Progress Energy

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

Ref: 10 CFR 50.73

March 9, 2007
3F0307-01

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

Subject: CRYSTAL RIVER UNIT 3 - LICENSEE EVENT REPORT 50-302/2007-001-00

Dear Sir:

Florida Power Corporation, currently doing business as Progress Energy Florida, Inc., hereby submits Licensee Event Report (LER) 50-302/2007-001-00. The LER discusses two conditions discovered during performance of the CR-3 Safe Shutdown Analysis Revalidation Project Fire Area Assessment where 10 CFR 50, Appendix R, Section III.G.2 cable separation criteria were not met. Both conditions are being reported in this LER since they involved the same two end devices, the conditions are related (i.e., they have the same general cause or consequences) and they occurred during a single activity over a reasonably short time (60 days for LER reporting). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B).

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

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

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
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: 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
CRYSTAL RIVER UNIT 3	05000302	1 OF 8

4. TITLE
Design Oversight Results In 10 CFR 50, Appendix R, Cable Separation Criteria Not Being Met

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
01	11	2007	2007	- 001 -	00	03	09	2007	N/A	05000
									N/A	05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

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 – 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

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 19:01, on January 11, and at 13:16, on January 23, 2007, Progress Energy Florida, Inc., Crystal River Unit 3 (CR-3) was operating in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER when conditions not meeting 10CFR50, Appendix R, Section III.G.2 cable separation criteria were identified for high/low pressure interface valves. During performance of the CR-3 Safe Shutdown Analysis Revalidation Project Fire Area Assessment for the Intermediate and Reactor Buildings, respectively, reviews revealed that the power cable for Decay Heat Removal (DH) System valve DHV-3 and the power and control cables for DHV-4 were routed with other energized cables. Low probability three-phase external cable hot shorts of the proper voltage due to a hypothetical fire could cause spurious opening of both valves, resulting in an unanalyzed loss of coolant condition in the Auxiliary Building. The cause for this event was a misunderstanding of 10CFR50, Appendix R cable separation criteria in 1997 and 2002 pertaining to high/low pressure interfaces resulting in missed opportunities to correct the identified conditions. Appropriate compensatory measures have been put into place. This report is being submitted pursuant to 10CFR50.73(a)(2)(ii)(B). This condition does not represent a reduction in the public health and safety. Previous similar occurrences have been reported to the NRC.

1. FACILITY NAME	2. DOCKET NUMBER (2)	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000302	2007	- 001	- 00	2 OF 8

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

EVENT DESCRIPTION

At 19:01, on January 11, and at 13:16 on January 23, 2007, 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 conditions not meeting 10CFR50, Appendix R, Section III.G.2 cable separation criteria were identified for high/low pressure interface valves. During performance of the CR-3 Safe Shutdown Analysis Revalidation Project Fire Area Assessment for the 119' elevation of the Intermediate Building [NF] and for the 95' elevation inside the Reactor Building [NH], respectively, reviews of cable routing data revealed that the power cable [BP, CBL4] for Decay Heat Removal (DH) System valve DHV-3 [BP, ISV] and the power and control cables [BP, CBL3] for DHV-4 were routed with other energized cables. Low probability three-phase external cable hot shorts of the proper voltage due to a hypothetical fire could cause spurious opening of DHV-3 and DHV-4. A three-phase hot short is a cable to cable fault that aligns the three phases of an energized cable with the three phases of a de-energized cable; therefore, energizing the high/low pressure interface device. A cable to cable hot short on the DHV-4 control cable could also cause DHV-4 to spuriously open.

10CFR50, Appendix R, Section III.G.2 states: “Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:” Subsections a, b and c of Section III.G.2 contain the specific criteria.

10CFR50, Appendix R, Section III.G.2 continues by stating: “Inside noninerted containments one of the fire protection means specified above or one of the following fire protection means shall be provided:” Subsections d, e and f of Section III.G.2 contain the additional specific criteria.

In the Intermediate Building (Fire Area IB-119-201), the DHV-3 power cable was previously rerouted in a dedicated conduit to prevent a three-phase hot short from spuriously opening the valve. No rerouting of the DHV-4 power or control cables was required due to the rerouting performed on the DHV-3 power cable. A recent walkdown of the DHV-3 circuit revealed that the field installation for that modification did not document a short section of the cable run. This resulted in the DHV-3 power cable being routed with other energized three-phase power cables. These circuits have been reviewed and found to have sufficient energy to reposition the DHV-3 valve in the unlikely event of a three-phase hot short. Therefore, 10CFR 50, Appendix R, Section III.G.2 cable separation requirements were not met in this area.

For the condition in the Reactor Building (Fire Area RB-95-301), no radiant energy shield separates the DHV-3 and DHV-4 power cables on the north side. There appears to be greater than 20 feet of separation between the power cables. However, there are other cable tray sections between the affected cables which constitute intervening combustibles. Therefore, separation requirements are considered not met in this area.

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

On the south and west sides of the Reactor Building, no radiant energy shield exists between a DHV-4 control cable and a DHV-3 power cable. There are intervening combustibles in the form of cable trays throughout the routing of both cables. In this area of the Reactor Building, there appears to be less than 20 feet of separation. Therefore, separation requirements are considered not met in this area.

During normal shutdown operations, reactor coolant enters the DH System through a common suction header from the "B" Reactor Coolant System (RCS) [AB] hot leg (decay heat drop line). The reactor coolant flows through motor operated isolation valves DHV-3 and DHV-4 (inside containment) and DHV-41 (outside containment). The common suction header supplies the running DH System pump (DHP-1A or DHP-1B) [BP, P] through DHV-39 or DHV-40. DHV-3 is the inboard motor operated isolation valve between the RCS and DH System and is designed for 2500 pounds per square inch gauge (psig) and 650 degrees Fahrenheit. DHV-4 is the outboard isolation valve between DHV-3 and DHV-41 and is designed for 2500 psig and 650 degrees Fahrenheit. Piping between DHV-4 and DHP-1A/1B is designed for 345 psig and 300 degrees Fahrenheit.

During power operation, DHV-3 and DHV-4 are normally closed to isolate the decay heat drop line from the RCS hot leg. Power is removed from DHV-3 to prevent spurious opening in the event of a fire (Appendix R concern). This action prevents inadvertent DH System overpressurization. If both DHV-3 and DHV-4 were to spuriously open during power operation, the potential exists to overpressurize and rupture portions of the DH System piping between DHV-4 and DHV-41 outside the Reactor Building. A loss of coolant condition in the Auxiliary Building [NF] is an unanalyzed condition.

At 17:19 on January 12, 2007, a notification was made to the NRC Operations Center (Event Number 43098) of a condition reportable under 10CFR50.72(b)(3)(ii)(B). At 16:27 on January 23, 2007, a notification was made to the NRC Operations Center (Event Number 43116) of a condition reportable under 10CFR50.72(b)(3)(ii)(B). This Licensee Event Report addresses both conditions and is submitted pursuant to 10CFR50.73(a)(2)(ii)(B).

SAFETY CONSEQUENCES

For a three-phase hot short to occur that could cause a high/low pressure interface valve to reposition to the undesired position (open), the three-phase cabling for the high/low pressure interface valve would have to align perfectly with the three phases of the aggressor power cable with sufficient energy in the same raceway to reposition the high/low pressure interface valve. This would have to occur downstream of the motor control center [ED, MCC]. This aggressor cable would have to be supplying a load of sufficient magnitude such that the overcurrent protective relaying (specifically, the time overcurrent feature) would not trip when the interface valve motor initially started running.

1. FACILITY NAME	2. DOCKET NUMBER (2)	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000302	2007	- 001	- 00	4 OF 8

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

The high/low pressure interface valve cabling conductors, as well as the aggressor's conductors, could not be shorted to ground or shorted to each other at any time. Since three-phase cabling is typically in a triplex configuration (three cables, each separately insulated, wound around each other), for three hot shorts to occur, the insulation would have to be broken down sufficiently on all three hot shorts in both cables such that a direct short would occur. However, the rest of the cables would have to be insulated sufficiently such that any other area of insulation breakdown would not result in a ground or a short to any of the other conductors within the cables. This combination of circumstances is extremely improbable.

NUREG/CR-6850, EPRI (Electric Power Research Institute)/NRC-RES Fire PRA (Probabilistic Risk Assessment) Methodology for Nuclear Power Facilities, considers three-phase hot shorts as not risk-significant and contributes a Core Damage Frequency (CDF) of less than 1E-7 per year. The conservative annual CDF of fires leading to an interlacing loss of coolant accident by spurious opening of both DHV-3 and DHV-4 is calculated by multiplying the total damaging fire frequency with the cable damage probability. That value is 8E-8 per year. The actual CDF is expected to be several orders of magnitude lower than that value. The overall risk significance of this hypothetical fire scenario is extremely low.

Based on the above discussion, PEF concludes that the identified conditions did not represent a reduction in the public health and safety.

The identified condition is not reportable under 10CFR50.73(a)(2)(v) and does not represent a condition that would have prevented the fulfillment of a safety function. Therefore, this event does not meet the Nuclear Energy Institute (NEI) definition of a Safety System Functional Failure (Reference NEI 99-02, Revision 2).

CAUSE

The cause for this event was misunderstanding of 10CFR50, Appendix R cable separation criteria pertaining to high/low pressure interface requirements, resulting in missed opportunities to correct the identified conditions.

CR-3 identified two opportunities where DHV-3 power cable and DHV-4 power and control cable routing deficiencies could have been corrected to meet the 10CFR50, Appendix R, Section III.G.2 requirements. Individuals involved in those opportunities were thought to clearly understand the 10CFR50, Appendix R, Section III.G.2 criteria and associated clarifying documents (Generic Letter 81-12 and Generic Letter 86-10), the CR-3 Topical Design Basis Document for Appendix R requirements and CR-3 Electrical Design Criteria #5.

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

First, an independent assessment of 10CFR50, Appendix R implementation was performed in 1996. That assessment identified concerns related to spurious operation of DHV-3 and DHV-4 with respect to Generic Letter 86-10 clarifications. Problem Report (PR) 96-0401 was initiated to document this condition in the CR-3 Corrective Action Program. To achieve compliance with the Appendix R, Section III.G.2 requirements, CR-3 contracted an industry expert to perform the technical evaluation of the DHV-3 and DHV-4 separation issue. The technical evaluation concluded that both valves were susceptible to fire-induced three-phase hot shorts in the Intermediate Building and portions of the Reactor Building. The cable spreading room was not identified to be an area of concern. Modification Approval Record (MAR) 97-02-18-01 was prepared to reroute power cables for both valves.

Second, the CR-3 Nuclear Assessment Section (NAS) performed an assessment of the Fire Protection Program and Appendix R implementation in 2002. That assessment identified that the cable routing for DHV-3 and DHV-4 in the cable spreading room had not been evaluated for spurious actuation as defined in Generic Letter 86-10 clarifications. The content of MAR 97-02-18-01 was evaluated and confirmed to not address cable routing through the cable spreading room for DHV-3 and DHV-4. A root cause investigation performed under Nuclear Condition Report (NCR) 61855 concluded that PR 96-0401 and MAR 97-02-18-01 lacked rigor and did not adequately address the entire cable route of DHV-3 and DHV-4. Criteria from the NRC and industry interpretation of Generic Letter 86-10 for high/low pressure interfaces had not been applied. NCR 61855 focused on the cable spreading room and did not adequately evaluate the remaining fire areas for similar adverse conditions. The NCR 61855 extent of condition investigation should have reviewed the entire cable routing for DHV-3 and DHV-4 to identify additional areas where 10CFR50, Appendix R, Section III.G.2 requirements were not being met.

Note the conditions in the cable spreading room were later determined to be acceptable.

CORRECTIVE ACTIONS

1. For the condition identified on January 11, 2007, interim compensatory measures were put in place to: 1) establish a 1-hour roving fire patrol on the 119' elevation of the Intermediate Building in the area of the Personnel Hatch (affected area); 2) allow no hot work in the affected area; 3) limit combustibles in the affected area; and, 4) establish a continuous fire watch with compression capability if fire detection in the affected area is out of service.

2. For the condition identified on January 23, 2007, interim compensatory measures were put in place to: establish a 1-hour roving fire patrol to ensure no alarms exist in Fire Service Control Panel FSCP-12 (Module 1 Reactor Building). The affected area is the 95' elevation inside the Reactor Building.

3. Other actions associated with this event, including restoration of compliance to 10CFR50, Appendix R, Section III.G.2 cable separation requirements for DHV-3 and DHV-3 power and control cables, are being addressed in CR-3 Corrective Action Program NCR 218852.

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

PREVIOUS SIMILAR EVENTS

Although a number of Licensee Event Reports (LERs) have been submitted to the NRC by CR-3 related to Appendix R compliance issues, the following LERs reported problems associated with hot shorts:

LER 50-302/97-033-00
LER 50-302/96-022-01
LER 50-302/96-002-00
LER 50-302/95-025-02
LER 50-302/95-013-01
LER 50-302/95-012-00
LER 50-302/89-039-01

ATTACHMENTS

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

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

ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
DH	Decay Heat Removal System
DHP	Decay Heat Removal System Pump
DHV	Decay Heat Removal System Valve
EPRI	Electric Power Research Institute
FSCP	Fire Service Control Panel
LER	Licensee Event Report
MAR	Modification Approval Record
NAS	Nuclear Assessment Section
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRC-RES	NRC, Office of Nuclear Regulatory Research
NUREG/CR	Nuclear Regulation – Contractor Prepared
PEF	Progress Energy Florida, Inc.
PRA	Probabilistic Risk Assessment
PR	Problem Report
psig	pounds per square inch gauge
RCS	Reactor Coolant 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]}.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER (2)	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000302	2007	- 001	- 00	8 OF 8

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

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.	