Progress Energy

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

Ref: 10 CFR 50.73

November 30, 2005
3F1105-02

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

Subject: LICENSEE EVENT REPORT 50-302/2005-003-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/2005-003-00. The LER discusses a manual reactor trip and subsequent Emergency Feedwater System actuation due to loss of the only operating condensate pump during shutdown for a refueling outage. 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/dwh

Enclosure

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

IE22

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

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
Manual Reactor Trip and Subsequent Emergency Feedwater Actuation Due To Condensate Pump Loss

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
10	29	2005	2005	- 003 -	00	11	30	2005	N/A	05000

9. OPERATING MODE: 1

10. POWER LEVEL: 16%

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)
Dennis W. Herrin – Lead Engineer (Licensing & Regulatory Programs)	(352) 563-4633

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
D	SD	CPLG	E120	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 approximately 01:26, on October 29, 2005, Progress Energy Florida, Inc., Crystal River Unit 3, was in MODE 1 (POWER OPERATION) at approximately sixteen percent RATED THERMAL POWER when a manual reactor trip was initiated in anticipation of an Emergency Feedwater System actuation. The plant was being shutdown for a refueling outage when the only operating Condensate Pump motor tripped on electrical overload and resulted in a loss of both Main Feedwater Pumps. The second Condensate Pump had been shutdown by procedure and removed from service for maintenance. The cause for loss of the operating Condensate Pump was insufficient guidance and information available to the Reactor Operators to prevent operation at critical speeds. The Condensate Pump motor/magnetic coupling/pump assembly has been replaced. Procedural guidance is being developed to ensure operation at critical speeds is avoided. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A). The Reactor Protection System and Emergency Feedwater System performed as designed. This condition does not represent a reduction in the public health and safety. No previous similar occurrences have been reported to the NRC.

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

EVENT DESCRIPTION

At approximately 01:26, on October 29, 2005, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR-3) was operating in MODE 1 (POWER OPERATION) at approximately sixteen (16) percent RATED THERMAL POWER when a manual reactor trip was initiated in anticipation of an Emergency Feedwater System (EFW) [BA] actuation. At the time, CR-3 was being shutdown for a refueling outage in accordance with Operating Procedure OP-209A, "Refueling Outage Plant Shutdown and Cooldown."

At 17:54 and 23:04, on October 28, 2005, Condensate Pump CDP-1A [SD, P] and Feedwater Booster Pump FWP-1B [SJ, P] were shutdown in accordance with OP-209A, respectively. CDP-1A was subsequently taken out of service for maintenance. Both Main Feedwater Pumps (FWP-2A and FWP-2B) [SJ, P] were being supplied by CDP-1B [SD, P] and FWP-1A [SJ, P], with CDP-1B in manual control. (The CDP motors run at a constant speed. Pump flow rate is controlled by varying the magnetic coupling between the pump and the motor.)

At 00:46, on October 29, 2005, a valve lineup required by OP-209A was completed that resulted in the Deaerator Feed Tank (DFT) [SJ, DEA] level being controlled by manually adjusting the speed of CDP-1B. (The DFT acts as a head tank, supplying water to the suction of the Feedwater Booster Pumps.) Shortly after entering this mode of level control, Reactor Operators (ROs) noticed that condensate flow [SD] and DFT level were slowly increasing. The ROs began slowly decreasing CDP-1B speed and continued to make this adjustment periodically for the duration of this event.

Between 00:58 and 01:09, CDP-1B speed and flow spiked three times. On the third occurrence, ROs took the speed controller to minimum demand but CDP-1B did not respond. At 01:09, on October 29, 2005, the CDP-1B motor tripped on electrical overload.

Tripping of the CDP-1B motor resulted in a total loss of condensate flow to the DFT. This condition left the ROs with approximately twenty minutes of DFT inventory before automatic actions would trip both Main Feedwater Pumps and initiate subsequent automatic anticipatory reactor and turbine trips. CDP-1A could not be restored to an operable status in time to recover DFT level.

Abnormal Procedure AP-510, "Rapid Power Reduction," was entered. ROs took action to reduce reactor power within the limits of the EFW System and set action triggers to trip the reactor at the four foot level in the DFT and to start the diesel driven Emergency Feedwater Pump (EFP-3) [BA, P] and trip the Main Feedwater Pumps at three feet in the DFT. ROs reduced power below where the Reactor Protection System (RPS) [JC] would have initiated an automatic anticipatory reactor and turbine trip signal. At approximately 01:26, on October 29, 2005, ROs initiated a manual reactor trip to avoid an EFW System actuation with the reactor critical.

At 01:30, on October 29, 2005, FWP-1B tripped on low level in the DFT. Both Main Feedwater Pumps tripped and an Emergency Feedwater Initiation and Control System [JB] actuation signal started EFP-3 and steam driven Emergency Feedwater Pump EFP-2 [BA, P].

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With the exception of CDP-1A having been taken out of service for planned maintenance, no structures, systems or components were inoperable at the start of the event that contributed to the event. No other pertinent maintenance or surveillance activities were in progress. Plant systems operated normally during the manual reactor trip and subsequent EFW System actuation, with exception of the following:

Both EFP-3 and EFP-2 started, but the EFP-2 auto-start alarm failed. Differences were noted in EFP-2 and EFP-3 flow responses between the A and B Once Through Steam Generators (OTSGs) [AB, SG]. This appears to be related to calibration tolerances for B OTSG level control for EFP-2 and EFP-3. Proper OTSG levels were maintained during EFW operation.

Post-trip, the Condensate System had backflow from the DFT to the condenser hotwell [SD, COND], causing water to flash to steam. This continued until Condensate System valve CDV-36 [SD, V] was manually closed.

The Main Steam Safety Valve (MSSV) [SB, RV] monitor in the Main Control Room was obscured by EFP-2 exhaust steam, making it difficult to verify that the MSSVs had reseated.

Manual actuation of the RPS and automatic actuation of the EFW System are reportable to the NRC. At 04:39, on October 29, 2005, a non-emergency four-hour notification and a non-emergency eight-hour notification was made to the NRC Operations Center (Event Number 42094) in accordance with 10CFR50.72(b)(2)(iv)(B)(1) and 10CFR50.72(b)(3)(iv)(B)(6), respectively. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

SAFETY CONSEQUENCES

Manual actuation of the RPS and automatic EFW System actuation occurred to shut down the reactor and maintain adequate OTSG levels. Upon initiation of the manual reactor trip, the RPS responded as expected, control rods fully inserted and safety systems functioned as required. No challenges to the RPS were identified. Loss of both main FWPs resulted in an automatic EFW actuation. The diesel driven EFP-3 and steam driven EFP-2 started to provide flow to the OTSGs. Proper OTSG levels were maintained during EFW operation.

The event did not result in the release of radioactive material. No design safety limits were exceeded and no fission product barriers or components were damaged as a result. The loss of Main Feedwater is an event analyzed and bounded by the Final Safety Analysis Report accident analysis.

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Based on the above discussion, PEF concludes that the RPS and EFW System performed as designed and did not represent a reduction in the public health and safety. Since no loss of safety function occurred, this event does not meet the Nuclear Energy Institute definition of a Safety System Functional Failure (NEI 99-02, Revision 2).

CAUSE

The most likely cause for loss of CDP-1B was failure of the motor magnetic coupling due to pump operation in the critical speed range. CDP-1B was operated in the critical speed range because insufficient guidance and information were available to the ROs to prevent operation of CDP-1B at this speed. ROs were not aware that plant operation at low power resulted in operation of the condensate pumps near the critical speed. Operators are aware of the need to avoid critical speed but thought the condensate pumps would only approach critical speed during pump startup.

CDP-1B critical speed occurs at 413 to 415 revolutions per minute (rpm). OP-209A cautions ROs to minimize operation between 300 and 600 rpm to avoid the critical frequency region. However, the procedure is silent on when to measure the speed or determine if the critical frequency has been successfully avoided.

The CDP-1B motor magnetic coupling is a Model Ampli-Speed Magnetic Drive manufactured by Electric Machinery Manufacturing Company/McGraw Edison. The CDP-1B motor Ampli-Speed Magnetic Drive Serial Number is 193331211.

The following information represents the most likely CDP-1B motor magnetic coupling failure scenario:

By 00:58, on October 29, 2005, CDP-1B speed was believed to be at or near critical at 413 rpm using pump curves. This generated high vibration with large scale deflections and dynamic loads which momentarily disrupted the speed signal from the magnetic sensor at the toothed wheel mounted on the clutch's output shaft. The lost speed signal caused the CDP-1B speed to spike high and appear as an upward flow spike to the ROs. The high dynamic loads also progressively fractured the sixteen 3/8 inch diameter cap screws attaching the lubricating oil reservoir cover to the thrust stand. The low speeds and high dynamic loads result in high temperatures in the lower and upper magnetic coupling bearings. Ultimately the high temperatures result in the magnetic coupling bearings failing. The heat from the upper magnetic coupling bearing is radiated to the lower motor bearing which begins to heat up.

At 01:07, on October 29, 2005, there was another momentary disruption in the signal which again appeared as upward spike in condensate flow to the RO.

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At 1:09:05, the CDP-1B speed signal was completely lost when the shaft sleeve containing the toothed wheel finally vibrated far enough down the shaft to be out of the magnetic sensor's range. The controller saw this as a zero speed signal and within a few seconds drove the pump speed up until it nearly matched the motor speed of 1196 rpm. The condensate flow went up to approximately 5.4 x 10(+6) pounds mass per hour. Quadrupling the speed with no radial restraint for the thrust bearing intensified the vibration and dynamic loads. Continued high speed operation produced additional mechanical damage and the pump speed drew closer to motor speed as binding developed in the magnetic coupling pilot bearings.

At 1:09:37, the increasing motor current reached the over current alarm point (115 percent of rated current). The combination of high speed with failed magnetic coupling bearings caused the overcurrent condition that resulted in the CDP-1B motor trip.

The cause for this event applies to large, variable speed pieces of rotating equipment that have critical speeds within their operating range. CDP-1A, CDP-1B, and the Main Turbine Generator are the only items at CR-3 that fit in this category. The Main Turbine Generator has installed vibration monitoring equipment and is closely monitored, especially on start-ups and shutdowns.

CORRECTIVE ACTIONS

1. CDP-1B motor/magnetic coupling/pump assembly replacement and other actions associated with this event are being addressed in the CR-3 Corrective Action Program under Nuclear Condition Report (NCR) 174442.

2. Determining EFP-2 feed response to the B OTSG is being addressed under NCR 175040.

3. Determining Condensate System flashing effects on the plant system and evaluating if having the Condensate System aligned through CDV-36 is a potential water hammer concern is being addressed under NCR 174448.

4. Determining the MSSV monitor resolution is being addressed under NCR 174454.

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PREVIOUS SIMILAR EVENTS

Based on a keyword search of the CR-3 electronic Licensee Event Report database, no previous similar events involving failure of a Condensate Pump have been reported to the NRC by CR-3.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms
Attachment 2 - List of Commitments

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

AP	Administrative Procedure
CDP	Condensate Pump
CDV	Condensate System Valve
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
DFT	Deaerator Feed Tank
EFP	Emergency Feedwater Pump
EFW	Emergency Feedwater System
FWP	Feedwater System Pump
FWV	Feedwater System Valve
MSSV	Main Steam Safety Valve
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
OTSG	Once Through Steam Generator
PEF	Progress Energy Florida, Inc.
RO	Reactor Operator
rpm	Revolutions Per Minute
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 PEF in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. 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.	