleaning dated 06/06/2003
452330, SWHE-1D Cleaning, dated 11/17/2003
409401, DCHE-1A Cleaning dated 07/21/2003
372955, DCHE-1B Cleaning dated 06/30/2003
230943, B Flume and Pit Inspection dated 07/25/2002
305024, B Flume and Pit Inspection dated 01/02/2003
278292, RW Liner Inspection dated 08/15/2003
278288, DC Liner Inspection dated 07/22/2003

Completed Procedures (1R07)

PT-136A, SW System Flow Balance, Rev. 0 completed 04/16/96
PT-136B, DC System Flow Balance and EGDG KW Loading, Rev. 0 completed 11/13/1995

Nonconformance Reports (1R07)

55642, Pinhole leak in RW flush piping
57896, SWHE-1A cathodic protection fell off
58559, Supports RWH-36 and RWH-73 not fully qualified
58934, SWHE-1D tube leak
62079, RW thermal relief valves unreliable
62228, DCP-1A unavailability longer than expected
64451, RWP-1 and 3A vibration unacceptable
66864, Unavailability for B DC projected to be exceeded
67496, DHV-35 increased thrust values
67948, SWHE-1C found 83% blocked
69477, Contamination found inside RW piping
68576, RWP-3A vibration in Alert
78462, RWP-2A in Alert
81007, RWP-3A in Alert
94247, Spent Fuel Pool heat exchanger flow lower than normal
98511, Ultimate Heat Sink temperature requires evaluation
105387, RWV-130 found with leaking weld
106096, Pinhole leak on SWV-354
106795, PM-164 RW pipe liner inspection
113287, SWHE-1C found with 80% blockage
114401, SWP-1B in Alert

Miscellaneous (1R07)

System Health Reports (DC, RW, and SW) for July-December 2004
Chemistry data base for DC and SW Systems
Pump RWP-2B Axial Vibration and Differential Pressure Graph
Pumps RWP-2A, 2B, and 3A Differential Pressure Graphs
SW System Heat Exchanger Blockage History
Memorandum for Temporary Guidance for RW System dated December 9, 2003