

**Florida  
Power**

CORPORATION  
Crystal River Unit 3  
Docket No. 50-302

June 23, 1997  
3F0697-16

U. S. Nuclear Regulatory Commission  
Attention: Document Control Desk  
Washington, D. C. 20555-0001

Subject: Licensee Event Report (LER) 96-001-02

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-001-02. This supplemental report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73. It provides an update of corrective actions relative to the field validation of Appendix R drawings and a self assessment of configuration controls.

Sincerely,

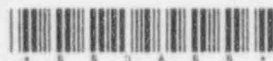
J. J. Holden, Director  
Nuclear Engineering and Projects

JJH/TWC

Attachment

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

9707070134 970623  
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S PDR



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## LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503.

FACILITY NAME (1)

CRYSTAL RIVER UNIT 3 (CR-3)

DOCKET NUMBER (2)

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PAGE (3)

TITLE (4)

Personnel Error by Contractor Results in Operation Outside 10 CFR 50 Appendix R Design Basis

EVENT DATE (5)

LER NUMBER (6)

REPORT DATE (7)

OTHER FACILITIES INVOLVED (8)

MONTH			DAY			YEAR			YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		MONTH			DAY			YEAR			FACILITY NAMES			DOCKET NUMBER(S)		
0	1	1	0	9	6	9	6	0	0	1	0	2	0	6	2	3	9	7	N/A	0	5	0	0	0	0	0	0	0	0

OPERATING MODE (9)

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 50. (CHECK ONE OR MORE OF THE FOLLOWING) (11)

POWER LEVEL (10)

0 0 0 0

20.402(b)

20.405(a)(1)(i)

20.405(a)(1)(ii)

20.405(a)(1)(iii)

20.405(a)(1)(iv)

20.405(a)(1)(v)

20.405(c)

50.36(c)(1)

50.36(c)(2)

50.73(a)(2)(i)

X 50.73(a)(2)(ii)

50.73(a)(2)(iii)

50.73(a)(2)(iv)

50.73(a)(2)(v)

50.73(a)(2)(vii)

50.73(a)(2)(viii)(A)

50.73(a)(2)(viii)(B)

50.73(a)(2)(ix)

73.71(b)

73.71(c)

OTHER (Specify in Abstract below and in Text, NRC Form 366A)

NAME

T.W. Catchpole, Sr. Nuclear Licensing Engineer

TELEPHONE NUMBER

AREA CODE

3 5 2 5 6 3 - 4 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS

SUPPLEMENTAL REPORT EXPECTED (14)

EXPECTED SUBMISSION DATE (15)

MONTH DAY YEAR

YES (If yes, complete EXPECTED SUBMISSION DATE)

X NO

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On January 10, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE THREE (HOT STANDBY). During a walkdown to develop a modification for upgrading CR-3's Thermo-Lag fire barriers, an Appendix R separation criteria deviation was identified in that two conduits containing circuits for controlling "B" Train Emergency Feedwater flow to the steam generators were noted to pass through the same fire area as "A" Train circuits with no fire barrier protection in that area. Although the condition was initially determined not reportable, upon further review on January 11, 1996 while the unit was in MODE FOUR (HOT SHUTDOWN), a 4-hour prompt notification was made to identify a degraded condition while the plant was shutdown. This report documents the separation problem as a condition outside CR-3's design basis. A justification for continued operation was established based on the existence of continuous roving fire watches in the affected area. The cause of this event was cognitive personnel error by personnel involved in preparing a modification to install conduit and cable during a 1985 refueling outage. Affected circuits have been brought back into compliance with Appendix R and a field validation of selected Appendix R drawings has been completed. A self-assessment of configuration controls covering multi-discipline modifications was also performed resulting in strengthening of procedural controls and review board oversight.

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (MNBB 77-14), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503.

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TEXT (If more space is required, Use additional NRC Form 366A's (17))

**EVENT DESCRIPTION**

On January 10, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE THREE (HOT STANDBY). During a walkdown by personnel involved in developing a modification for upgrading CR-3's Thermo-Lag fire barriers, it was noted that conduits EFS56 and EFS57, which contain "B" Train circuits for Emergency Feedwater System [BA] (EFW) components, were routed through the "A" 480-Volt Engineered Safeguards (ES) Switchgear Room. The portion of the "B" Train conduits in the "A" Switchgear Room were not protected by fire barriers and are within 20 feet of redundant "A" train circuits. See Figure 1.

Conduits EFS56 and EFS57 contain circuits which provide flow control signals and motive power to controllers [BA,FC] for the "B" train Emergency Feedwater Flow Control Valves [BA,FCV]. The redundant "A" train circuits are associated with controllers for "A" train Emergency Feedwater Flow Control Valves. A Problem Report was generated to identify a noncompliance with 10CFR50 Appendix R, Section III.G.2.b separation criteria. The Problem Report was initially considered not to be reportable on the basis that the conduit was in the confirmed route of the continuous roving fire watch established in response to NRC Bulletin 92-01 "Failure of Thermo-lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function".

Upon further review of the Problem Report on January 11, 1996, while the unit was in MODE FOUR (HOT SHUTDOWN) it was determined the condition warranted a 4-hour report in accordance with 10CFR50.72. The installed condition did not conform with 10CFR50 Appendix R during the time period from 1985, when the conduits were installed, to the establishment of the compensatory fire watch in 1992. The notification was made at 1828 hours in accordance with 10CFR50.72(b)(2)(i) as a degraded condition found while the reactor was shutdown and was assigned Event Number 29826.

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B) to describe a condition outside the design basis of CR-3 with regard to 10CFR50 Appendix R Fire analyses during the time period described above.

**EVENT EVALUATION**

The control circuits for the EFW Flow Control Valves are necessary for safe shutdown and are subject to the requirements of 10CFR50 Appendix R Section III.G. This section addresses requirements for protection features needed to ensure one train of a safe shutdown system is free of fire damage. CR-3's design basis requires separation by a fire barrier with a 3-hour rating, or separation by horizontal distance of more than 20 feet with no intervening combustibles or fire hazards, or enclosure in a fire barrier having a 1-hour rating. In the second and third cases, the area must also be equipped with fire detection and automatic fire suppression. CR-3 complied with the enclosure requirements by use of a fire

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 60.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503.

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TEXT (If more space is required, Use additional NRC Form 366A's (17))

resistive material called Thermo-Lag 330. As shown on Figure 1, Conduits EFS56 and EFS57, which contain control power and instrumentation circuits for the "B" train EFW flow control valves, were routed with no protection through the "A" 480 Volt Engineered Safeguards (ES) Switchgear Room within 20 horizontal feet of the controllers for the "A" train control valves. The 480V Switchgear Rooms are 3-hour fire areas. Conduits EFS56 and EFS57 were intended to be routed to the "B" 480 Volt ES Switchgear Room through the corridor area from EFIC Room "B". The Corridor area is a 1-hour fire area and the portion of EFS56 and EFS57 in this area is coated with Thermo-Lag. If there was a fire in the "A" 480 ES Switchgear Room in the area of the EFV-57 and EFV-58 controllers and the as-installed conduits EFS56 and EFS57, then flow control or level fill rate of emergency feedwater could be lost.

An operability assessment of the EFW System was conducted in accordance with Compliance Procedure CP-150, "Identifying and Processing Operability Concerns." The assessment provided a justification for continued operation which recognized the existence of continuous roving fire watches that include the "A" 480 Volt ES Switchgear Room. These fire watches had been established in response to NRC concerns which showed Thermo-Lag material to be deficient. The operability assessment also noted the fire watch in the "A" 480V ES Switchgear Room supplements the installed Pyrotechnics fire detection system in the area. The Shift Supervisor on Duty (SSOD) accepted the operability assessment on January 15, 1996 and closed the concern.

**CAUSE**

The primary cause of this event was cognitive personnel error by engineering personnel. Conduits EFS56 and EFS57 were installed in 1985 during Refuel 5 as part of a field change to modification 80-10-66-08A "EFIC Electrical Conduit and Cable Installation". The field change was a multi-discipline (electrical and structural) change initiated by the electrical discipline for the purpose of addressing Appendix R. After the electrical design was completed, structural engineering completed the balance of the design for the conduit installation and approved the change notice. The structural design engineer selected a different route for conduit supports from that depicted on layout sketches provided by electrical engineering. It is surmised that this route was selected to take advantage of existing conduit supports and/or due to interferences present in the route selected by electrical engineering. The structural engineer was apparently not aware of the impact this change had on Appendix R criteria. Further, design verification of the field change did not include a final review by electrical engineering.

Two other factors may have contributed to this event or the failure to discover the discrepancy. One was insufficient procedures in place during Refuel 5 to guide interface between engineering disciplines. The current guidelines in Nuclear Engineering Procedure (NEP) 210 "Modification Approval Records", require



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TEXT (If more space is required, Use additional NRC Form 366A's (17))

a multi-discipline review and co-approval signatures of the Design Engineers and Verification Engineers involved in developing the modification. Another factor was that Appendix R drawings were developed using conduit layout drawings which are diagrammatic and do not depict the exact route.

**IMMEDIATE CORRECTIVE ACTION**

When the Problem Report for this condition was first presented to the Shift Manager on January 10, 1996 it was confirmed the separation discrepancy was in a fire area covered by the roving fire watch.

**ADDITIONAL CORRECTIVE ACTION**

1. Drawings which indicate the routing of Conduits EFS56 and EFS57 have been revised to depict the correct routing.
2. FPC conducted a validation of Appendix R drawings using a sample-based approach to compare the as-built configuration against the configuration confirmed by field walkdown. The sample involved a horizontal and vertical validation of all fire barrier protected raceways associated with the Emergency Feedwater system and Fire Areas CC-108-107 and CC-124-116 which are the 4160V 'B' and 480V 'B' Switchgear Rooms, respectively. Although additional discrepancies were identified consisting of tray labeling and color coding problems, small breaks in firewrap, and information missing from drawings, none were representative of design basis issues similar to those described in this report. Based on the results of the validations completed, additional walkdowns were determined to be unnecessary.
3. Circuits contained in Conduit EFS56 were relocated and wrapped with Mecatiss per the commitment described in FPC letter 3F1295-05 to NRC dated December 21, 1995. Circuits contained in Conduit EFS57 were rerouted such that firewrap is no longer required.

**ACTION TO PREVENT RECURRENCE**

Both short-term and long-term actions have been identified to address the root cause for this event. A copy of the initiating Problem Report has been provided to Nuclear Engineering Design personnel to re-emphasize the need for proper design interface between engineering disciplines. In addition, a self-assessment was performed of the effectiveness of existing (enhanced) configuration controls in the area of design interface between engineering disciplines. The results of the self assessment indicated that the existing interface controls needed to be enhanced. FPC has taken the following actions to enhance interdisciplinary design controls:

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TEXT (If more space is required, Use additional NRC Form 366A's (17))

1. Utilization of the precursor card process has been increased for documenting and resolving configuration and design issues.
2. A Design Review Board (DRB) has been established consisting of interdisciplinary members from the design engineering, systems engineering, operations, maintenance, training, and licensing organizations. Generally, projects which have an effect on nuclear safety will be selected for DRB review.
3. Nuclear Engineering Procedure (NEP) 104, "Interface Design Control," has been revised to strengthen expectations regarding the need for verification of the final design by the submitting design discipline when inputs are provided to other disciplines.

PREVIOUS SIMILAR EVENTS

LER 89-39 reported a similar event. In this LER, it was discovered that a fuse was relocated to satisfy Appendix R requirements; however, the new location did not fulfill the desired objective. The cause of the separation problem was identified as design error in that personnel preparing and reviewing a field change for installation in Refuel 5 were not knowledgeable of Appendix R Separation criteria. The actions to prevent recurrence as identified in the LER were to review all modifications subsequent to Refuel 5 to assure Appendix R separation criteria was met. The evaluation of the problem report which initiated the LER noted it was considered to an isolated case of misinterpretation of Appendix R design requirements by field personnel.

ATTACHMENT

Figure 1 - Control Complex Elevation 124 Fire Areas

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 0.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (NMBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503.

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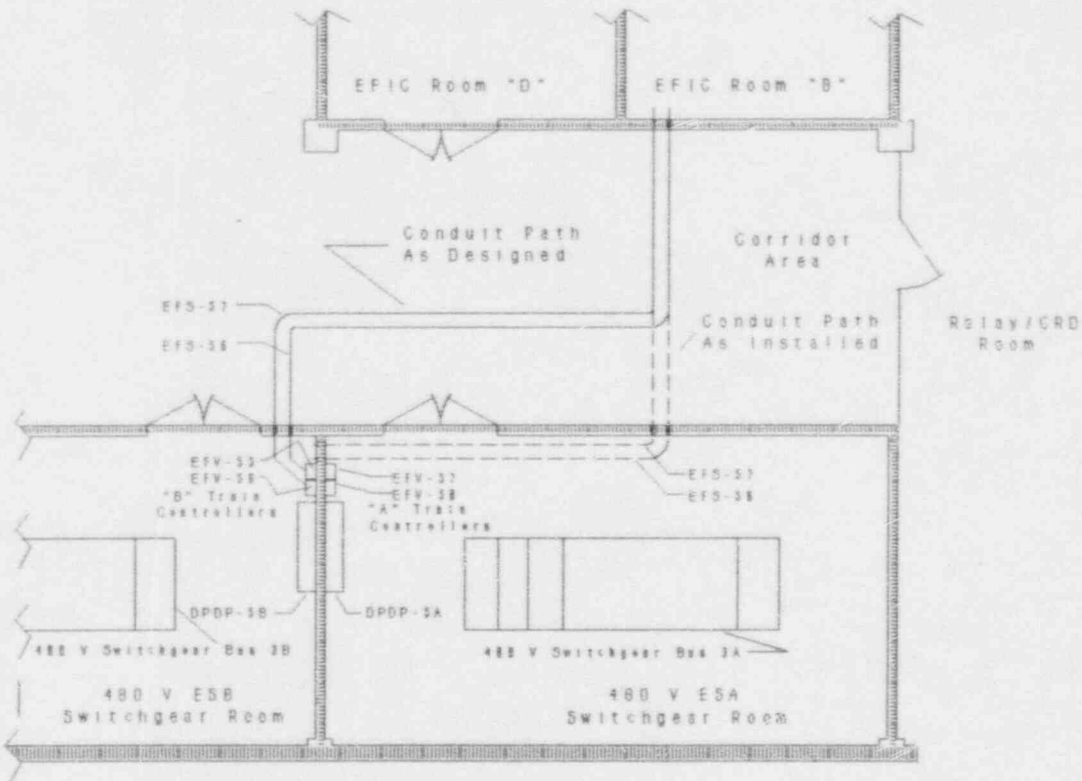
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CONTROL COMPLEX EL. 124 FIRE AREAS

FIGURE 1