Progress Energy

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

Ref: 10 CFR 50.73

January 17, 2011
3F0111-03

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

Subject: LICENSEE EVENT REPORT 50-302/2010-001-01

Reference: Crystal River Unit 3 (CR-3) to NRC letter, dated December 8, 2010, "LICENSEE EVENT REPORT 50-302/2010-001-00"

Dear Sir:

Florida Power Corporation, currently doing business as Progress Energy Florida, Inc., hereby submits Revision 1 to Licensee Event Report (LER) 50-302/2010-001-00 (Reference). The LER discusses the as-found lift setpoint for both Cycle 16 Pressurizer Code Safety Valves being outside the maximum tolerance allowed by the CR-3 Improved Technical Specifications. This revision incorporates the results of the completed cause evaluation for this condition. This condition is reportable under 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D).

No new regulatory commitments are made in this letter.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, 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
15760 W. Power Line Street
Crystal River, FL 34428

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NRR

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
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
As-Found Cycle 16 Pressurizer Code Safety Valve Setpoints Outside Improved Technical Specification Limit

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
10	13	2010	2010	- 001 -	01	01	17	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: No Mode

10. POWER LEVEL: 0%

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 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
X	AB	RV	D243	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)*

On September 1, 2010, and October 5, 2010, Progress Energy Florida, Inc., (PEF) Crystal River Unit 3 (CR-3) was in NO MODE (core off loaded) when Wyle Labs provided Notices of Anomaly for each of two Pressurizer Code Safety Valves (PCSVs) that had been removed during the Cycle 16 refueling outage (R16) and replaced with rotating spare PCSVs. The as-found lift setpoint for one PCSV was 5.32 percent above the Improved Technical Specification (ITS) setpoint and the other PCSV was 2.08 percent above the ITS setpoint. ITS 3.4.9 states that two PCSVs shall be operable in MODES 1, 2 and 3. To be operable, the PCSV lift setpoints must be within the maximum allowable tolerance of +/- 2 percent.. The existence of similar discrepancies in these relief valves is an indication that the discrepancies may have developed over a period of time. PEF concludes that both PCSVs were inoperable during plant operation and that the condition is reportable under 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(v)(D). This condition does not represent a reduction in the public health and safety. The selected cause is failure to manage vendor quality. CR-3 will issue a PCSV specification to include specific repair requirements, surveillance requirements, valve settings, documentation requirements and a steam test procedure obtained from Dresser Industries. Similar occurrences have not been previously reported to the NRC.

BACKGROUND

The Reactor Coolant System (RCS) [AB] forms a barrier against the release of reactor coolant and radioactive material to the Reactor Building [NH] or to the Main Steam System [SB]. Establishing a system pressure limit helps to assure the integrity of the RCS. The design pressure of the RCS is 2500 pounds per square inch gauge (psig). The maximum transient pressure of the RCS pressure vessel, RCS piping, valves and fittings is 110 percent of design pressure. Thus a safety limit of 2750 psig has been established for the RCS. Before initial plant operation, the RCS was hydrostatically tested at 3125 psig.

Normal RCS pressure control is by the pressurizer [AB, PZR] steam cushion in conjunction with the pressurizer spray and pressurizer heaters. The RCS is protected from overpressure by the Reactor Protection System [JC] features, such as the RCS high-pressure reactor trip, one Power-Operated Relief Valve (PORV) [AB, PCV], and the two pressurizer code safety valves (PCSVs) [AB, RV]. Because of these other protective features, it is unlikely that the PCSVs will ever lift during operation. RCS pressure setpoints for these features are as follows:

Pressurizer Code Safety Valves	2500 psig
Power-Operated Relief Valve	2450 psig
Reactor trips on high RCS pressure	2355 psig
RCS high pressure alarm	2255 psig
Pressurizer Spray Valve opens	2205 psig

The PCSVs protect the RCS against overpressurization during transients and accidents which involve a mismatch between the primary plant heat source and the secondary plant heat sink. Effluent from the PORV and PCSVs discharges to the Reactor Coolant Drain Tank [AB, TK].

Improved Technical Specification (ITS) 3.4.9 requires that both PCSVs be operable with a lift setting of 2500 psig +/- 2 percent (≥ 2450 psig and ≤ 2550 psig) in Modes 1, 2 and 3. When a PCSV is removed from the pressurizer for testing, it shall be reset to +/- 1 percent of the nominal setpoint.

Crystal River Unit 3 (CR-3) has four Model 31739A PCSVs manufactured by Dresser Industries with two in service during operation. PCSV testing is performed by Wyle Laboratories. During plant operation, two of the four valves are installed on the pressurizer as PCSVs (Reactor Coolant Valves (RCV)-8 and RCV-9) and the other two valves are spares. Hence, the individual valves "rotate" through their assignment as PCSVs on a once-per-fuel-cycle basis between tests. Both valves are removed at the end of each operating cycle, sent out for testing, and the two valves which had been tested and stored at the site since the previous cycle are installed on the pressurizer.

EVENT DESCRIPTION

On September 1, 2010, Progress Energy Florida, Inc., (PEF) CR-3 was in NO MODE (core off loaded) at 0 percent RATED THERMAL POWER when Wyle Laboratories provided a Notice of Anomaly for a PCSV (Serial Number BU-03149). This valve had been installed on the pressurizer as RCV-9 during Cycle 16 operation and was sent to Wyle Laboratories during the

Cycle 16 refueling outage (R16). The highest lift pressure recorded (2633 psig) was 5.32 percent above the ITS setpoint and 3.32 percent higher than the ITS maximum allowed "as-found" lift pressure. This condition was documented in the CR-3 Corrective Action Program as Nuclear Condition Report (NCR) 420022 on September 2, 2010.

The above condition was not considered to be reportable based on the guidance of NUREG-1022, Section 3.2.2, Example 3, "As discussed above, discrepancies found in technical specifications surveillance tests should be assumed to occur at the time of the test unless there is firm evidence, based on a review of relevant information (e.g., the equipment history and the cause of the failure) to indicate that the discrepancy occurred earlier." Relevant information at this time did not support a reportable condition.

On October 5, 2010, CR-3 was in NO MODE (core off loaded) at 0 percent RATED THERMAL POWER when Wyle Laboratories provided a Notice of Anomaly for a second PCSV (Serial Number BL-08899). This valve had been installed on the pressurizer as RCV-8 during Cycle 16 operation and was sent to Wyle Laboratories during R16. The highest lift pressure recorded (2552 psig) was 2.08 percent above the ITS setpoint and 0.08 percent higher than the ITS maximum allowed "as-found" lift pressure. This condition was documented in the CR-3 Corrective Action Program as NCR 426852 on October 13, 2010.

The above condition is considered to be reportable based on the further guidance of NUREG-1022, Section 3.2.2, Example 3, "However, the existence of similar discrepancies in multiple valves is an indication that the discrepancies may well have arisen over a period of time and the failure mode should be evaluated to make this determination. If so, the condition existed during plant operation and the event is reportable under § 50.73(a)(2)(i)(B)."

Valve	Serial Number	Set Pressure	Acceptable Range	As-Found Set Pressure	Result %
RCV-9	BU-03149	2500 psig	2450 - 2550	2633	+ 5.32
RCV-8	BL-08899	2500 psig	2450 - 2550	2552	+ 2.08

ITS 3.4.9 states that the two PCSVs shall be OPERABLE in MODES 1, 2 and 3. With one PCSV inoperable, restore the valve to an OPERABLE status within 15 minutes or be in MODE 3 within 6 hours and be in MODE 4 within 12 hours. With two PCSVs inoperable, be in MODE 3 within 6 hours and be in MODE 4 within 12 hours. Since CR-3 was in NO MODE when the PCSVs as-found lift setpoints were identified as being outside of the maximum allowable tolerance range, ITS 3.4.9 Required Actions A.1, B.1 and B.2, were not applicable.

Both PCSVs being inoperable during plant operation is a condition prohibited by the CR-3 ITS. This condition is reportable under 10CFR50.73(a)(2)(i)(B).

CAUSE

A conclusive "root" or "common" technical cause could not be identified following the evaluation by CR-3 and review of the PCSVs (BU-03149 and BL-08899) refurbishment and calibration reports. A "selected" cause (a causal factor that most likely describes the root cause of the event) was therefore identified.

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The selected cause identified for both PCSVs is failure to manage vendor quality. Vendor testing has not met expectations due to failure to provide a proper relief valve specification to the vendor, including a detailed testing procedure, repair plan and acceptance criteria. Currently, the specification consists of Progress Energy acceptance criteria for the relief valve that are followed by the vendor. The vendor uses their testing procedure and repair plan to achieve the acceptance criteria of the Progress Energy specification. Progress Energy has not provided sufficient guidance in the current mini-specification to ensure critical aspects of testing are specified.

A contributing factor is the current ITS 3.4.9 requirement that states that PCSVs removed from the pressurizer for testing shall be reset to +/- 1 percent of the nominal setpoint. Long-term storage conditions after rebuild and certification testing prior to installation creates the potential for setpoint drift. Additionally, the internal moving parts of the valve are not lubricated from the process fluid due to the lack of actuation during the operating cycle, causing the parts to adhere to each other. These factors result in a greater potential for initial as-found test failures to be high over the maximum setpoint pressure with the present as-left pressure acceptance criteria of +/- 1 percent of the nominal setpoint.

SAFETY CONSEQUENCES

The design pressure for the RCS is 2500 psig. Enhanced Design Basis Document Tab 6/1, "Reactor Coolant System," states the total PCSV capacity to be such that RCS pressure will not exceed 110 percent of system design pressure (2750 psig) to protect the RCS from exceeding the American Society of Mechanical Engineer (ASME) code safety limit. The set pressure of the PCSVs is +/- 2 percent (≥ 2450 psig and ≤ 2550 psig) of the lift setpoint (2500 psig) with a design capacity for each valve of 317,973 pounds mass per hour.

An engineering evaluation concluded that the credited protection criterion of the RCS not exceeding 110 percent of the ASME code allowable pressure could not be demonstrated to have existed over the last operating cycle (Cycle 16). Using straight line projection, the PCSV that had an as-found lift pressure of 2.08 percent above the ITS setpoint would have allowed an RCS peak pressure of approximately 2752.1 psig. The PCSV that had an as-found lift pressure of 5.32 percent above the ITS setpoint would have allowed an RCS peak pressure of approximately 2839.9 psig. Accidents that may be adversely impacted due to the PCSVs being set above the ITS setpoint are the Moderator Dilution Accident, the Startup Accident and the Loss of Feedwater Accident.

Also, the Feedwater Line Break Accident would likely exceed its currently assessed value of 110 percent of design pressure (2500 psig), or 2750 psig. However, as stated in Section 14.2.2.9.2 of the CR-3 Final Safety Analysis Report, the Feedwater Line Break Accident is considered a limiting fault. The acceptance criteria for a limiting fault includes RCS pressure not exceeding 125 percent of design pressure (2500 psig), or 3125 psig.

Although the RCS safety limit of 2750 psig may have been exceeded, the RCS was hydrostatically tested at 3125 psig prior to initial plant operation. This value would not have been exceeded during the accident scenarios identified above.

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Based on the above, the identified condition is reportable as a condition that could have prevented the fulfillment of the PCSV safety function to mitigate the consequences of an accident and is reportable under 10CFR50.73(a)(2)(v)(D). Since the identified condition could not have prevented the fulfillment of the PCSV safety function to mitigate the consequences of an accident at the time of discovery, it is not reportable under 10CFR50.72(b)(3)(v)(D).

The identified condition meets the definition of a Safety System Functional Failure as defined in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," since it is reportable under 10CFR50.73(a)(2)(v)(D).

A probabilistic safety assessment evaluation was performed for the identified condition. With either PCSV failing to open, the change in Core Damage Frequency (CDF) was less than 1e-06 (low safety significance). With both PCSVs failing to open, the change in CDF was approximately 5e-05 (high to moderate safety significance). However, failing both PCSVs is not representative of the identified condition. Using the change in CDF associated with one PCSV failing to open is more accurate, although still a very conservative bounding analysis, since the as-found condition is that the PCSVs opened late instead of failing to open. The overall conclusion is that the identified condition is of low safety significance.

Based on the above, PEF concludes that the inoperable condition of RCV-8 and RCV-9 did not represent a reduction in the public health and safety.

CORRECTIVE ACTIONS

Purchase Order 00494187 has been revised to require the as-left pressure acceptance criteria for valves BU-03149 and BL-08899 to be + 0/- 1 percent of the nominal setpoint.

Progress Energy engineering source surveillance has been completed for re-testing valves BU-03149 and BL-08899 to verify the as-left pressure acceptance criteria of + 0/- 1 percent of the nominal setpoint.

Additional corrective actions developed as part of the root cause evaluation that are being tracked in the CR-3 Corrective Action Program under NCR 426852 include, but are not limited to:

Replace the currently installed PCSVs with recently refurbished valves BU-03149 and BL-08899 that have an as-left pressure acceptance criteria of + 0/- 1 percent of the nominal setpoint. This will occur prior to entering MODE 3 from the current extended refueling outage.

Revise Catalog IDs 66081638 (valves BU-03149 and BL-08900) and 66081640 (valves BU-03148 and BL-08899) to include the revised as-left pressure acceptance criteria, and other administrative repair/test detail to serve as an interim process until an Engineering Change (EC) is issued.

Obtain the services of Dresser Industries to create a test procedure for steam testing the PCSVs to meet Progress Energy standards.

Issue a PCSV specification under the EC process to include specific repair requirements, surveillance requirements, valve settings, documentation requirements and the test procedure obtained from Dresser Industries.

ADDITIONAL INFORMATION

All four PCSVs (two installed, two rotational spares) are Model 2-1/2-31739A-1 closed bonnet maxiflow valves manufactured by Dresser Industrial Valve & Instrument Division.

PREVIOUS SIMILAR EVENTS

Previous occurrences of PCSV setpoints being found outside their required tolerance have not been reported by CR-3 to the NRC in a LER.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms

Attachment 2 - List of Commitments

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

ASME	American Society of Mechanical Engineers
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
EC	Engineering Change
ITS	Improved Technical Specifications
LER	Licensee Event Report
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
NUREG	NRC Nuclear Regulation
PCSV	Pressurizer Code Safety Valve
PEF	Progress Energy Florida, Inc.
PORV	Power-Operated Relief Valve
psig	pounds per square inch gauge
R16	Refueling Outage 16
RCS	Reactor Coolant System
RCV	Reactor Coolant Valve

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 Superintendent, 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