Progress Energy

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

Ref: 10 CFR 50.73

March 20, 2009
3F0309-02

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

Subject: LICENSEE EVENT REPORT 50-302/2009-001-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/2009-001-00. The LER discusses a manual reactor trip due to loss of 'A' 4160 volt Unit Bus loads caused by incorrectly connected test leads. 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. Dan Westcott, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

J. W. Holt

James W. Holt
Plant General Manager
Crystal River Nuclear Plant

JWH/dwh

Enclosure

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

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
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Crystal River, FL 34428

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NLR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 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.

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4. TITLE
Manual Reactor Trip Due To Loss of A 4160V Unit Bus Loads Caused By Incorrectly Connected Test Leads

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 05000
01	27	2009	2009	- 001 -	00	03	20	2009	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 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)
10. POWER LEVEL	☐ 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)
100%	☐ 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 and 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

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 10:17, on January 27, 2009, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR-3), was operating in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER when the Reactor Coolant System pressure was observed rising toward the automatic reactor trip setpoint and the reactor was manually tripped. Prior to this event, technicians were performing preventive maintenance on the 'A' 4160 volt Unit Bus indication located in the Main Control Board. Doble Relay Test Set leads were incorrectly connected in place of the Fluke Multimeter leads during a voltage check. This resulted in a path to ground that blew the 'A' and 'B' Secondary Side potential transformer fuses, resulting in a loss of voltage indication on the 'A' 4160 volt Unit Bus and tripping loads. The Reactor Protection System responded as expected to the manual trip signal, control rods fully inserted and safety systems functioned as required. No reduction in the public health and safety was created. The causes for the manual reactor trip were improper use of human performance tools and inadequate closure of corrective actions for a previous similar event. All relay work was immediately stopped and a Human Performance Review Board Meeting was held. This report is submitted pursuant to 10CFR50.73(a)(2)(iv)(A). A previous similar occurrence was documented in Nuclear Condition Report (NCR) 133661.

EVENT DISCRIPTION

At 10:17, on January 27, 2009, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR-3) was operating in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER when the Control Room staff received multiple alarms and observed Reactor Coolant System (RCS) [AB] pressure rising toward the automatic reactor trip setpoint [JD]. The reactor [AC] was manually tripped prior to reaching the automatic reactor trip setpoint. Emergency Operating procedure EOP-2, "Vital System Status Verification," was entered before eventually transitioning to EOP-10, "Post-Trip Stabilization."

Prior to this event, two experienced relay technicians from the Progress Energy Florida Substation Maintenance Department were tasked with performing Work Order Task 1072772-01 utilizing Preventive Maintenance (PM) procedure PM-290, "Calibration of Switchboard Meters and Transducers." While setting up to perform calibration on the 'A' 4160 volt (V) Unit Bus [EB, BU] indication in the Main Control Board, a relay technician was verifying voltage on the 'A' 4160V Unit Bus potential transformer (PT) [EB, XPT] prior to opening the links and connecting the Doble Relay Test Set. This is accomplished by using a Fluke Multimeter with blue and yellow test leads. (The Fluke Multimeter blue and yellow test leads were a result of the corrective actions established by Nuclear Condition Report (NCR) 133661. At all other Progress Energy plants, the Fluke Multimeter test leads are red/black.)

The black and red test leads from the Doble Relay Test Set were incorrectly connected in place of the Fluke Multimeter blue and yellow test leads during performance of a live-dead-live voltage check. This resulted in a path to ground that blew the 'A' and 'B' Secondary Side PT fuses [EB, FU] on the 'A' 4160V Unit Bus and a loss of voltage indication to the 'A' 4160V Unit Bus loads. This produced the actuation of undervoltage motor protection relays [EB, 27], tripping of the 4160V loads attached to the 'A' 4160V Unit Bus (e.g., Circulating Water System (CW) [KE] pumps CWP-1A/1C [KE, P], Main Feedwater System (FW) [SJ] pump FWP-1A [SJ, P], Condensate System (CD) [SG] pump CDP-1A [SG, P], Secondary Services Closed Cycle Cooling System (SC) [KB] pump SCP-1A [KB, P], etc). Nuclear Services and Decay Heat Raw Water System (RW) [KI] pump RWP-2B [KI, P] and SCP-1B automatically started per design.

Upon initiation of the manual reactor trip, the main turbine [TA] automatically tripped and the 'A' 4160V Unit Bus transferred from the Unit Auxiliary Transformer [EB, XMFR] to the Startup Transformer per design. The 'A' 4160 Unit Bus and the 'A' 480V Bus remained energized.

At 11:03, on January 27, 2009, the MODE 3 (HOT STANDBY) lineup was established. At 12:49 on January 27, 2009, EOP-10 was exited.

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 protection and non-protection systems operated normally during the manual reactor trip, with the exception of the following:

FWP-2B suction line relief valve FWV-16 [SJ, RV] lifted and did not reseat.

Heater Drain System piping [SN, PSP] above Condensate System heat exchanger CDHE-3B [SD, HX] experienced a water hammer.

Check valve RWV-36 [KI, V] did not seat and allowed RWP-1 to reverse rotate.

Manual actuation of the Reactor Protection System (RPS) [JD] is reportable to the NRC. At 12:41, on January 27, 2009, a non-emergency four-hour notification was made to the NRC Operations Center (Event Number 44807) in accordance with 10CFR50.72(b)(2)(iv)(B). This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

SAFETY CONSEQUENCES

Manual actuation of the RPS occurred to shut down the reactor while the main FW System maintained adequate Once-Through Steam Generator (OTSG) [SB, SG] 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 setpoints were identified. Both Main Feedwater Pumps remained in operation throughout this event. No Emergency Feedwater Initiation and Control (EFIC) System [JB] actuation occurred or was required.

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 manual reactor trip is bounded by the Final Safety Analysis Report accident analysis.

Based on the above discussion, PEF concludes that the RPS 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 (NEI) definition of a Safety System Functional Failure (NEI 99-02, Revision 2).

CAUSE

Two causes have been identified for the need to manually trip the reactor. The first cause was improper use of human performance tools which led to the connection of the incorrect test leads to the 'A' 4160V Unit Bus. The three causal factors for this human error were: 1) self-checking not applied to ensure correct intended action; 2) failure to effectively use peer checking; and, 3) procedure use and adherence failure (i.e., performance of the voltage check outside of the work order guidance).

The second cause was inadequate closure of corrective actions for a similar event that occurred at CR-3 in 2004 (Priority 2 NCR 133661). The CR-3 Plant Nuclear Safety Committee (PNSC) established an action to: "Add a corrective action to complete a risk assessment for this type of work (calibrating volt meters on the main control board), whether it should be completed on-line vs. off-line." The response to this corrective action was not adequate and was not reviewed by the PNSC or the PNSC Chairman. This resulted in the failure to move relay activities that could result in a plant transient from on-line to outage.

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CORRECTIVE ACTIONS

1. FWV-16 was manually reseated. NCR 316546 was initiated.
2. A walkdown was performed on the piping above CDHE-3B. No damage from the water hammer was identified. NCR 316552 was initiated.
3. RWV-36 was isolated. NCR 316682 was initiated.
4. A Human Performance Review Board Meeting was held on this event.
5. All relay work was stopped immediately until a more detailed risk assessment of work activities could be performed.
6. Additional corrective actions are identified in NCR 316543.

PREVIOUS SIMILAR EVENTS

Although not previously reported to the NRC, a similar occurrence in 2004 was documented in NCR 133661. In that event, only one secondary side PT fuse blew on the 'A' 4160V Unit Bus. Two of three fuses are needed to blow in order to result in actuation of undervoltage relays and loss of voltage indication to the 'A' 4160V Unit Bus loads.

ATTACHMENTS

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

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Attachment 1

Abbreviations, Definitions, and Acronyms

CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
CD	Condensate System
CDP	Condensate System Pump
CDHE	Condensate System Heat Exchanger
CW	Circulating Water System
CWP	Circulating Water System Pump
EFIC	Emergency Feedwater Initiation and Control System
EOP	Emergency Operating Procedure
FW	Main Feedwater System
FWP	Main Feedwater Pump
FWV	Main Feedwater Valve
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OTSG	Once-Through Steam Generator
PEF	Progress Energy Florida, Inc.
PM	Preventive Maintenance
PNSC	Plant Nuclear Safety Committee
RCS	Reactor Coolant System
RPS	Reactor Protection System
RW	Nuclear Services and Decay Heat Raw Water System
RWP	Nuclear Services and Decay Heat Raw Water System Pump
RWV	Nuclear Services and Decay Heat Raw Water System Valve
SC	Secondary Services Closed Cycle Cooling System
SCP	Secondary Services Closed Cycle Cooling System Pump
V	Volt

NOTES: Improved Technical Specification 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 by PEF in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	DUE DATE
No new regulatory commitments are contained in this submittal.	N/A