



Florida Power

CORPORATION
Crystal River Unit 3
Docket No. 80-302

December 4, 1995
3F1295-01

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

Subject: Licensee Event Report (LER) 95-025-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 95-025-00. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73. A supplement to this report will be provided by December 20, 1996.

Sincerely,

B. J. Hickle, Director
Nuclear Plant Operations

TWC:ff

Attachment

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

120141

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) 0 5 0 0 0 3 0 2			PAGE (3) 1 OF 0 7		
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TITLE (4)
Personnel Errors by Architect Engineer Result in Operation Outside Design Basis Due to Inadequate Safety/Non-Safety Circuit Isolation

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 N/A		DOCKET NUMBER(S) 0 5 0 0 0	
1	1	0 2 9 5	9 5	0 2 5	0 0	1	2	0 4 9 5	N/A		0 5 0 0 0	

OPERATING MODE (9) 1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (CHECK ONE OR MORE OF THE FOLLOWING) (11)										
POWER LEVEL (10) 1 0 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)							
	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)							
	20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)							
	20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)								
	20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)								
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)									

LICENSEE CONTACT FOR THIS LER (12)										
NAME T.W. Catchpole, Sr. Nuclear Licensing Engineer								TELEPHONE NUMBER		
								AREA CODE 9 0 4		
								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	

SUPPLEMENTAL REPORT EXPECTED (14)								EXPECTED SUBMISSION DATE (15)		MONTH	DAY	YEAR
<input checked="" type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)								<input type="checkbox"/> NO		1	2	2 0 9 6

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

On November 2, 1995, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 96% reactor power and generating 847 megawatts. While reviewing the control power scheme associated with Reactor Building Purge System containment isolation valves, a design engineer noted that several safety related circuits used non-safety related terminal boxes and other unisrciated devices in their routing. This is a condition outside the design basis for CR-3 which requires Class 1E electrical systems to have sufficient isolation to prevent a common failure mode for any design basis event. The cause of this event was determined to be cognitive personnel error by the Architect Engineer during original plant design. The safety significance of the event is negligible in that the safety function of the valves is to assist in maintaining Reactor Building Integrity, but their "closed" function is not assumed in the Fuel Handling Accident analysis for CR-3. FPC will provide a supplemental report by December 20, 1996 indicating corrective actions to be taken after an evaluation of alternatives is considered.

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TEXT CONTINUATION

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TEXT (If more space is required, Use additional NRC Form 306A's (17))

EVENT DESCRIPTION

On November 2, 1995, Florida Power Corporation's Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 96% reactor power and generating 847 megawatts. While resolving a precursor card concerning an open link in the starter cubicle for non-safety related Air Handling Fan (AHF) [UC, FAN] AHF-6A, it was noted that the link is a permissive to automatically close safety related Containment Purge Supply Valve [VA, ISV] AHV-1C. The design engineer, using a questioning attitude, decided to review the control power scheme associated with AHV-1C and other Reactor Building Purge [VA] System (AH-XK) containment isolation valves which includes Containment Purge Exhaust Valves AHV-1A and 1B, and Containment Purge Supply Valves AHV-1C and 1D. The review identified that several safety related circuits [VA, JX] utilized non-safety related terminal boxes [JBX] and other unisolated devices in their routing. The results were reported to the NRC at 1400 via the Emergency Notification System per 10CFR50.72(b)(1)(ii)(B) as a condition suspected to be outside the design basis for CR-3 and Event Number 29545 was assigned. FPC's FSAR Section 8.1 states Electrical Systems for CR-3 satisfy the IEEE 308 proposed criteria for Class 1E Electrical Systems, dated June, 1969 which requires sufficient electrical isolation to prevent the occurrence of common failure mode in Class 1E systems for any design basis event.

The non-safety related devices within the subject control power scheme which functionally interface with AHV-1A through 1D, include Reactor Building Purge Duct Radiation Monitor [IL, MON] RM-A1 and the Reactor Building Purge Exhaust Fans [VA, FAN] AHF-7A and AHF-7B. RM-A1 provides a closure signal to AHV-1A through 1D on high radiation. Interlocks in the circuits close AHV-1A through 1D when AHF-7A and 7B stop.

The following specific conditions were noted:

A. With respect to AHV-1B and 1C:

1. Circuits AHC61 and AHC81 are safety related Engineered Safeguards (ES) Train "A" cables routed from safety related 480V Motor Control Center [MCC] MCC 3A1 to the non-safety related 480V Ventilating MCC 3A.
2. Circuits AHC62 and AHC82 are safety related ES Train "A" cables routed from 480V Ventilating MCC 3A to 480V Ventilating MCC 3B. Both MCC's are non-safety related.
3. Circuits AHC58 and AHC78 are safety related ES Train "A" cables routed to the Radiation Monitor Control Console [IL, PL] from the ESF(A) section of the Main Control Board [MCBD]. The circuit is tied to Radiation Monitor RM-A1 which is a non-safety device.

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TEXT (If more space is required, Use additional NRC Form 386A's (17))

B. With respect to AHV-1A and 1D:

1. Terminal Boxes AH-23 (Control Station for Controlled Access Area Exhaust Fans AHF-20A and AHF-20B) and AH-24 (Control Station for Controlled Access Area Supply Fan AHF-30) are the common terminal points for AHV-1A and 1D and are both non-safety related. Both boxes have safety related ES Train "B" and non-safety related cables routed to them. AH-24 is located in the Turbine Building which is a non-seismic area. The safety related cables routed to and terminated in AH-23 include AHE41 and AHE46 through AHE49. Those routed to and terminated in AH-24 include AHE51, AHE52 and AHE54.
2. Several devices are tied to the control power system [VA,PL] (Distribution Panel DPDP-5B, Fuse 22) and are safety related; however, the cables to these devices are non-safety related. Devices include solenoid valves [VA,PSV] AHV-1A-SV1 and SV2, AHV-1D-SV1 and SV2, and limit switches [VA,33] AHV-1A-KS1 and KS2, and AHV-1D-KS1 and KS2.
3. Circuits AHE46 and AHE51 are safety related ES Train "B" cables routed to the Radiation Monitor Control Console from Terminal Boxes AH-23 and AH-24. The circuits tie to Radiation Monitor RM-A1 which is a non-safety related device.
4. Differential Pressure Switches [VA,PDIS] AH-266-DPS (Reactor Building Purge Supply) and AH-552-DPS (Intake and Discharge to Reactor Building Purge Supply and Exhaust Fans [VA,FAN] AHF-7A and 7B) are non-safety related, but are in series with safety related devices with no isolation devices. The circuits to these devices are terminated to safety related circuits in Terminal Boxes AH-23 and AH-24.

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B).

EVENT EVALUATION

The safety function of the Reactor Building Purge System (AH-XC) is to assist in maintaining the Reactor Building (RB) integrity in the event of an accident and to assist in providing hydrogen purge to the RB through purge exhaust filters following a Loss of Coolant Accident (LOCA). The AH-XC System reduces building airborne contamination, and filters potentially contaminated particles and gases prior to discharging exhaust air into the atmosphere. The Reactor Building Purge System operates as required during Operational MODES 5 and 6 with two supply fans with isolation valves and 2 exhaust fans with isolation valves. RB purge operation may be required for internal RB inspection or maintenance but is only

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allowed during MODES 5 and 6. The containment penetration requirements addressed in Improved Technical Specification 3.9.3 require AHV-1A through 1D to be in their closed position during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is the period of highest risk potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by the OPERABILITY requirements of Improved Technical Specification 3.6.3, "Containment Isolation Valves." When CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

The RB purge isolation valves are butterfly type and 48 inches in diameter. The valve operators for the purge valves located outside of the RB (AHV-1A/1D) are of the spring return air cylinder type with the spring driving the valve to the closed position. Each pneumatic valve is controlled with two 3-way solenoid valves, either capable of permitting the valve to close in two seconds maximum time. The valve operators for the purge valves located on the inside of the RB (AHV-1B/1C) are electric motor driven with a totally enclosed, non-ventilated type motor and gear drive. The valves close in 5 seconds maximum.

The safety significance of this event is negligible. Even if a fault were to have occurred in any of the Non-Class 1E circuits described above, OPERABILITY would not have been a concern because the pneumatic valves are spring loaded to their safeguards position (closed) and fail closed on loss of air when the solenoids are deenergized, and the motor operated valves fail as-is on loss of power. Further, Chapter 14 of CR-3's FSAR does not take credit for the closure of the Reactor Building Purge Valves as part of the Fuel Handling Accident analysis. For fuel handling accidents inside containment, escaped gases are assumed to be released to the environment via the Reactor Building Purge System filters and dispersed into the atmosphere. Based on both design and realistic analyses, the thyroid and whole body doses are "well within" the limits of 10 CFR 100. Therefore, there is no safety concern with respect to the valve operators for AHV-1A through 1D.

An evaluation of control power scheme for AHV-1A through 1D indicate that three other safety related components are powered from the same breaker and fuse. These components include Reactor Coolant Pump Seal Bleedoff Shutoff Valve [CB,SHV] MUV-253, Pressurizer & Letdown Outside Penetration Sample Isolation Valve [KN,ISV] CAV-2 and Steam Generator Outside Penetration Sample Isolation Valve [KN,ISV] CAV-6. These valves will all fail to their safeguards position when their solenoid valves are deenergized.

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CAUSE

This condition is considered to be a result of cognitive personnel error by the Architect Engineer during original plant design. The design engineer for the subject circuits failed to recognize the design criteria for electrical isolation as described in IEEE 308 "Proposed Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations" dated June, 1969.

IMMEDIATE CORRECTIVE ACTION

An evaluation of the control circuits for AHV-1A through 1D confirmed that the closure function of the valves can be maintained by local control and that faults on AHV-1A through 1D will not affect the safeguards position of other safety related components which share the same circuits.

ADDITIONAL CORRECTIVE ACTION

FPC will evaluate, by December 20, 1996 alternatives to the present non-isolated design of the control circuits for AHV-1A through 1D. Alternatives to be considered will be either to revise the safety classification of the circuitry or to correct the non-safety interface. Based on the alternatives selected, additional actions may be generated to resolve this problem. FPC considers the date of December 20, 1996 to be reasonable in view of the relatively minor safety significance of this problem and the desire to reconcile this effort with other priorities and tasks within the electrical engineering effort. This LER will be supplemented by December 20, 1996 to describe the corrective actions to be taken.

ACTION TO PREVENT RECURRENCE

Updated design criteria have been developed which, in conjunction with improved industry standards, provide specific guidance in this area. Engineering procedures and tracking and documentation systems presently in place will preclude such an incident from recurring during installation or modification of control circuitry in the future. In addition, the Electrical Design Criteria Manual was developed in 1991 as a single, complete, reliable source of design documentation for electrical circuit routing and other technical areas for the purpose of assuring consistency in the design process.

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

PREVIOUS SIMILAR EVENTS

There have been two previous events involving electrical isolation problems. LER 89-034 reported discovery of safety related components powered from a non-1E distribution panel and non-safety related testing solenoid valves which shared common circuits with safety related actuation solenoid valves. LER 93-05 reported the failure during original plant design, to specify an isolation device for installation between the safety related and non-safety related portions of the control circuit for Makeup Valve MUV-49, thus placing the unit in a condition outside the design basis.

ATTACHMENT

Attachment 1 -Abbreviations, Definitions and Acronyms

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ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

- AH Air Handling System
- AH-XK Reactor Building Purge System
- CR-3 Crystal River Unit 3
- ES Engineered Safeguards
- FPC Florida Power Corporation
- IEEE-308 Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations
- ISOLATION DEVICE A device in a circuit which prevents malfunctions in one section of a circuit from causing unacceptable influences in other sections of the same circuit or in other circuits.
- LCO Limiting Condition for Operation
- MODE ONE POWER OPERATION (Greater Than 5 Percent Rated Thermal Power)
- MCC Motor Control Center
- Precursor A Precursor Card is used to provide initial identification of concerns, incidents or conditions which may result in minor corrective actions or by escalation to a more formal correction process, such as the Problem Report System.
- 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)).
 EIIS codes appear in square brackets (e.g. Makeup Tank [CB,TK])