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

July 26, 2005

EA-05-03

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 05000302/2005003 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Young:

On June 30, 2005, 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 July 8, 2005, with 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.

The report closes one issue involving a small amount of pressure boundary leakage from a reactor vessel head penetration which was identified during your refueling outage in 2001. Because Technical Specifications (TS) prohibit operation with reactor coolant pressure boundary leakage, the NRC concluded that a violation of the TS occurred. However, the violation involved reactor coolant system pressure boundary leakage not avoidable by the reasonable quality assurance measures and management controls that were employed by you. Although this issue constituted a violation of NRC requirements, we have concluded that your actions did not contribute to the degraded condition and, thus, no performance deficiencies were identified. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation. An evaluation was performed and we have determined this issue involved a low to moderate safety significance. The reactor head at Crystal River has been replaced with a new head constructed of material that is less susceptible to cracking and leaking of penetrations. This generic problem is the subject of NRC Bulletins 2001-01, and 2002-02, and NRC Order EA 03-009 and its first revision. NRC actions to generically address this problem, have resulted in new requirements for licensees to effectively examine the reactor vessel head penetrations for flaws on a periodic basis.

In addition, the report documents one inspector identified finding and one self-revealing finding, both of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the issues, and because each was entered into your corrective action program, the NRC is treating the issues as Non-Cited violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, NRC Region II; The Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Crystal River Unit 3 site.

In 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/

Charles A. Casto, Director
Division of Reactor Projects

Docket No.: 50-302
License No.: DPR-72
Enclosure: Inspection Report 05000302/2005003
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-302
License Nos:	DPR-72
Report No:	05000302/2005003
Licensee:	Progress Energy Florida (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	S. Stewart, Senior Resident Inspector J. Brand, Acting Senior Resident Inspector R. Reyes, Resident Inspector J. Ortiz, Reactor Inspector (Section 4OA3.3) L. Miller, Senior Emergency Preparedness Inspector (Section 1EP2-1EP5, 4OA1) R. Aiello, Senior Operations Engineer (Section 1R11.2) S. Ninh, Senior Project Engineer
Approved by:	Joel T. Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2005-003;04/01/2005 - 06/30/2005; Crystal River Unit 3; Surveillance Testing and Event Followup.

The report covered a three month period of inspection by the resident inspectors, a regional project engineer, two regional inspectors, and a senior emergency preparedness inspector. 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. 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation when the licensee failed to establish appropriate quantitative acceptance criteria to assure offsite power operability for compliance with Crystal River 3 Technical Specification 3.8.1.

  This finding is more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The finding was of very low safety significance according to the SDP Phase 1 worksheet since none of the functions identified in phase 1 were degraded as a result of this deficiency. (Section 1R22)

- Green. A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures occurred in the ‘A’ EGDG during fast start surveillance tests, conducted on April 23, 2004 (NCR 125149) and March 23, 2005 (NCR 154522). A third related failure was identified in July 5, 2001, when the ‘B’ EGDG failed to start during a monthly surveillance test (NCR 44603).

  This finding is more than minor because it directly affected the mitigating system cornerstone objective of ensuring the reliability and operability of a mitigating system. This issue was of very low safety significance because the condition did not affect the capability of the ‘A’ EGDG to perform its design safety function. In addition, the slower fast start time was bounded by the accident analysis calculations. Corrective actions included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system. (Section 4OA3)

B. Licensee-identified Violations

None

REPORT DETAILS

Summary of Plant Status:

Crystal River Unit 3 reduced power to 60 percent for secondary plant maintenance on April 18 and returned to full power on May 2. On June 17, operators reduced power from 100 percent to 85 percent due to an unexpected trip of one of four non-safety related circulating water pumps (CWP-1D). Unit 3 was returned to full power operation on June 20 where it operated through the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, [RO]; Emergency Preparedness [EP]

1R01 Adverse Weather Protection

Seasonal Susceptibility: Hurricanes

a. Inspection Scope

The inspectors reviewed the licensee's hurricane season preparations using the licensee's Emergency Management Procedure EM-220, Violent Weather, Revision 29. The inspectors checked that the licensee maintained the ability to protect vital systems and components from high winds and flooding associated with hurricanes. Additionally, the inspectors toured the six plant areas listed below to check for any vulnerabilities, such as inadequate sealing of water tight penetrations, inoperable sump pumps, or degraded barriers, that could affect the associated systems. The inspectors verified that the licensee's violent weather committee had been established and that an initial preparatory walkdown had been completed. The inspectors verified the spent fuel pool roof damage caused by a previous hurricane, had been repaired. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting adverse weather protection issues. The inspectors attended the Plant Manager's meeting on Hurricane Readiness to assess whether the licensee was adequately prioritizing any open issues.

- Emergency diesel generator (EGDG) building flood walls and doors
- Emergency feedwater pump (EFP) 3 building, including internal sump
- Berm area
- Control complex roof
- Spent fuel pool building and roof
- Intake canal area

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope

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

- April 12, Emergency Feedwater Pump EFP-3 and associated piping using operating procedure OP-450, Emergency Feedwater System, when Feedwater Pump FWP-7 was out-of-service for maintenance

- May 25, Emergency Core Cooling System (ECCS) Train 'B' (DC Closed Loop, Decay Heat, and Building Spray) using procedures OP-404, Decay Heat Removal System, and OP-405, Reactor Building Spray System, during an ECCS Train 'A' outage.

- May 31, Emergency Diesel Generator EGDG-1A, using Operating Procedure OP-707, Operation Of The Engineered Safeguards Diesel Generators, while the EGDG-1B was out of service for testing.

- June 1, Emergency Core Cooling System Train 'A' (DC Closed Loop, Decay Heat, and Building Spray) using procedure OP-404, Decay Heat Removal System, and OP-405, Reactor Building Spray System, during an ECCS Train 'B' outage.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection Walkdowns

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's implementation of their fire protection program. The inspectors checked that safety equipment was 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 any conditions or deficiencies existed which could impair the function of that equipment. The inspectors selected the areas based on a review of the licensee's probabilistic risk assessment. The inspectors toured the following eleven areas important to reactor safety:

- Emergency Feed Pump EFP-1, EFP-2, Areas
- Emergency Diesel Generator A and B Rooms
- Auxiliary Building 143' Level, charcoal filters, misc Waste tanks
- Auxiliary Building 162' Level, refuel floor
- Cable Spreading Room
- 1E Battery Rooms
- Remote Shut Down Panel Room
- Unit 3 Control Room
- Emergency Feed Water Pump EFP-3 Building, and diesel fuel tank room
- Decay Heat / Building Spray Pump 'A' Cubicle
- Decay Heat / Building Spray Pump 'B' Cubicle

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

On April 6, 2005, the inspectors observed the licensee's fire brigade response to an unannounced simulated fire in the Control Complex Alternate Shutdown Panel. 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 used and that any discrepancies were discussed and resolved. In addition, the drill observation Administrative Instruction AI-2205, Administration Of CR-3 Fire Brigade Organization And Duties Of The Fire Brigade, and the fire drill evaluation report were reviewed to assure the acceptance criteria were evaluated and deficiencies were documented and corrected.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

External Flood Protection

a. Inspection Scope

The inspectors performed an inspection of the external flood protection features for Crystal River, Unit 3. The inspectors reviewed the Final Safety Analysis Report (FSAR),

Chapter 2.4.2.4, Facilities Required for Flood Protection, that depicted the design flood levels and protection for areas containing safety-related equipment to identify areas that may be affected by external flooding. The inspectors interviewed the violent weather committee manager, and conducted a general site walkdown of all external areas of the plant including the turbine building, auxiliary building, and berm to ensure that flood protection measures were erected in accordance with design specifications. Emergency procedure EM-220, Violent Weather was checked to verify that adequate measures were established to protect against external flooding due to hurricanes. Specific plant attributes that were checked included structural integrity, sealing of penetrations below the design flood line, and adequacy of watertight doors between flood areas. The inspectors also reviewed work order package 00546764, which documented inspections completed by the licensee on the watertight doors and flood gates.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Observed Simulator Session

a. Inspection Scope

On May 23, 2005, the inspectors observed licensed operators response and actions on the Crystal River Unit 3 Simulator Evaluated Session, SES–28. In addition to responding to multiple equipment failures, the session required the crew to use plant abnormal and emergency operating procedures (EOPs) to respond to a Loss Of Service Water and a Large reactor coolant system (RCS) Hot Leg Leak which eventually escalated into an Emergency Alert Declaration. The EOPs entered included EOP-03, Inadequate Subcooling Margin, and EOP-08, LOCA Cooldown. The inspection focused on high-risk operator actions performed during implementation of the emergency operating procedures; emergency plan implementation using emergency management procedure EM-202, Duties of the Emergency Coordinator; and the incorporation of lessons learned from previous plant events and simulator sessions. Through observations of the critique conducted by training instructors and plant management following the session, the inspectors assessed whether appropriate feedback was provided to the licensed operators regarding any identified weaknesses.

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
- Implementation of Emergency Operating Procedures
- Control Board operation and manipulation, including operator actions
- Oversight and direction provided by supervision, including ability to identify 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.

.2 Annual Operating Test Results

a. Inspection Scope

On February 21, 2005, the licensee completed the comprehensive requalification written examinations and annual operating tests, required to be given to all licensed operators by 10 CFR 55.59(a)(2). The inspectors reviewed the overall pass/fail results of the individual operating tests, and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, Operator Requalification Human Performance Significance Determination Process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components within the scope of the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored, and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors conducted this inspection for the degraded equipment condition associated with the item listed below.

- NCR 156692, Briefing for SP-311, Diesel Fuel Transfer Pump Surveillance

- NCR 156168, Service Water Pump (SWP)-1B North Pump Vibration In Alert Range

- NCR 154522, March 5, 2005, EGDG-1A Excessive Fast Start Time during surveillance test per SP-354A

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the risk impact of removing from service those components listed below and verified the licensee's associated risk management activities. This review primarily focused on equipment determined to be risk significant within the maintenance rule. The inspectors also assessed the adequacy of the licensee's identification and resolution of problems associated with risk management including emergent work activities. The licensee's implementation of their compliance procedure CP-253, Power Operation Risk Assessment, was verified in each of the following work week assessments.

- Work Week 05W15, Risk assessment for operations with the pressurizer block valve (RCV-11) shut and feedwater pump FWP-7 out of service for preventive maintenance item (elevated risk condition orange).

- Work Week 05W16, Risk assessment for operations with the pressurizer block valve (RCV-11) shut, and EGDG-1A fuel oil transfer pump surveillance testing.

- Work Week 05W19, Risk assessment for operations with the pressurizer block valve (RCV-11) shut revised when Service Water Heat Exchanger SWHE-1C was removed from service to repair minor leakage from the channel head.

- Work Week 05W21, Risk assessment for operation in condition Yellow due to pressurizer block valve (RCV-11) shut and the 'A' train emergency core cooling system out of service for planned maintenance.

- Work Week 05W22, Risk assessment for operation in condition Yellow due to pressurizer block valve (RCV-11) shut and the 'B' train emergency core cooling system out of service for planned maintenance.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

On June 17, the inspector evaluated the operators response to an unexpected trip of the non-nuclear safety related 'D' circulating water pump (CWP-1D) which resulted in a controlled manual power reduction from 100% power to 85% power. The CWP-1D pump trip occurred during restoration of cooling water flow to the 'D' circulating water pump. The root cause investigation was in progress per NCR 161558. Preliminary evaluations indicated the most likely cause of the failure was water spraying on the motor windings from a hole in the cooling water line for the motor cooler. The hole is believed to have resulted from galvanic corrosion. The inspectors interviewed control room operators, and reviewed operator logs, records, and applicable procedures to determine that the power reduction evolution was properly conducted. The inspectors also verified that station personnel initiated actions to investigate an unexpected power increase from 85% power to 90% power after completion of the power reduction. The

investigations are being performed under NCRs 162702, Unexpected Integrated Control System (ICS) Response After Exiting Track," and NCR 161564, "SG/Rx Demand Station Did Not Respond As Expected."

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following NCRs to verify that the 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 were 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 NCRs, work schedules, and engineering documents to check if operability issues were being identified at an appropriate threshold and documented in the corrective action program, consistent with 10 CFR 50, Appendix B requirements, and licensee procedure NGGC-CAP-200, Corrective Action Program.

- NCR 159013, Non-Conservative Engineered Safeguards Bus Voltage Limits in Surveillance Procedure SP-321, Power Distribution Breaker Alignment and Power Availability Verification

- NCR 159784, Increasing Reactor Building Sump Level Due to Decay Heat Valve DHV-42 Leak-By

- NCR 157407, Pressurizer Heater Breakers Tripping During Spray Operations

- NCR 152691 2.5 gpm leak from the pressurizer power operated block valve RCV-11.

- NCR 160653, Repetitive 'B" Battery Ground Alarms

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

Selected Issue Review

On May 6, 2005, the inspector reviewed the operator work around (OWA) listed below, taken from the licensee's OWA list. The inspector reviewed the operations activity and the nuclear condition report associated with the OWA. Compensatory actions addressing the OWA were reviewed. The inspector checked that the affected safety function was maintained with appropriate compensatory measures while the deficient condition existed.

- NCR 152691, Increased Reactor Coolant System Leakage following Stroke Test of Reactor Coolant Valve, RCV-11.

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 five 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.

1R17 Permanent Plant Modifications

a. The inspectors evaluated the design change package listed below for the potential adverse effects on the 'A' emergency diesel generator fast start function relating to time-to-start the engine. This modification replaced a 2-psi check valve with a 10-psi check valve. The check valve is located downstream of the fuel header ring and provides a valve seating force to assist in maintaining fuel header prime pressure. The inspector observed the as-built configuration of the modification and observed installation, including licensee quality inspections. Documents reviewed included procedures, design and implementation packages, work orders, site and vendor drawings, corrective action documents, applicable sections of the updated final safety analysis report, Technical Specifications, and design basis information. Post maintenance diagnostic data and acceptance criteria were reviewed with engineering and verified prior to starting the engine.

- Engineering Change 60670RO; Replace DFV-61 With a Check Valve Set at 10 PSI Cracking Pressure

b. Findings

No findings of significance were identified

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and/or reviewed post-maintenance testing 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 tests reviewed are listed below:

- Surveillance Procedure SP-112, Calibration Of Reactor Protection System, performed on April 6, after replacement of a buffer amplifier module per WO 662025-1.

- Surveillance Procedure SP-354A, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1A (fast start portion), performed on April 20, after completing a modification which installed a 10-psi check valve down stream of the fuel header per WO 692223-02 to address a slow fast start failure that occurred on March 5, 2005 during a surveillance test.

- Surveillance Procedure SP-340A, RWP-3A, DCP-1A And Valve Surveillance, performed on May 25, after performing maintenance on the ECCS Train ‘A’ per WO 686519-01.

- Surveillance Procedure SP-375A, CHP-1A And Valve Surveillance, performed on June 16, after performing maintenance on the ‘A’ chiller motor and pump per WO 386034-02.

- Surveillance Procedure SP-179B, Containment Leakage Test-Type B, performed on June 15, after replacement of a leaking rupture disc (DHRS-1) for containment penetration No. 329.

- WO 726765-04 ICS Operational Test Instructions, performed on June 28, after replacement of ICS Rate Follower limiter Module.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests listed below to verify that technical specification surveillance requirements were followed and that test acceptance criteria were properly specified. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met. Additionally, the inspectors also verified that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance or as part of surveillance testing. The following seven activities were observed/reviewed:

In-Service Test:

- SP-349B, Emergency Feedwater Pump (EFP-2) And Valve Surveillance Testing
- SP-344B, Service Water Pump SWP-1B Surveillance Testing
- SP-334B, Spent Fuel Pool SFP-1B Quarterly Surveillance

Reactor Coolant Leakage Detection System Surveillantce

- SP-317 Reactor Coolant System Leak Rate Determination

Other Surveillance Tests:

- SP-348A, Auxiliary Feedwater Pump (FWP-7) Testing and MTDG-1 Surveillance Testing
- SP354A, Monthly Functional Test Of The Emergency Diesel Generator EGDG-1A (Fast Start Only)
- SP-321, Power Distribution Breaker Alignment and Power Availability Verification

b. Findings

Introduction: A Green NCV was identified when the licensee failed to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply.

Description: During a review of licensee surveillance procedure SP-321, Power Distribution Breaker Alignment and Power Availability Verification, the inspector identified a number of quantitative acceptance criteria in the surveillance that would not assure adequate voltage for the offsite power supply to determine the operability of the circuits between the off-site transmission network and the on-site Class 1E distribution. Operability verification of the offsite power supply for compliance with Crystal River 3

Technical Specification 3.8.1, (in Surveillance Requirement 3.8.1) is done by weekly performance of SP-321.

Analysis: A performance deficiency, affecting the Mitigating Systems cornerstone, was identified when the licensee did not establish appropriate quantitative acceptance criteria to assure operability of the offsite power supply to the on-site Class 1E distribution. The issue was more than minor because if left uncorrected, a more significant safety concern could occur if a low voltage condition of the offsite power supply was not immediately recognized and corrected. The significance determination process, phase 1 review was completed with all questions in the mitigating systems checklist (page 2) answered no, and the issue screened as Green.

Enforcement: 10 CFR 50, Appendix B, Criterion V, Procedures, states, in part, that procedures shall include appropriate quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Crystal River 3 Technical Specification 3.8.1 requires that two circuits between the offsite transmission network and the onsite Class 1E electrical power distribution system shall be operable. Offsite power operating limits, established by Crystal River 3 calculation E-90-0077, Revision 3, "Safety related busses scenario based steady state voltage drop and load flow analysis from offsite power sources," define the operating parameters that ensure operability under all conditions. This requirement is verified by licensee surveillance SP-321, Power Distribution Breaker Alignment and Power Availability Verification, which ensures correct breaker alignment, voltage, and indicated offsite power availability, and other parameters. Contrary to the above, as of May 15, 2005, Surveillance Procedure SP-321 did not include the correct quantitative acceptance criteria as indicated below.

1) Engineered Safeguards 4160 bus voltage between 4000 and 4400 volts - A minimum voltage of 4010 volts is required to reset a degraded voltage protection relay and a minimum voltage of 4110 is necessary for operability

2) 230 KV Bus Voltage approximately 230 KV - A minimum voltage of 235 KV is required to assure degraded voltage isolation and protection does not occur in an engineered safeguards actuation.

3) Control room annunciator SSF-A2-07-07, 230 KV GRID DEGRADING, alarms when 4160 volt Bus A voltage is less than 4019 volts for greater than 2 seconds. SP-321 verifies that ES 4160 volt busses are Operable by Absence of this alarm. The licensee informed the inspector that a minimum voltage of 4110 would be required to assure no degraded voltage protection during ES loading.

When identified to the licensee, the issue was documented in the corrective action program. In addition, an interim compensatory measure to log actual 4160 volt bus voltage (greater than 4110 volts) during performance of SP-321 was implemented as well as a procedure change proposed to upgrade the procedure. Because this failure to have adequate acceptance criteria for the technical specification surveillance was of very low safety significance and had been entered into the licensee's corrective action program as NCR 159013, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000302/2005003-

01 , Failure to assure adequate acceptance criteria for technical specification surveillance.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the 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 60676R0, Perform Leak Repair on MSV-178

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors evaluated the adequacy of licensee methods for testing the alert and notification system in accordance with NRC Inspection Procedure 71114, Attachment 02, “Alert and Notification System (ANS) Testing”. The applicable planning standard 10 CFR 50.47(b)(5) and its related 10 CFR Part 50, Appendix E, Section IV.D requirements were used as reference criteria. The criteria contained in NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, were also used as references.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation

a. Inspection Scope

The inspectors reviewed the ERO augmentation staffing requirements and the process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The results of the March 23, 2004 unannounced off-hours augmentation drill were reviewed. The inspectors conducted a review of the backup notification systems. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current. A sample of problems identified from augmentation drills or system tests performed since the last inspection was reviewed to assess the effectiveness of corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, "Emergency Response Organization Augmentation Testing." The applicable planning standard, 10 CFR 50.47(b)(2) and its related 10 CFR Part 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspectors evaluated the 10 CFR 50.54(q) reviews associated with non-administrative emergency plan, implementing procedures and EAL changes. The revisions covered the period from February 2004 to April 2005.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 01, "Emergency Action Level and Emergency Plan Changes." The applicable planning standard, 10 CFR 50.47(b)(4) and its related 10 CFR Part 50, Appendix E requirements were used as reference criteria. The criteria contained in NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels", Revision 2, and Regulatory Guide 1.101 were also used as references.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors reviewed the corrective actions identified through the EP program to determine the significance of the issues and to determine if repeat problems were occurring. The facility's self-assessments and audits were reviewed to assess the licensee's ability to be self-critical, thus avoiding complacency and degradation of their EP program. In addition, the inspectors reviewed licensee self-assessments and audits to assess the completeness and effectiveness of all EP-related corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, "Correction of Emergency Preparedness Weaknesses and Deficiencies." The applicable planning standard, 10 CFR 50.47(b)(14) and its related 10 CFR Part 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed and reviewed two emergency response activities to verify the licensee was properly classifying emergency events, making the required notifications, and appropriate protective action recommendations. The inspectors assessed the licensee's ability to classify emergent situations and make timely notification to State and Federal officials in accordance with 10 CFR Part 50.72. The inspectors also evaluated classification and notification activities completed by the licensee's technical support center and emergency offsite facility staff for the emergency response facility drill. 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.

- On May 11, 2005, the licensee conducted an emergency response facility involving a large break loss of coolant accident scenario which included a number of complicating events

- On May 23, 2005, licensed operator Simulator Evaluated Session, SES-28, involving a loss of service water and a large RCS hot leg leak

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors reviewed the licensee's procedure for developing the data for the EP PIs, which are: (1) Drill and Exercise Performance (DEP); (2) ERO Drill Participation; and (3) ANS Reliability. The inspectors examined data reported to the NRC for the period April 2004 to March 2005. Procedural guidance for reporting PI information and records used by the licensee to identify potential PI occurrences were also reviewed. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee's records of periodic system tests.

The inspection was conducted in accordance with NRC Inspection Procedure 71151, "Performance Indicator Verification." The applicable planning standard, 10 CFR 50.9 and NEI 99-02, Revision 2, "Regulatory Assessment Performance Indicator Guidelines," were used as reference criteria. This inspection activity represents three samples on an annual basis.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Screening of Items Entered Into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. 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. However, the inspectors identified three trends as discussed in Section 4OA2.2.

.2 Semi-Annual Trend Review

c. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in section 4OA2.1 above, plant status reviews, plant tours, and licensee trending efforts. The inspectors' review nominally considered the six month period of January 2005 through June 2005, although some examples may have expanded beyond those dates when the scope of the trend warranted. The review also included issues documented outside the normal CAP in equipment problem lists such as: the Online Degraded Equipment Audit Report dated June 20, 2005; the Emergency Diesel Generator Health Report (July to December 2004); the Active Boron Leak List (including boric acid corrosion program walkdowns) of the reactor building; the 2005 Spent Fuel Pool Liner Leakage Monitoring And Management Plan; the Crystal River Unit 3 First Quarter 2005 CAP Rollup And Trend Analysis; and various maintenance rule assessments.

d. Assessment and Observations

No findings of significance were identified. However, the inspectors noted three negative trends in reviewing licensee performance over the last six months. Except for the 'A' EGDG finding discussed in Section 4OA3.2, none of the issues identified were significant nor do they constitute a violation of NRC requirements. A minor procedural non-compliance was also identified regarding the negative trend involving the installation of seismic scaffolds as shown below. The licensee entered these issues in their corrective action process under NCR-163266, and initiated a series of management debriefs to enhance employee awareness on the use of the corrective action process.

First, a negative trend was noted in the threshold for documenting issues in a condition report. In some cases, the inspector noted that multiple disciplines missed opportunities to identify degraded conditions. In other cases, the inspector noted that repeat occurrences of known degraded conditions were not entered into the CAP for trending or evaluation. Examples where CRs were not initially written included; a December 3, 2004 unexpected reactivity change; concerns raised by a laboratory analysis report which documented the results of the 'A' EGDG slow fast start failure (i.e., damaged O ring, potentially incorrectly reassembled check valve, small debris particles and fibers identified, and localized fretting on the valve bore); battery grounds not consistently documented for trending or investigation; concerns raised by a worker regarding slow draining of the reactor building 'D' ring drains; and, building spray pump BSP-1A boric acid leak.

The second negative trend indicates an acceptance of long standing degraded conditions In some cases, the inspector noted that repeat occurrences of known degraded conditions were not entered into the CAP for trending or evaluation. Examples included:
Emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves contributed to three EGDG engines failing to start properly; 1B EGDG on July 5, 2001; 1A EGDG on April 23, 2004; and 1A EGDG on March 23, 2005.

- Containment ventilation fan (AHF-1B) slow "fast start" was incorrectly evaluated by operators as an operator human performance issue (AR 144148) on November 18, 2004. A similar slow "fast start" occurred on February 10, 2005 and was evaluated by engineering as a relay problem (AR 150801, WO 673978), however the issue was not corrected. As a result, the condition repeated on June 28, 2005 (NCR 162272). A preliminary engineering evaluation has determined that there was no loss of function. The licensee is currently evaluating this issue for past operability.

- Repeated battery ground alarms. "B" Battery ground alarm reported on December 3, 2004 (NCR 145092), but not corrected. NCR 155388, dated April 2, 2005, for a "B" battery ground alarm that lasted for over one hour was closed stating the condition could not be repeated.

The third trend involves improperly installed seismic scaffold structures and transient materials used for scaffolding that could potentially degrade the reliability of safety-related components. Specifically the inspectors identified three scaffolds that did not meet the 2 inch seismic separation clearance per procedure MNT-NGGC-004 (DHV-34, NCR 162609; AHF-15A, NCR 160445; and Fire Protection panel FSCP-2). The inspectors determined that this procedural non-compliance was minor because operability of the components was not affected and the issue has been entered in the licensee's CAP program. The inspectors also identified three scaffold carts and other rolling equipment located in the service water pump and heat exchanger area that were not properly secured (NCR 160445). When identified to the licensee, the listed NCRs were initiated to corrected the improperly installed scaffolds.

.3 Annual Sample Review - Emergency Diesel Generator Inoperable Due to Fuel Header Outlet Check Valve Leaking Past Seat

a. Inspection Scope

The inspectors selected NCRs 125149 and 154522 and associated documents for a detailed review due to the repetitive nature of the condition. The NCRs were initiated to evaluate 1) a March 23, 2005, condition when the emergency diesel generator EGDG-1A did not achieve the steady state voltage and frequency reading in less than or equal to 10 seconds from standby conditions as required by technical Specification 3.8.1.6 during performance of a bi-annual fast start surveillance test run, and 2) An April 23, 2004 condition when EGDG-1A was also not able to start within the specified time, during performance of the same surveillance test.

b. Findings

No findings of significance were identified.

The inspectors noted that although the specific fuel oil components described in each event were different, both events involved a loss of fuel oil header prime. In addition, the inspectors observed that a previous event associated with a failure to start of the EGDG-1B (July 5, 2001 - NCR 44603), was attributed to a loss of fuel header prime. The inspector reviewed the corrective actions proposed to eliminate this problem and verified that actions seem reasonable to preclude repetition. (Reference section 4OA3.2 for finding associated with this issue.)

During this inspection the inspectors noted several issues that potentially contributed to these events, but were not addressed by the licensee. The inspectors reviewed an independent laboratory analysis 05-089, conducted to determine the cause of leakage past the two fuel oil header check valves on the 'A' EGDG on March 23, 2005. The inspector noted that this report raised several other concerns that potentially contributed to the loss of prime condition, but had not been specifically addressed by the licensee in the root cause evaluation. These concerns included; 1) a damaged O-ring, 2) potentially incorrectly reassembled check valve after it was removed from the system for onsite investigation and testing, 3) several small debris particles (possible corrosion products) and fibers identified, and 4) localized fretting on the valve bore and mechanical abrasion of one of the check valves poppet. The report also stated that there was an interface issue that could affect the performance of the larger check valve (DFV-39). In addition, the inspector noted that the check valves have been purchased as non-safety and commercially dedicated since 1996. Since that time, many of the new valves have not passed the leak test used for the dedication process prior to being installed in the system. The inspector questioned the appropriateness of the valves in this application.

4OA3 Event Followup

.1 (Closed) LER 05000302/2004-003-00: Reactor Trip and Emergency Feedwater Actuation Caused by 230 Kilovolt Switchyard/Transmission Faults

e. Inspection Scope

The event report summarized an event that occurred on September 6, 2004, when during Tropical Storm Frances, two independent faults occurred in the offsite power network. The faults resulted in the loss of power to the B Engineered Safeguards 4160 volt bus and startup transformer, loss of power to the reactor coolant pumps, a reactor trip and actuation of emergency feedwater. The inspectors reviewed the licensee's nuclear condition report NCR 136752 which included the root cause investigation report and summary of corrective actions. The root cause was determined to be the loss of the Brookridge 230 Kilovolt transmission line due to corrosion/erosion and tripping of the 230 Kilovolt Bus B due to phase to ground flashover, both during tropical storm conditions. No performance deficiency was identified. Corrective actions included replacement of the failed (and similar) insulators in the offsite transmission network and

upgrade of the preventive maintenance programs for the offsite transmission network. The inspectors checked the accuracy and completeness of the LER and the appropriateness of the licensee's review, safety assessment, and corrective actions. The occurrence was screened using NRC Manual Chapter 0612, Appendix B.

f. Findings

No findings of significance were identified. The LER is closed.

.2 (Closed) LER 05000302/2005-002-00: Emergency Diesel Generator Inoperable Due to Fuel Header Outlet Check Valve Leaking Past Seat

a. Inspection Scope

The inspectors reviewed LER 2005-002-00, which summarized an event that occurred on March 23, 2005, when the emergency diesel generator EGDG-1A did not achieve the steady state voltage and frequency reading within the time required by technical specifications. The inspectors checked the accuracy and completeness of the LER and the appropriateness of the licensee's review, safety assessment, and corrective actions.

b. Findings

Introduction: A self revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified for failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, two separate slow fast start failures that affected the reliability and operability of the 'A' EDG occurred.

Description: On March 23, 2005, EGDG-1A did not achieve steady state voltage and frequency in less than or equal to 10 seconds from standby conditions as required by technical specification 3.8.1.6 during performance of a surveillance test run per procedure SP-354A, "Monthly Functional test Of The Emergency Diesel Generator EGDG-1A." Specifically, the fast start time was 14.8 seconds. An engineering evaluation (NCR 154522) determined that the cause of this event was a loss of fuel oil header prime due to leakage past the diesel fuel oil header check valves DFV-39 and DFV-61. The evaluation also determined that the contributing cause to this event was an inadequate fuel oil header design in which the fuel oil header is the system high point allowing the fuel oil to gradually drain back through the header to the diesel fuel oil tank, partially depleting the fuel oil header prime. Corrective actions for this event included, replacing the fuel oil check valves with a higher closing spring force valve, priming the system three times per month, and initiating actions to modify the fuel oil system.

A similar condition occurred on April 23, 2004, when the EGDG-1A took 11.4 seconds during the fast start surveillance test. The engineering evaluation of this failure (NCR 125149) determined the failure to be caused by a small piece of foreign material lodged in the seat of the EGDG fuel oil header outlet check valve DFV-61, which allowed fuel header pressure to drop.

The licensee analyzed the foreign material and determined it to be small fragments of Teflon, PVC, and red iron oxide. That issue was documented in LER 2004-002-00

A third related failure was identified in July 5, 2001, when the 'B' EGDG failed to start during a monthly surveillance test. The engineering evaluation of this condition (NCR 44603) determined that the cause was a loss of prime of the engine fuel header due to multiple fuel oil system degraded parts, including; a degraded injection pump discharge check valve cage O-ring, which provided an air in-leakage site during engine standby condition; a damaged seat on the fuel header inlet check valve; and, a stuck open fuel header outlet check valve.

The inspector and the licensee noted that all of the events involved the loss of fuel header prime. In addition, the licensees evaluation of the March 2005 fast start failure determined that corrective actions for the August 2001 and April 2004 events had been ineffective because there was no change to the system design or surveillance procedure that would have assured the fuel header prime would not be lost.

The inspectors identified a self revealing finding for ineffective corrective actions to assure that conditions adverse to quality are properly identified and corrected.

Analysis: The inspectors determined that failure to properly evaluate and correct a long standing emergency diesel generator (EGDG) loss of fuel oil header prime condition is a performance deficiency that caused repeat slow fast starts or failures to start in both EGDGs which affected their reliability and operability. This finding is more than minor because it directly affected the mitigating system cornerstone objective of the reliability and operability of a mitigating system. Using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1, this finding was determined to be of very low safety significance (Green) since the condition did not result in an actual failure of the 'A' EGDG, and the equipment remained available to perform its design function. In addition, the slower fast start time was bounded by the accident analysis calculations. The primary cause of the finding was related to the cross cutting area of Problem Identification and Resolution, because station personnel missed several opportunities to properly evaluate and correct this degraded condition.

Enforcement: Appendix B, Criterion XVI of 10 CFR Part 50 states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected." Contrary to this requirement, station personnel failed to properly evaluate and correct a long standing EGDG loss of fuel oil header prime condition caused by leakage past the fuel header check valves. As a result, multiple slow fast start and start failures have occurred that affected the reliability and operability of both EGDGs. Because this issue was of very low safety significance and because it has been entered into the licensee's corrective action program (CR-15422), the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy, NCV 05000302/2005-003-02, Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves. This LER is closed.

.3 (Closed) Licensee Event Report (LER) 050000302/2001-004-00, Reactor Pressure Vessel Head (RVH) Leakage Due to Control Rod Drive Mechanism (CRDM) Nozzle Degradation

On October 1, 2001, while performing a visual inspection of the reactor vessel head, the licensee identified evidence of Reactor Coolant System (RCS) leakage, in the form of boric acid residue, on CRDM nozzle (# 32). Further ultrasonic examination confirmed that two axially oriented cracks (through-wall) on CRDM nozzle # 32 were the source of RCS leakage. No other CRDM nozzles inspected revealed a similar boric acid buildup. Technical Specification Limiting Condition for Operation 3.4.12.a, limits RCS operational leakage to "no pressure boundary LEAKAGE" in MODE 1 - 4. The residual boric acid indicated that RCS operational leakage had existed while at power, therefore this constitutes a violation of Technical Specification 3.4.12.a. The licensee's corrective actions were: 1) repair the affected nozzle using the ambient temperature temper bead method, 2) examine the new weld by ultrasonic testing (UT) and penetrant testing, 3) perform an in-service leakage test at nominal operating pressure and temperature, and 4) perform UT on eight additional CRDM nozzles. Additionally, the licensee replaced the reactor head in the fall of 2003. The new head was constructed with nozzles using material which is less susceptible to cracking. The issue was identified during the first inspection conducted by the licensee as a commitment in response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles", which was issued on August 3, 2001. The NRC did not identify a performance deficiency associated with this event.

The NRC evaluated this issue for risk and considered the fact that the nozzles had not previously been examined by techniques that would have identified circumferential flaws. The analysis used the most recent version of the NRC's risk assessment tool, a spreadsheet developed by Argonne National Laboratory. This tool was developed based on the results of plant inspections and laboratory experiments. It infers the probabilities for early crack initiation and rapid crack growth from operating experience of leaking CRDM nozzles that have been discovered. This information is adjusted for the operating temperature of the reactor head to place it in context with data from other plants and laboratory experiments. The calculation is performed for nozzles that intersect the head surface at several different angles, and interpolation is used to obtain a result for the angle(s) of the leaking nozzle(s) in a specific plant evaluation. The probability that a leaking nozzle will develop a circumferential crack is an empirical value (0.2) observed in the population of plants inspected to date. The spreadsheet calculates a distribution for the probability that a leaking nozzle will be ejected, as a function of plant age. The mean of the probability distribution for the year prior to discovery is used as the increase in the frequency of medium loss of coolant accident (LOCA) for that year (i.e., if a nozzle were ejected, a medium LOCA would result). The increase in the core damage frequency is estimated by multiplying this increase in medium LOCA frequency by the conditional core damage probability for medium LOCAs in the SPAR model for the plant. The parameters for this case are 1 leaking nozzle (22° angle to head), 69 total nozzles, 15.6 effective full power years (EFPY) at time of discovery, 602°F head temperature.

The evaluation determined this to have resulted in a risk of low to moderate safety significance. Pressure boundary leakage while operating in Modes 1 - 4 constitutes a violation of Technical Specification 3.4.12.a. However, as discussed in the NRC's Enforcement Policy, the NRC may refrain from issuing enforcement action for violations resulting from matters not within the licensee's control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls. Based on the circumstances of this violation, the NRC considers it appropriate to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for this violation.

4OA4 Cross-Cutting Aspects Of Findings

Section 4OA3.2 describes a finding in which a long standing emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves was not properly evaluated and corrected. This finding is a cross cutting issue in the area of problem identification and resolution, because station personnel missed several opportunities to properly evaluate and correct this degraded condition.

4OA5 Other

(Closed) Temporary Instruction (TI) 2515/163: Operational Readiness of Offsite Power

The inspectors collected data pursuant to TI 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems;" 10 CFR 50.63, "Loss of All Alternating Current Power;" 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this TI are listed in the attachment.

4OA6 Meetings

.1 Exit Meeting Summary

On July 8, 2005, the resident inspectors presented the inspection results to Mr. Daniel L. Roderick, Director Site Operations, and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On May 10, 2005, 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, 2004 - December 31, 2004. The major topics addressed were: the NRC ’s assessment program, the results of the Crystal River 3 assessment, and future NRC inspection activities. Attendees included FPC

management, FPC site staff, and one member of the public.

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 ML051370200. Licensee's handout was presented at the meeting is also available from the NRC's document system (ADAMS) as accession number ML051370209. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
M. Annacone, Manager, Engineering
W. Brewer, Manager, Maintenance
R. Davis, Manager, Training
J. Franke, Plant General Manager
D. Hanna, Supervisor, Self Evaluation and Emergency Preparedness
J. Hays, Manager, Outage and Scheduling
J. Holt, Manager, Operations
S. Powell, Supervisor, Licensing
M. Rigsby, Radiation Protection Manager
D. Roderick, Director Site Operations
J. Stephenson, Principal Nuclear Emergency Preparedness Specialist
R. Warden, Manager, Nuclear Assessment
D. Young, Vice President, Crystal River Nuclear Plant

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000302/2005003-01	NCV	Failure to establish appropriate quantitative acceptance criteria to assure Crystal River 3 Technical Specification 3.8.1 operability of the offsite power supply (Section 1R22)
05000302/2005003-02	NCV	Failure to properly evaluate and correct emergency diesel generator loss of fuel oil header prime condition caused by leakage past the fuel header check valves.(Section 4OA3)

Closed

05000302/2001-004-00	LER	Reactor Pressure Vessel Head Leakage Due to Control Rod Drive Mechanism Nozzle Degradation (Section 4OA3)
05000302/2004-003-00	LER	Reactor Trip and Emergency Feedwater Actuation Caused by 230 Kilovolt Switchyard/Transmission Faults (Section 4OA3)
05000302/2005-002-00	LER	Emergency Diesel Generator Inoperable due to Fuel Header Outlet Check Valves Leaking Past Their Seat (Section 4OA3)
2515/163	TI	Operational Readiness of Offsite Power (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1EP2: Alert and Notification System Testing

Whelen WPS-2800-5 Series High-Power Voice and Siren Systems
SOP-9, Citrus County Emergency Response Procedure for PIO/EOC-Siren Procedure

Section 1EP3: Emergency Response Organization (ERO) Augmentation

AI-4001, Conduct of Drills and Exercises Supporting the Radiological Emergency Response Plan, Rev. 1
AI-4000, Conduct of Emergency Preparedness and Schedule for Radiological Emergency Response Plan Maintenance, Rev. 4
EM-206, Emergency Plan Roster Notification

Section 1EP4: Emergency Action Level (EAL) and Emergency Plan Changes

Radiological Emergency Response Plan, Revision 24
EM-202, Duties of the Emergency Coordinator, Rev. 75
50.54(q) for EM-202, Duties of the Emergency Coordinator, Rev. 75
EM-911, Security Threats, Rev. 0

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

Hurricane Charley, Emergency Plan Activation Summary/Critique, 8/13/2004
Hurricane Frances, Emergency Plan Activation Summary/Critique, 9/05/2004
Hurricane Jeanne, Emergency Plan Activation Summary/Critique, 9/26/2004
CNAS 2004-0089, CR-3 Nuclear Assessment Section (NAS) Observation/Trend Report Through September 21, 2004
112988, Formal Benchmark Report, Orlando, Fl., 2/26-27/2004
113092, Cross-functional Self-assessment of CR3's Corrective Action Program, 2/19-27/2004
113094, Readiness for INPO Emergency Preparedness Assist Visit, 12/13-1/17/2004
128594, Self-Assessment of Emergency Response Organization (ERO) Personnel Qualification Database (PQD), 6/24-7/8/2004
AR 124152
AR 148065
AR 148698
AR 149350
AR 148552
AR 148071

Section 4OA1: Performance Indicator (PI) Verification

Emergency Response Organization Drill Participation data 4/04-3/05
Alert and Notification System Reliability data 4/04-3/05
Drill and Exercise Performance data 4/04-3/05
CP-217 NRC Performance Indicator (PI) Program, Revision 7
2005 Siren Trending (updated 4/5/2005)
SES-03, Scenario Licensed Operator Training, Rev. 12
SES-28, Scenario Licensed Operator Training, Rev. 7

SES-33, Scenario Licensed Operator Training, Rev. 2
Radiological Emergency Response Plan December 16, 2004 Training Drill Report
Radiological Emergency Response Plan August 11, 2004 Training Drill Report

Section 40A5:

Surveillance procedure SP-321, Power Distribution Breaker Alignment and Power Availability Verification
Compliance procedures CP-212, Safety Function Determination Program, Revision 3
Compliance procedure CP-151, External Reporting Requirements, Revision 15
Progress Energy Interface Agreement, NGGM-IA-0031, Transmission Florida Interface Agreement for Operation, Maintenance, and Engineering Activities
Licensee compliance procedure CP-253, Power Operation Risk Assessment and Management
Licensee procedure AI-500, Conduct of Operations, Appendix 7
Emergency operating procedure EOP-12, Station Blackout, Revision
Abnormal procedure AP-770, Emergency Diesel Generator Actuation, Revision 33