UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

June 16, 2005

EA-05-114

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC TRIENNIAL FIRE PROTECTION INSPECTION (FOLLOW UP) REPORT NO. 05000302/2005007; PRELIMINARY GREATER THAN GREEN FINDING

Dear Mr. Young:

On June 8, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an in-office examination of an unresolved item (URI) associated with the Crystal River 3 facility which was identified in NRC Inspection Report 05000302/2004009 (ADAMS Accession Number ML050740113) forwarded to you on March 14, 2005. Specifically, URI 05000302/2004009-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action, was identified as unresolved pending completion of a significance determination. The finding, as indicated by the enclosed supporting documentation, appears to be of greater than very low safety significance (GREEN). The finding and the preliminary significance determination were discussed on June 16, 2005, with you and other members of your staff.

As discussed in NRC Inspection Report 05000302/2004009, a fire in the 3A 4160V engineered safeguards (ES) compartment could result in a total loss of offsite power. In addition fire damage to the metering and protection electrical cables located in or just above the 3A 4160V ES switchgear could trip and lock out all feeder breakers to both 4160V ES busses causing a loss of all safety-related A.C. power. These protection and metering circuits were not physically separated or protected from fire damage as required by 10 CFR 50, Appendix R, Section III.G.2. Instead an unapproved local manual operator action was used to restore A.C. power. However, this local manual operator action to reset the 3B emergency diesel generator breaker lockout on the 3B 4160V ES switchgear was not feasible.

This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP). The finding was preliminarily determined to be Greater than Green. The finding has a greater than very low safety significance because it could affect fire protection defense in depth. In addition, uncertainties associated with identifying all the fire scenarios that can impact the metering and protection circuitry, the primary plant response to a delay in secondary side heat removal, the effectiveness of secondary side heat removal systems following an overcooling event, and operator response to these fires contribute to the preliminarily characterization of the finding as greater than green. This issue was also determined to be an apparent violation of NRC requirements, as discussed in the Enclosure. However, the finding does not represent a current safety concern because you have modified and corrected the nonconforming condition before the inspection team left the site.

Our SDP Phase 2 evaluation of this finding is provided in Attachment 2. Additional information from Progress Energy Florida (Florida Power Corporation) that addresses the assumptions in our evaluation will permit a more refined risk analysis.

Before we make a final decision on this matter, we are providing you an opportunity to (1) present to the NRC your perspectives on the facts and assumptions, used by the NRC to arrive at the finding and its significance, at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

This apparent violation is being considered for escalated enforcement action in accordance with the Enforcement Policy, because it is associated with a greater than Green finding. The current Enforcement Policy is included on the NRC's Web site at http://www.nrc.gov/reading-rm/adams.html

Please contact Mr. D. Charles Payne at (404) 562-4669 within seven days of the date of this letter to notify the NRC of your intentions regarding the regulatory conference for the preliminary Greater than Green finding. If we have not heard from you within 10 days, we will continue with our significance determination and associated enforcement processes on this finding, and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for the inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the referenced inspection report may change as a result of further NRC review. For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000302/2005007, and the above apparent violation is identified as AV 0500302/2005007-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action. Accordingly, URI 05000302/2004009-01 is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, portions of its enclosure and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). However, the NRC is continuing to review the appropriate classification of the SDP Phase 2 Evaluation (Attachment 2) within our records management program, considering changes in our practices following the events of September

11, 2001. Using our interim guidance, the attached analyses have been marked as Proprietary Information or Sensitive Information in accordance with Section 2.390(d) of Title 10 of the Code of Federal Regulations and will not be placed in the PDR. Please control the document accordingly (i.e., treat the document as if you had determined that it contained trade secrets and commercial or financial information that you considered privileged or confidential). We will inform you if the classification of these documents change as a result of our ongoing assessments. ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

If you have any questions regarding this letter, please contact me at 404-562-4600.

Sincerely,

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Victor M. McCree, Director
Division of Reactor Safety

Docket No.: 50-302
License No.: DPR-72

Enclosure: Inspection Report 05000302/2005007
w/Attachments: 1. Supplemental Information

cc w/encl:
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cc w/encl cont'd - (See page 4)

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J. Munday, RII
C. Evans, RII
R. Rodriguez, RII
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S. Sparks, RII
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X SISP REVIEW COMPLETE: Initials: WGR1 ☐ SISP REVIEW PENDING*: Initials: ________ *Non-Public until the review is complete
X PUBLICLY AVAILABLE NON-PUBLICLY AVAILABLE SENSITIVE X NON-SENSITIVE
ADAMS: X Yes ACCESSION NUMBER:_________________________

OFFICE	RII:DRS	RII:DRS	RII:DRP*	RII:OI			
SIGNATURE	WGR1	DCP	JTM	CFE			
NAME	WRogers	DPayne	JMunday	CEvans			
DATE	6/14/2005	6/14/2005	6/13/2005	6/16/2005	6/ /2005	6/ /2005	6/ /2005
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: E:\Filenet\ML051670575.wpd

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2005007
Licensee:	Progress Energy Florida (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	January 28, 2005 - June 16, 2005
Inspectors:	R. Rodriguez, Reactor Inspector R. Schin, Senior Reactor Inspector (Lead Inspector) W. Rogers, Senior Reactor Analyst
Approved by:	D. Charles Payne, Chief, Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000302/2005007; Crystal River Nuclear Plant, Unit 3; Significance Determination of Unresolved Item from Triennial Fire Protection Inspection.

This in-office review was conducted by two regional inspectors and a senior reactor analyst. One preliminary Greater than Green finding with an apparent violation was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events and Mitigating Systems

- Preliminary Greater than Green. An apparent violation of 10 CFR 50, Appendix R, Section III.G.2, for failure to physically protect or separate cables from fire damage and instead relying on an unapproved local manual operator action. The unprotected cables were associated with a common electrical protection and metering circuit which was installed such that fire damage to a cable in or just above the 3A 4160V engineered safeguards (ES) switchgear could result in tripping and locking out all feeder breakers to both 4160V ES busses, resulting in a loss of all safety-related alternating current power.

  In addition, the local manual operator action to reset the 3B emergency diesel generator breaker lockout on the 3B 4160V ES switchgear was determined to be non-feasible. During a severe fire in the adjacent 3A 4160V Switchgear Room the fire response activities would cause the location for the operator action (the 3B 4160V Switchgear Room) to be exposed to hot smoke, water mist, and water on the floor.

  This finding is greater than minor because it degraded the defense in depth for fire protection and also because it is associated with the protection against external factors attribute and degraded the reactor safety mitigating systems cornerstone objective. The finding adversely affected the reliability and capability of equipment required to achieve and maintain a safe shutdown condition following a severe fire in the 3A 4160V ES Switchgear Room. (Section 4OA5.01)

B. Licensee-identified Violations:

None

Report Details

4. OTHER ACTIVITIES

4OA5 OTHER

.1 (Closed) URI 05000302/2004009-01. Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action.

Introduction. An apparent violation (AV) of 10 CFR 50, Appendix R, Section III.G.2, for failure to physically protect or separate cables from fire damage and instead relying on a local manual operator action that was not approved by the NRC. The unprotected cables were in common electrical protection and metering circuits which were installed such that fire damage to a cable in or just above the 3A 4160V ES switchgear could trip and lock out all feeder breakers to both 4160V ES busses, resulting in a loss of all safety-related alternating current (a.c.) power.

In addition, the team found that the licensee's local manual operator action to mitigate this condition was not feasible. The action was to reset the 3B emergency diesel generator (EDG) breaker lockout on the 3B 4160V ES switchgear. However, that action was not feasible because the fire in the 3A 4160V Switchgear Room and fire fighting activities through the adjacent 3B 4160V Switchgear Room would cause the location for the operator action (in the 3B 4160V Switchgear Room) to be exposed to hot smoke, water mist, and water on the floor. This inspection finding was assessed using the SDP and preliminarily determined to be Greater than Green (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections.)

Description. During the baseline triennial fire protection inspection, the inspectors identified a finding involving cables for the electrical protection and metering circuit located in the 3A 4160V ES Switchgear Room were vulnerable to fire damage that could disable both the 3A 4160V ES switchgear and the redundant train 3B 4160V ES switchgear, having potential safety significance greater than very low significance. Specifically, the licensee's 10 CFR 50 Appendix R Fire Study and post-fire safe shutdown (SSD) procedure OP-880A, Appendix "R" Post-Fire Safe Shutdown Information, Rev. 5, Step 9-6, included a local manual operator action that was not approved by the NRC and also was not feasible. The action was to reset the 3B EDG breaker lockout on the 3B 4160V ES switchgear during a fire in the 3A 4160V ES Switchgear Room. This action was time critical and required to be completed within 30 minutes of entering OP-880A. Operators were to trip the reactor and enter OP-880A if a fire in the 3A 4160V Switchgear Room impacted safe operation of the plant. The licensee had considered that the action was needed because a fire in the 3A 4160V ES switchgear could affect cables for the electrical protection and metering circuit and could lock out all feeder breakers to both the 3A and the 3B 4160V ES switchgear. However, the licensee's post-fire SSD methodology relied upon equipment powered from the 3B 4160V ES switchgear. Specifically, the licensee's analysis determined that power to the 3B 4160V switchgear was needed within 30 minutes to enable operators to restore ventilation cooling to the Emergency Feedwater Initiation and Control (EFIC) rooms. The EFIC system was needed for automatic EFW flow control.

The team found that cables for the electrical protection and metering circuit were located within and directly above the 3A 4160V switchgear, where a fire originating in certain sections of the switchgear could immediately damage them. The team noted that these cables were four-conductor, #8 American Wire Gage (AWG), Institute of Institute of Electrical and Electronic Engineers (IEEE) 383 qualified, thermoset-type cables with no protective fire wrap. Damage to one of these cables could result in immediate loss of both the 3A and the 3B 4160V ES switchgear, and a loss of all safety-related a.c. power. Plant operators would conclude this had an impact on safe operation of the plant, would trip the reactor, and immediately enter OP-880.

During this fire condition, the primary plant operator (PPO) had a number of proceduralized time-critical local manual operator actions to perform in a prescribed sequence. Based on Pre-fire Plans and fire brigade drill results, the fire brigade would attack a fire in the 3A 4160V Switchgear Room through the 3B 4160V Switchgear Room about 15 minutes after confirmation of the fire. Based on licensee time validations and NRC team walkdowns of the actions, the team determined that the PPO would arrive at the 3B 4160V Switchgear Room about 25 minutes into the fire event. When the PPO arrived, the fire brigade would have the door between the two switchgear rooms open and would have sprayed fire water into the 3A 4160V Switchgear Room. Hot smoke from the fire would have filled both the 3A and 3B 4160V ES switchgear rooms and the hallway leading to those rooms because the fire brigade would have all doorways between the two rooms and the hallway blocked open with their fire hose. In addition, water from the fire hose would have created mist in the air and water on the floors of both switchgear rooms (the switchgear rooms had no floor drains). At 25 minutes into the fire event, the fire brigade would not yet have had time to evacuate the smoke with portable fans. In addition, the portable fans would not have electrical power available because the outlets for the fans were powered from the 3A and 3B 4160V switchgear, which would potentially be de-energized. While the fire brigade could obtain a portable generator to power the fans, this would take too long to allow the operator to complete the action within the time-critical 30 minutes. In view of all of these conditions, the team concluded that the operator action was not feasible.

This finding was an immediate safety concern and the licensee made modifications to correct the nonconforming condition before the inspection team left the site.

Analysis: This finding degraded the defense in depth for fire protection and also it is associated with the protection against external factors attribute and degraded the reactor safety mitigating systems cornerstone objective. The finding adversely affected the reliability and capability of equipment required to achieve and maintain a SSD condition following a severe fire. The finding is applicable to post-fire SSD from the control room during a fire in the 3A 4160V ES Switchgear Room. Because the finding affects fire protection, it was assessed in accordance with the NRC Reactor Oversight Process's SDP as described in NRC Inspection Manual Chapter 0609, Appendix F (MC 0609, App. F). In the Phase 1, the finding was associated with post-fire safe shutdown, it was assigned a high degradation rating and it existed for more than 30 days. As a result, a Phase 2 Risk Evaluation was required.

Summary of Phase 2 SDP Analysis

This evaluation was performed by Region II inspectors with the assistance of the regional SRA. The Crystal River Phase 2 SDP Analysis is included in this inspection report as Attachment 2.

The Phase 2 analysis involves a quantitative assessment of CDF increase given a finding. There are nine analysis steps and five screening checks. This assessment includes quantification of a Fire Frequency, Fire Damage State, Non-Suppression Probability and Conditional Core Damage Probability (CCDP). The report also contains several appendices documenting supplemental information used in the Phase 2 analysis.

Effects from a fire in the 3A 4160V switchgear room were postulated and evaluated. Not all ignition sources were counted in the fire area, only the cabinets where the CT circuitry is located were considered. The room has no automatic suppression and no manual suppression credit is applied with the target damaged within 1 minute. This yielded an Fire Frequency of 2.4E-04.

A loss of offsite power and a plant trip were postulated and the appropriate plant initiating event worksheet from the plant risk-informed inspection notebook was used to account for the plant SSD response and required human recovery actions in order to quantify the factor CCDP for each fire scenario of interest.

The Phase 2 analysis concluded that the change in Core Damage Frequency (Δ CDF) [the difference between the conforming case CDF and the non-conforming case CDF] was 2.4E-05 (substantial importance to safety).

SDP/Enforcement Review Panel (SERP) Evaluation

The total change in CDF due to the performance deficiency was found to be 2.4 E-05/yr for the unit. The dominant accident sequences that cause the largest Δ CDF are fully developed fires that require Emergency AC Power and Emergency Feedwater. The color associated with this magnitude of change in CDF is Greater than Green. Therefore, the SERP has preliminarily determined this issue to be a Greater than Green finding.

Enforcement: 10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R, Section III.G. Section III.G.2 applies to the ability to achieve and maintain hot SSD from the control room during a fire. It states, in part, that 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 three means of protecting cables to ensure that one of the redundant trains is free of fire damage shall be provided. The three means include, among others, the physical protection or separation of cables to preclude fire damage.

Contrary to the above, on January 26, 2005, cables for the electrical protection and metering circuit located in the 3A 4160V ES Switchgear Room were vulnerable to fire damage that could disable both the 3A 4160V ES switchgear and the redundant train 3B 4160V ES switchgear. Specifically, these protection and metering circuits were not physically separated or protected (as discussed above) from fire damage as required by 10 CFR 50, Appendix R, Section III.G.2. This apparent violation is identified as AV 05000302 /2005007-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action. Accordingly, URI 05000302/2004009-01 is closed.

4OA6 Meetings, Including Exit

On June 16, 2005, the inspectors presented the inspection results by telephone to Mr. Dale E. Young and other members of your staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

NRC personnel

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000302/2005007-01	AV	Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action (Section 4OA5.1)

Closed

05000302/2004009-01	URI	Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Procedures:

OP-880A, Appendix “R” Post-Fire Safe Shutdown Information, Rev. 5

Drawings:

E-215-031, Electrical Conduit Layout Control Complex, Rev. 56

Other Documents:

10CFR50 Appendix R Fire Study, Rev. 12

Licensee Event Report 50-302/2005-001, Design Change Create Engineered Safeguards Bus Protective Relay Scheme Single Failure Vulnerability, dated March 23, 2005

NRC Information Notice 2005-04: Single-failure and Fire Vulnerability of Redundant Electrical Safety Buses, dated February 14, 2005

LIST OF ACRONYMS

AV	Apparent Violation
AWG	American Wire Gage
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
EFIC	Emergency Feedwater Initiation and Control
EFW	Emergency Feedwater
ES	Engineered Safeguards
IEEE	Institute of Institute of Electrical and Electronic Engineers
IEL	Initiating Event Likelihood
MCR	Main Control Room
No.	Number
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PPO	Primary Plant Operator
SBO	Station Blackout
SDP	Significance Determination Process
SERP	SDP/Enforcement Review Panel
SSD	Safe Shutdown
URI	Unresolved Item
VIO	Violation