Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

December 22, 1995
3F1295-24

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

Subject: Licensee Event Report (LER) 95-025-01

Dear Sir:

Please find the enclosed revised Licensee Event Report (LER) 95-025-01. This revision is submitted to correct an error on page 1 of the LER which previously noted the wrong reference to 10CFR50.73 reporting requirements. Other changes were made in the text to provide clarification.

Sincerely,

Gregory H Halnon for BJH

B. J. Hickle, Director
Nuclear Plant Operations

TWC/lf

Attachment

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

200005

CRYSTAL RIVER ENERGY COMPLEX: 15760 W Power Line St • Crystal River, Florida 34428-6708 • (352) 795-6486
A Florida Progress Company

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104

EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	05000302	1 OF 07

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	DOCKET NUMBER(S)
									N/A	05000
11	02	95	95	025	01	12	22	95	N/A	05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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

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)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 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	TELEPHONE NUMBER (AREA CODE)
T.W. Catchpole, Sr. Nuclear Licensing Engineer	352 563-4601

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		12	20	96

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 unisolated 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 isolation valves are locked closed in MODES 1 through 4 and 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.

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-XC) 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. Isolation criteria is also contained in CR-3's Electrical Design Criteria Manual which is referenced in the FSAR.

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.

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. Portions of the system also assist in providing hydrogen purge to the RB through purge exhaust filters following a Loss of Coolant Accident (LOCA) but AHV-1A through 1D are not required for this purpose. Containment atmosphere is displaced from the RB using the pressurization portion of the Reactor Building Leak Rate Test System which intersects with the AH-XC system between AHV-1A and the Purge Exhaust Charcoal Filters [VA,FLT]. 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 for "normal" purge operations during Operational MODES 5 and 6 with two supply fans with isolation valves and 2 exhaust fans with isolation valves. Normal purge operation may be required for internal RB inspection or maintenance but is only 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.

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. AHV-1A/1D fail closed on loss of air when the solenoids are deenergized. 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 and fail as-is on loss of power.

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, Reactor Building Purge Supply Fans [VA,FAN] AHF-6A and AHF-6B, 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 cause AHV-1A through 1D to close when the purge supply and exhaust fans stop.

An evaluation of the control power scheme for AHV-1A through 1D indicates that three other safety related components are powered from the same 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.

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". These requirements are implemented by a surveillance procedure which is performed in MODES 1 through 5 and verifies on a monthly basis, that AHV-1A through 1D are closed. AHV-1A and 1D are maintained closed by removal of air to the solenoid valves achieved by chained and locked manual air supply valves. AHV-1B and 1C are maintained closed by locked breakers. Even if a fault were to have occurred during MODES 1 through 4 in any of the Non-Class 1E circuits described above, OPERABILITY would not have been a concern because the valves are maintained closed by positive means as noted above. The requirement for closure of the valves was established as a commitment in FPC Letter 3F-0383-01 to NRC dated March 1, 1983 titled "Valve Operability of Large Pratt Butterfly Purge and Vent Valves".

In MODE 5, there are no OPERABILITY requirements in the Improved Technical Specifications pertaining to containment integrity.

In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than that established for MODES 1 through 4. In order to make this distinction, the penetration requirements in ITS 3.9.3 are referred to as "containment closure" rather than "containment OPERABILITY.".

Several failure modes were reviewed due to the non-safety/safety circuit interface concerns relative to AHV-1A through 1D. As noted above, AHV-1B/1C will fail as-is on loss of power. Two different types of shorts apply to AHV-1A/1D in the present circuitry. A hot short may open AHV-1A/1D and another type short may cause a blown fuse and close AHV-1A/1D. This is not a significant safety concern because the valves are sealed closed in MODES 1 through 4 when a LOCA might occur, and because credit is not taken for the closure of the Reactor Building Purge Valves as part of the Fuel Handling Accident analysis. As described in Technical Specification BASES B3.9.3 "Containment Penetrations", 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. 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, since the analyzed design basis accidents do not assume the function of AHV-1A through 1D, the safety significance of this event is negligible and there is no impact on the health and safety of the public due to operation outside CR-3's design basis.

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 is maintained in MODES 1 through 4 by the locked closed position of the valves. Faults in the circuitry associated with 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.

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

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

AH	Air Handling System
AH-XC	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)
MODE FIVE	Cold Shutdown
MODE SIX	Refueling
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])