

# Florida Power

CORPORATION  
Crystal River Unit 3  
Doclet No. 90-302

March 8, 1996  
3F0396-04

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

Subject: Licensee Event Report (LER) 96-001-01

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-001-01. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73. It provides the criteria and expected completion date for a field validation of Appendix R drawings. An additional supplement to this report is expected to be provided by December 31, 1996 to include results of the field validation and a self assessment of configuration controls.

Sincerely,

B. J. Hickle, Director  
Nuclear Plant Operations

TWC:ff

Attachment

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

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EXPIRES 5/31/95

## 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)										DOCKET NUMBER (2)										PAGE (3)										
CRYSTAL RIVER UNIT 3 (CR-3)										0 5 0 0 0 3 0 2										1 OF 0 8										
TITLE (4)																														
Personnel Error by Contractor Results in Operation Outside 10CFR50 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)																
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OPERATING MODE (9)		3		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (CHECK ONE OR MORE OF THE FOLLOWING) (11)																										
POWER LEVEL (10)		0 0 0		20.402(b)					20.405(c)					50.73(a)(2)(iv)					73.71(b)											
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				20.405(a)(1)(ii)					50.36(c)(2)					50.73(a)(2)(vii)					OTHER (Specify in Abstract below and in Text, NRC Form 396A)											
				20.405(a)(1)(iii)					50.73(a)(2)(i)					50.73(a)(2)(viii)(A)																
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LICENSEE CONTACT FOR THIS LER (12)																														
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T.W. Catchpole, Sr. Nuclear Licensing Engineer										3 5 2					5 6 3 - 4 6 0 1															
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE IN THIS REPORT (13)																														
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS								
SUPPLEMENTAL REPORT EXPECTED (14)															EXPECTED SUBMISSION DATE (15)					MONTH	DAY	YEAR								
X YES (If yes, complete EXPECTED SUBMISSION DATE)																														

**ABSTRACT** (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines (18))

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 problem 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 contractor personnel involved in preparing a modification to install conduit and cable during a 1985 refueling outage. Corrective actions will include revision to drawings, bringing affected circuits into compliance, field validation of Appendix R drawings and a self-assessment to determine the effectiveness of configuration controls covering multi-discipline modifications.

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**EVENT DESCRIPTION**

On January 10, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE THREE (HOT STANDBY). The plant was in the process of shutting down due to a forced outage associated with an earlier saltwater intrusion through a condenser tube leak. 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 are 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 in that circuits and components for both trains of Emergency Feedwater are located in the same fire area and both trains of Emergency Feedwater are subject to loss of flow control if a fire occurred. 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 since 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 via the Non-Emergency Event Notification system 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**

Train "B" Emergency Feedwater Valves EFV-55 and EFV-56 and Train "A" valves EFV-57 and EFV-58 control flow and level for various design basis accidents including Loss of Main Feedwater, Main Feedwater Line Break, Main Steam Line Break, and Loss of Coolant Accidents (LOCA). These flow control valves are part of the

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Emergency Feedwater [BA] (EFW) system which provides secondary coolant to the Once Through Steam Generators [AB,SG] (OTSG) at a rate sufficient to remove decay heat in the event the Main Feedwater System [SJ] is unable to perform this function. EFW is also used to promote and enhance natural circulation in the Reactor Coolant System [AB] (RCS). The Emergency Feedwater Initiation and Control System [JB] (EFIC) is the system that determines the need for EFW based on plant conditions. It initiates EFW by starting the pumps and controlling the valves to provide a flow path, and then controls flowrate in order to maintain water level in the steam generators. Once the system is operating, EFIC will automatically control the level in the OTSG's at the appropriate setpoint depending upon plant conditions. EFIC will also control the valves to limit the OTSG fill rate. This is done to minimize overcooling of the Reactor Coolant System when EFW is initiated. In order to prevent excessive EFW flow, which could cause damage to the EFW pumps or to the steam generator tubes, the EFW flow control valves will automatically reduce EFW flow when an EFW actuation occurs.

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 resistive material called Thermo-Lag 330. As noted above, and as shown on Figure 1, Conduits EFS56 and EFS57, which contain control power and instrumentation circuits for the "B" train EFW flow control valves, are 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 any flow control or level fill rate of emergency feedwater could be lost. The valves would fail 100 per cent open on loss of control signal or loss of power. There would be no flow control to limit EFW Pump runout and cavitation. There would be no flow control to limit RCS overcooling and there would be a potential for losing RCS Subcooling Margin due to loss of pressurizer level in the RCS.

An operability assessment of the EFW System was conducted in accordance with Compliance Procedure CP-150, "Identifying and Processing Operability Concerns." This procedure provides a structured approach toward determining the operability of plant components required for accident mitigation and safe shutdown of the



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plant. 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 Pyrotronics 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 contractor engineering personnel. Conduits EFS56 and EFS57 were installed in 1985 during Refuel 5 as part of a field change to modification 88-10-66-08A "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.

A secondary cause of this event was the failure to identify the conduit layout discrepancy during the installation and completion of modification 82-10-19-04 which was also installed during Refuel 5 for the purpose of Thermo-Lag installation.

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", requires, for a multi-discipline modification package, the co-approval signatures of the Design Engineers and Verification Engineers involved in developing the MAR; however, this was not specifically addressed in the engineering procedures in effect during Refuel 5. Another factor was that Appendix R drawings were developed using conduit layout drawings which are diagrammatic and do not necessarily depict the exact route. Since the conduit layout drawings were as-built according to the sketches provided by electrical engineering at a different time than the conduit support drawings, which were as-built per the sketches provided by structural engineering, a comparison was not performed and therefore, the discrepancy was not identified during the as-building stage.

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**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 will conduct 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 will be both horizontal and vertical. The horizontal sample will consist of validating the location of all fire barrier protected raceways for a selected plant system. The vertical sample will consist of validating the location of all fire barrier protected raceways for a selected fire area. It is expected that field validation will be completed by the end of 1996 consistent with other actions noted in Item 3, below. Depending on the results of the initial validation, a decision will be made to proceed with additional validation.
3. Circuits contained in Conduits EFS56 and EFS57 will be brought back into compliance with 10CFR50 Appendix R. This may include wrapping with Mecatiss per the commitment described in FPC letter 3F1295-05 to NRC dated December 21, 1995. Consistent with commitments in the letter for Mecatiss wrapping, compliance with Appendix R will be achieved by the end of 1996.

**ACTION TO PREVENT RECURRENCE**

Both short-term and long-term actions have been identified as appropriate 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 to this, a self-assessment will be performed by April 30, 1996 of the effectiveness of existing (enhanced) configuration controls in the area of design interface between engineering disciplines. Additional actions to prevent recurrence may be identified at that time based on the results of the assessment.

FPC expects to provide a supplement to this report by December 31, 1996 to inform NRC of any additional actions that may be identified as a result of the field validation of Appendix R drawings as well as the above self-assessment.

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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

Attachment 1 -Abbreviations, Definitions and Acronyms

Figure 1 - Control Complex Elevation 124 Fire Areas

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## ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

10 CFR 50  
Appendix R

"Fire Protection Program for Nuclear Facilities"

CR-3

Crystal River Unit 3

EFIC

Emergency Feedwater Initiation and Control

EFW

Emergency Feedwater

FPC

Florida Power Corporation

LOCA

Loss of Coolant Accident

MODE THREE

HOT STANDBY

MODE FOUR

HOT SHUTDOWN

OTSG

Once Through Steam Generator

Problem Report

A Problem Report documents a condition or event which impacts CR-3 and warrants evaluation, root cause analysis, or corrective actions beyond what it would receive if documented and processed by other methods.

## NOTES:

ITS defined terms appear capitalized in LER text (e.g. MODE ONE)

Defined terms/acronyms/abbreviations appear in parentheses when first used (e.g. Reactor Building (RB) ).

EIIIS codes appear in square brackets (e.g. Makeup Tank [CB,TK] )



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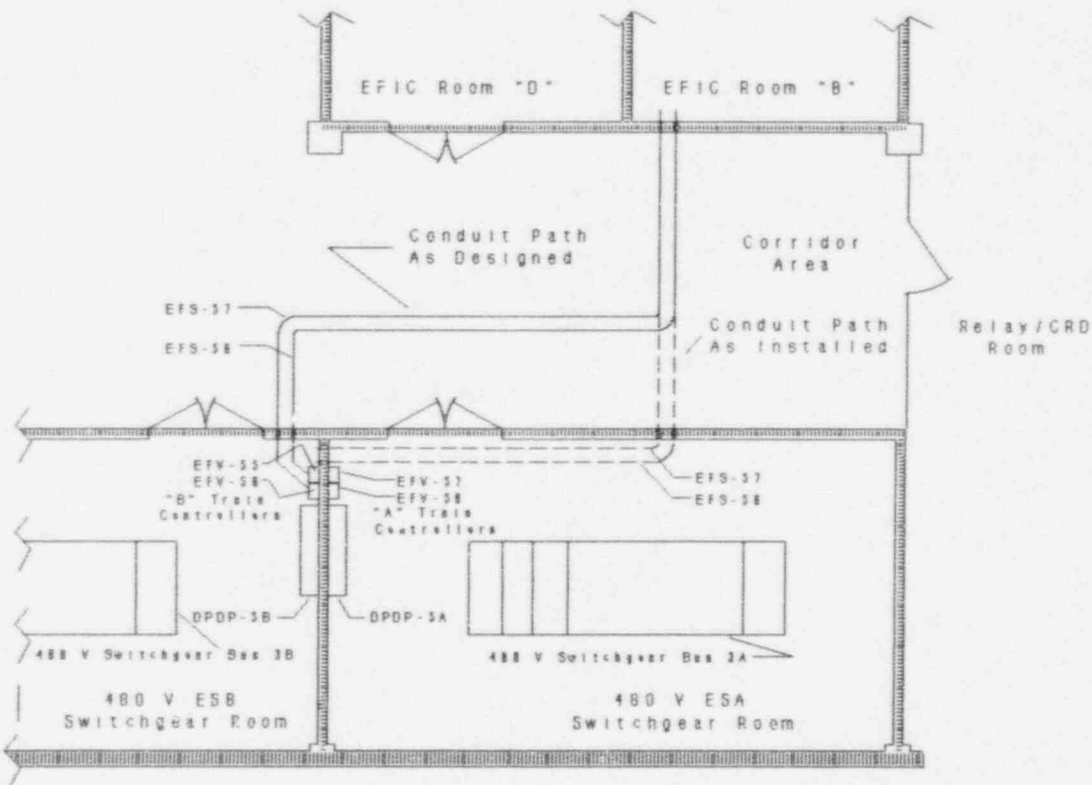
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CONTROL COMPLEX EL. 124' FIRE AREAS

FIGURE 1