Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.73

April 19, 2007
3F0407-03

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: CRYSTAL RIVER UNIT 3 - LICENSEE EVENT REPORT 50-302/2007-002-00

Dear Sir:

Florida Power Corporation, currently doing business as Progress Energy Florida, Inc., hereby submits Licensee Event Report (LER) 50-302/2007-002-00. The LER discusses actuation of the Reactor Protection System and Emergency Feedwater System caused by a failed circuit board within the Main Feedwater Integrated Control System on February 21, 2007. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

No new regulatory commitments are made in this letter.

If you have any questions regarding this submittal, please contact Mr. Paul Infanger, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant

JAF/seb/dar

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

**Progress Energy Florida, Inc.**
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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**4. TITLE**
Reactor Trip Caused by Failed Circuit Board in the Main Feedwater Integrated Control System

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | N/A | 05000 |
| 02 | 21 | 2007 | 2007 | - 002 - | 00 | 04 | 19 | 2007 | N/A | 05000 |

**9. OPERATING MODE**: 1

**10. POWER LEVEL**: 71%

**11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§:** *(Check all that apply)*

| | | | |
|---|---|---|---|
| ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☒ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER |
| ☐ 20.2203(a)(2)(vi) | ☐ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| David Rothrock – Engineer (Licensing & Regulatory Programs) | (352) 563-4771 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| A | JA | 90 | B045 | Y | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* | ☒ NO | | | | |


ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 19:14, on February 21, 2007, Crystal River Unit 3 (CR-3), operated by Florida Power Corporation (FPC) doing business as Progress Energy Florida, Inc., was in MODE 1 (POWER OPERATION) at approximately 71 percent RATED THERMAL POWER. CR-3 was at reduced power for planned maintenance on one of four main condenser water boxes when the Integrated Control System (ICS) for Main Feedwater (FW) became erratic causing actual FW flow to drop to zero in both FW trains which underfed the Once-Through Steam Generators. The Reactor Protection System (RPS) actuated on high Reactor Coolant System (RCS) Pressure causing a reactor trip. Emergency Feedwater Initiation and Control (EFIC) actuated on low steam generator levels. The cause for this event was inadequate human performance in the implementation of the ICS circuit card refurbishment program which resulted in an age-related failure of the zener diodes in the +15 volt regulator circuit for a Bailey 820 Control Module in the ICS. RPS and Emergency Feedwater System valid actuations are reportable under 10CFR50.73(a)(2)(iv)(A). This condition does not represent a reduction in the public health and safety. LER 50-302/2004-001-00 documents a previous similar event at CR-3.

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**17. NARRATIVE** *(If more space is required, use additional copies of NRC Form 366A)*

EVENT DESCRIPTION

At 19:14, on February 21, 2007, Crystal River Unit 3 (CR-3), operated by Florida Power Corporation (FPC) doing business as Progress Energy Florida, Inc., was in MODE 1 (POWER OPERATION) at approximately 71 percent RATED THERMAL POWER. CR-3 was at reduced power for planned maintenance on one of four main condenser water boxes [SG, COND] when the Integrated Control System (ICS) [JA] for Main Feedwater (FW) [SJ] became erratic causing actual FW flow to drop to zero in both FW trains which underfed the Once-Through Steam Generators [AB, SG]. The Reactor Protection System (RPS) [JC] actuated on high Reactor Coolant System (RCS) [AB] pressure causing an automatic reactor and RPS anticipatory Main Turbine [TA, TRB] trip. Emergency Feedwater Initiation and Control (EFIC) [JB] actuated on low steam generator levels. Immediately prior to the reactor trip, conditions were present for initiation of the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) [JE]. However, AMSAC did not actuate because it had been placed in bypass prior to the event due to neutron detector work. Operators entered Emergency Operating Procedure, EOP-02, "Vital System Status Verification," and performed immediate actions which include backing up the automatic trips from RPS by inserting manual trip signals to both the reactor and Main Turbine.

No structures, system or components were inoperable at the start of the event that contributed to the event. Plant safety systems responded as expected during the reactor trip with the following exceptions:

Although conditions were met for AMSAC initiation at 19:12, AMSAC did not initiate due to ongoing corrective maintenance on the associated neutron detector [JE, DET].

Main Steam Safety Valve MSV-47 [SB, RV] lifted prior to reaching the minimum setpoint per installed gauges. The valve was conservatively declared inoperable.

Upon transfer from the Unit Auxiliary Transformer [EA, XFMR] to the Startup Transformer, various plant loads lost power. Although a loss of some equipment is expected, this loss of equipment was greater than expected and resulted in operator burden following the trip.

Water hammer was observed in heater drain piping [SB] downstream of Heater Drain Valves HDV-249 [SB,V] and HDV-250 at the entry to the hotwell after the reactor trip.

RPS and Emergency Feedwater System (EFW) [BA] valid actuations are reportable to the NRC. At 22:11 on February 21, 2007, a non-emergency four hour notification and a non-emergency eight hour notification were made to the NRC Operations Center (Event Number 43179) in accordance with 10CFR50.72(b)(2)(iv)(B) and 10CFR50.72(b)(3)(iv)(A), respectively. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

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SAFETY CONSEQUENCES

Based on high RCS pressure caused by reduced feedwater flow, valid actuations of RPS and EFW occurred as expected to shut down the reactor, trip the Main Turbine, and maintain adequate steam generator levels. When FW flow decreased to the steam generators RCS pressure increased and RPS initiated a reactor trip. Operators entered EOP-02, "Vital System Status Verification," to ensure automatic safety systems had actuated properly and to verify balance of plant system status. In addition, based on low levels in the steam generators, EFIC automatically initiated EFW. The FW pumps [SJ, P] were manually tripped approximately 55 minutes after the reactor trip in accordance with EOP-10, "Post-Trip Stabilization." Reactor operators properly executed the EOPs for plant shutdown.

Based on the above discussion, FPC concludes that actuation of the RPS and EFW did not represent a reduction in the public health and safety. This event does not meet the Nuclear Energy Institute definition of a Safety System Functional Failure (NEI 99-02).

CAUSE

The cause for this event was inadequate human performance in the implementation of the ICS circuit card refurbishment program which resulted in an age-related failure of the zener diodes in the +15 volt regulatory circuit for a Bailey 820 Control Module in the ICS.

On March 24, 2004, the plant tripped due to the failure of ICS Module 3-8-4 [JA, IMOD]. The resulting corrective action plan installed refurbished multipliers in four critical ICS locations and established a refurbishment program for all ICS modules. In November 2005, the refurbished replacement for ICS circuit card IC-384-IC failed calibration. This was discussed between the maintenance technician and the system engineer during turnover. The system engineer decided to install a non-refurbished multiplier card into the IC-384-IC since no refurbished spares were available at the time. The system engineer was fully aware of past actions to use only refurbished cards and he intended to ensure follow up activities. However, the turnover of this decision was incomplete in that supervisory personnel were not notified, and no follow up actions were taken to ensure that appropriate compensatory measures were established to install a refurbished multiplier card at the first available opportunity.

CORRECTIVE ACTIONS

1. CR-3 Administrative Instruction AI-704, "Reactor Trip Review and Analysis," was performed.

2. ICS Module IC-384-IC was replaced with a refurbished module.

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3. An extent of condition review was performed. The deficiency on the original card was a degraded +15 volt power regulating circuit on the multiplier module. Three other multiplier cards were identified in critical applications that contained the same power regulating circuit that could potentially degrade in a similar way. All three cards were inspected and found to be acceptable.

4. An investigation concerning MSV-47 lifting prior to reaching the minimum setpoint is being addressed in the CR-3 Corrective Action Program under Nuclear Condition Report (NCR) 223478.

5. An investigation of the unexpected loss of equipment that occurred during the transfer to the startup transformer following the reactor trip is being addressed under NCR 223456.

6. An investigation of the water hammer event in the heater drain piping is being addressed by NCR 223345.

7. Other actions associated with this event are being addressed in the CR-3 Corrective Action Program in Nuclear Condition Report 223337.

PREVIOUS SIMILAR EVENTS

LER 50-302/2004-001-00 documents a previous similar event at CR-3, "Reactor Trip Caused by Failed Circuit Board in the Main Feedwater Integrated Control System." The cause of this plant trip was age related circuit card failure.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms

Attachment 2 - List of Commitments

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

| | |
|---|---|
| AI | Administrative Instruction |
| ATWS | Anticipated Transient Without Scram |
| AMSAC | ATWS Mitigation System Actuation Circuitry |
| CFR | Code of Federal Regulations |
| CR-3 | Crystal River Unit 3 |
| EFIC | Emergency Feedwater Initiation and Control |
| EFW | Emergency Feedwater System |
| EOP | Emergency Operating Procedure |
| FPC | Florida Power Corporation |
| FW | Main Feedwater |
| HDV | Heater Drain Valve |
| ICS | Integrated Control System |
| LER | Licensee Event Report |
| MSV | Main Steam Valve |
| NCR | Nuclear Condition Report |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| RCS | Reactor Coolant System |
| RPS | Reactor Protection System |

NOTES: Improved Technical Specifications defined terms appear capitalized in LER text {e.g., MODE 1}

Defined terms/acronyms/abbreviations appear in parenthesis when first used {e.g., Reactor Building (RB)}.

EIIS codes appear in square brackets {e.g., reactor building penetration [NH, PEN]}.

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ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by FPC in this document. Any other actions discussed in the submittal represent intended or planned actions by FPC. They are described for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing & Regulatory Programs, of any questions regarding this document or any associated regulatory commitments.

| RESPONSE SECTION | COMMITMENT | DUE DATE |
|---|---|---|
| | No regulatory commitments are being made in this submittal. | |