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

February 8, 1996
3F0296-01

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

Subject: Licensee Event Report (LER) 96-002-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-002-00. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73.

Sincerely,

BJ Hickle
B. J. Hickle, Director
Nuclear Plant Operations

BJH/JWT

Attachment

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

IE22
1/1

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)	0 5 0 0 0 3 0 2	1 OF 1 0

TITLE (4)

Personnel Errors by Engineering 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	0 5 0 0 0
0 1	1 0	9 6	9 6	0 0 2	0 0	0 2	0 8	9 6	N/A	0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

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)
J.W. Tunstill, Sr. Nuclear Licensing Engineer	3 5 2 5 6 3 - 4 4 9 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)

YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On January 10, 1996, Florida Power Corporation's Crystal River Unit 3 (CR-3) was in MODE THREE (HOT STANDBY) shutting down due to a forced outage. While investigating corrective actions for a previous LER, a concern was noted with the Post Accident Hydrogen Purge Valves, LRV-70 through 73. A request was made to review the control power scheme associated with these valves which are part of the Post Accident Venting System. The circuitry to close these valves is routed through the Reactor Building Purge Duct Radiation Monitor cabinet, which is non-safety-related. 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 is cognitive personnel error by FPC engineers during a plant modification in 1988. The safety significance of the event is negligible since failure of the valves to close would be recognized by the operators and the valves can be closed manually from the control room. Isolation relays will be added between the RM-A1 non-safety signal and the LR valves safety-related control circuits by February 29, 1996. Engineering personnel will be briefed on this error by March 28, 1996. Other non-safety-related cabinets, panels, and motor control centers will be examined to determine if similar safety/non-safety problems exist by June 30, 1996.

EVENT DESCRIPTION

On January 10, 1996, Florida Power Corporation's Crystal River Unit 3 (CR-3) was in MODE THREE (HOT STANDBY) in the process of shutting down due to a forced outage. While investigating corrective actions for a previous LER (95-025-00), a concern was noted with the Post Accident Hydrogen Purge Valves [JM,ISV] LRV-70 through 73. An action item from an Operability Concern Resolution (OCR) Report meeting for Problem Report (PR) 95-0223 prompted the review of the control power scheme associated with these valves which are part of the Post Accident Venting [VA](LR) System. The action was prompted by a questioning attitude of an senior reactor operator trainee attending the OCR meeting. LRV-70 through 73 are containment isolation valves and receive a signal to close on certain Engineered Safety Features (ESF) actuations. It was discovered that the circuitry to close these valves is routed through the Radiation Monitor Control console in the Main Control Room. This console interfaces with the Reactor Building Purge Duct Radiation Monitor [IL,PL](RM-A1) cabinet, which is non-safety-related. A Problem Report was issued and an investigation began to determine if the control power design scheme represented a condition outside the design basis for CR-3.

The investigation revealed that an open circuit in the RM-A1 portion of the circuitry would normally cause the valves to close (i.e., move to the ESF required position). A short circuit in the RM-A1 portion of the circuitry could cause the valves to remain in their existing position and would leave intact, the capability of the valves to close on an ESF actuation. A "hot" short in the RM-A1 portion of the circuitry, however, could cause the valves to remain open despite an ESF actuation to close the valves. Based on the circuit design, this postulated failure would not cause the valves to open spuriously and the Main Control Board [MCBD] (MCB) switch for these valves would not be disabled by the postulated failure.

The investigation results were reported to the NRC at 0224 on January 13, 1996 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 29832 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 power to the valves was removed and the valves were placed under administrative controls until a complete evaluation of the effects of the problem can be made.

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.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

EVENT EVALUATION

The Post Accident Venting System (LR) has three safety functions. The system provides containment isolation for the system piping that penetrates the Reactor Building [NH](RB), provides post accident hydrogen control capability for the RB, and provides a leak rate testing capability for the RB. Portions of the system also provide a capability to depressurize the RB during power operations (MODES ONE through FOUR).

Refer to the piping arrangement shown in Figure 1. The LR valves allow purging of the RB in all operating modes. The two lines provide direct communication between the RB atmosphere and the outside atmosphere when either LRV-70 and LRV-71 or LRV-72 and LRV-73 are open. LRV-70 through 73 receive automatic closure signals from the Engineered Safeguards (ES) system on high RB pressure and on high radiation from RM-A1. RM-A1 and its associated circuitry are non-safety-related.

All four valves are 6-inch Schedule 80, Seismic Category I, solenoid operated globe valves which are energized-to-open. Therefore, loss of power closes the valves. Each of the LR valves is fed from a Vital Bus Distribution Panel (VBDP) breaker and there are no other components fed from that breaker. Therefore, faults in the non-safety-related portions of the circuit do not propagate to other safety-related components.

Refer to the circuit diagram in Figure 2. The non-safety related contact of RM-A1 is between, and in series with the ES actuation signal and the Main Control Board switch. This arrangement allows closure of the valves at all times by the MCB switch. The circuitry is routed through the non-safety-related radiation monitoring cabinet. Due to the RM-A1 contact being after the ES actuation signal and the circuit routing through the radiation monitoring cabinet, the following failure modes can be postulated:

1. An open circuit or the contacts failing in the open position of the RM-A1 portion of the circuit will close the valves. This is not a concern since the safety position for these valves is "closed" during normal full power operation.

2. A short circuit (circuit line to line short) or the contacts failing in the closed position of the RM-A1 part of the circuit will maintain the valves in whatever position they are in at the time of the failure. If open, the ES signal and the MCB switch can close the valves.

3. A hot short in the radiation monitoring cabinet can keep the valves open if they are already open, but will not open the valves since the control switch is downstream of the non-safety part of the circuit. This hot

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short can, however, prevent the valves from closing if an ES signal is received. The valves can still be closed by the MCB switch in this case.

Note that none of the failure modes can spuriously open the valves nor can a failure in these circuits propagate upstream into the balance of the ES system.

If both LR valves in the same line (LRV-70 and LRV-71 or LRV-72 and LRV-73) do not close on ES, containment integrity will be lost. For high energy line breaks inside containment such as steam line breaks and Loss of Coolant Accidents (LOCAs), containment integrity is presumed in the dose analyses presented in FSAR Section 14. Automatic closure of containment isolation valves on high RB pressure is a design feature used to ensure containment integrity after these containment pressurization events. If these valves did not close after an ES signal, it would be recognized and corrected by the operators as part of their post-ES equipment verification. Closure of the valves from the MCB would be available and is not affected by this problem. The FSAR Chapter 14 dose analyses for the Design Basis LOCA are below the limits specified in 10 CFR 100 and 10 CFR 50, Appendix A, Criterion 19 and would remain bounding for this event. The consequences to the public health and safety for this deficiency are negligible. This discussion applies in MODES ONE through FOUR, consistent with the applicability of Limiting Condition for Operations (LCO) 3.6.3, Containment Isolation Valves as described in CR-3's Improved Technical Specifications (ITS).

There are no applicable ITS sections for this equipment in MODE FIVE or in MODE SIX provided no CORE ALTERATIONS or movement of irradiated fuel in containment is in progress. There are no design basis accidents in these modes of operation for which OPERABILITY of the LRVs is credited. When CORE ALTERATIONS or movement of irradiated fuel in containment is in progress in MODE SIX, ITS LCO 3.3.15 and LCO 3.9.3 are applicable to cover the conditions where a Fuel Handling Accident could occur in containment. The Fuel Handling Accident analysis was performed under two sets of assumptions: design basis and realistic basis. Accidents occurring both inside the RB and in the spent fuel pool were analyzed. The accident inside the RB is discussed here. The assumptions of the design basis analysis are conservative and bounding for the realistic basis analysis and represent the licensing basis for this event per FSAR section 14.2.2.3.

The release of the radionuclide inventory due to a Fuel Handling Accident is described two different ways in FSAR Section 14.2.2.3. In FSAR Table 14-31, the release is described as a "plume release," indicating an instantaneous release of all the activity to the environs. The FSAR refers to the activity being released via the Reactor Building Purge System directly to the environs with no credit taken for charcoal filtration of the radionuclides. Per the FSAR and ITS BASES 3.3.15, no credit is taken for isolation of the purge system in the analysis of the event. Closure of the purge valves on high radiation is only required to maintain 10 CFR 20 limits during normal operations.

LCO 3.9.3.c.1 limits the number of penetrations providing direct access from the containment atmosphere to the outside atmosphere to less than or equal to the equivalent flow rate through a 48 inch containment purge penetration. Additional penetrations beyond the 48-inch equivalent flow criterion may be opened provided they are capable of being closed by an OPERABLE purge or mini-purge valve in accordance with LCO 3.9.3.c.2. For the LRVs, that closure signal is provided by RM-A1 on high radiation. This LCO is based on a purge flow rate of 50,000 CFM and was done in an effort to quantify the "plume release" for use in technical specifications. It also provides defense in depth beyond what the accident analysis assumes. This makes the technical specification requirement more restrictive than the assumptions of the accident analysis. The non-safety classification of RM-A1 or associated circuits does not render the LRVs inoperable.

If a design basis accident were postulated, any impact on the public health and safety would require an ES actuation while mini-purging coincident with a hot short in the radiation monitoring cabinet. FPC reviewed CR-3 history data from March 1977 to the present for ES actuations and RM-A1 failures and applied Probabilistic Safety Assessment (PSA) techniques to the mini-purge process to determine that the frequency of the simultaneous occurrence of these events is 1.1E-7/yr. This value does not represent a core damage frequency since no ECCS failures are assumed. If the CR-3 core damage frequency is substituted for the frequency of ES actuation, the resulting event frequency for the three simultaneous events is 8E-12/year. Another assessment of the likelihood of the event is the frequency of a hot short in the radiation monitoring cabinet while mini-purging. That frequency is 9.6E-7/year.

IMMEDIATE CORRECTIVE ACTION

After the investigation began on January 10, 1996, an evaluation of the control circuits for LRV-71 through 73 confirmed on January 13, 1996 that the closure function of the valves is maintained in MODES ONE through FOUR by the locked closed position of the valves. Faults in the circuitry associated with these valves will not affect the safeguards position of other safety-related components. The Nuclear Shift Supervisor on Duty (SSOD) immediately assured that the valves were closed and placed them under administrative controls until a complete evaluation of the effects of the problem could be made. The valves were listed as inoperable for MODES ONE through FOUR. Since CR-3 was in the process of shutting down due to an earlier saltwater intrusion through a condenser tube leak, entry into the ACTIONS of ITS 3.6.3 for these inoperable valves was not necessary. CR-3 entered MODE FIVE at 2116 on January 11, 1996. Following cleanup of the secondary side of the plant, CR-3 entered MODE FOUR on January 19, 1996, and entered MODE ONE on January 28, 1996.

Based upon an indeterminate conclusion regarding the impact of not meeting the electrical design basis on operability during CORE ALTERATIONS and the movement of irradiated fuel in the RB, FPC conservatively considers the valves inoperable until further evaluations are completed.

CAUSE

This condition is considered to be a result of cognitive personnel error by the FPC Nuclear Engineering Department during a plant modification in 1988. The design engineer for the subject circuits failed to recognize the design criteria for electrical isolation of Class 1E to non-Class 1E circuits.

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.

A modification to add isolation relays between the RM-A1 non-safety signal and valves LRV-70 through 73 safety-related control circuits will be completed by February 29, 1996. The Design Engineering Review Board will be briefed on this design error and a summary of this briefing will be distributed to engineering design personnel by March 28, 1996. In addition, other non-safety-related cabinets, panels, and motor control centers will be examined using a sampling approach to determine if similar safety/non-safety problems exist. This sampling will be completed by June 30, 1996.

PREVIOUS SIMILAR EVENTS

There have been three 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. LER 95-25 reported that the control power scheme associated with Reactor Building Purge, Exhaust, and Supply containment isolation valves used several safety related circuits in non-safety related terminal boxes and other unisolated devices in their routing.

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ATTACHMENT

Attachment 1 - Abbreviations, Definitions and Acronyms

FIGURE 1 - Simplified piping diagram taken from 302-722 drawing.

FIGURE 2 - Simplified circuit diagram taken from 208-067 drawing.

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

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.
OCR	Operability Concern Resolution (OCR) Report is a report documenting an evaluation comparing the intended safety function(s) of a component with accident scenario(s) it is designed to mitigate. Available information is gathered and reviewed to arrive at a recommendation concerning whether a component is operable or inoperable.
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])

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104

EXPIRES 5/31/95

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.

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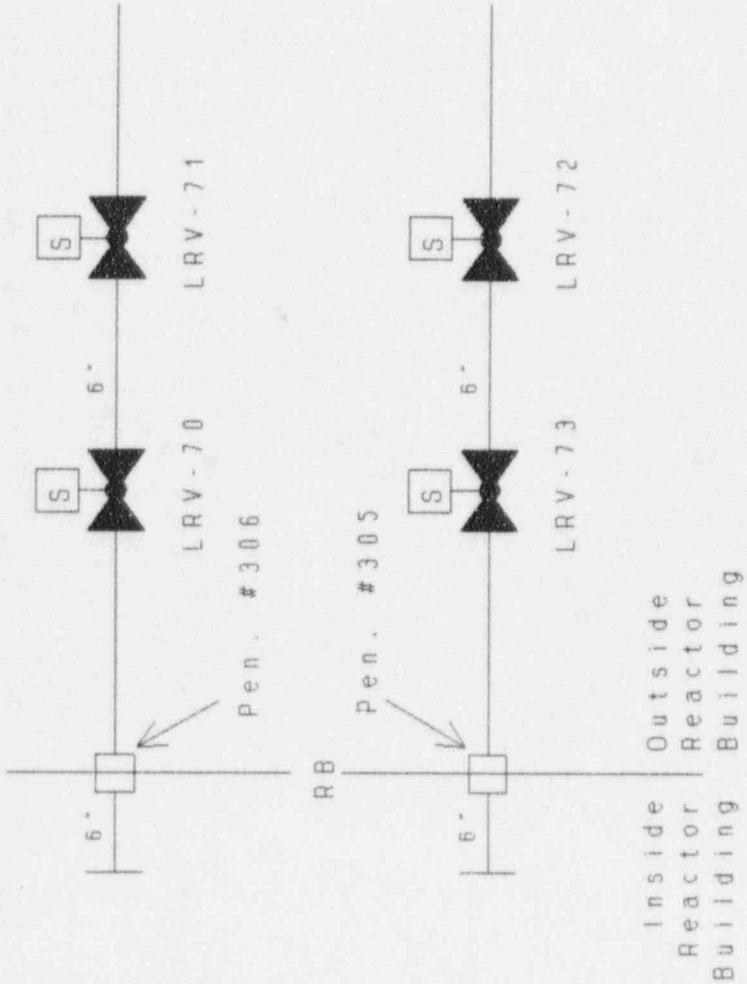

SIMPLIFIED PIPING DRAWING

FIGURE 1

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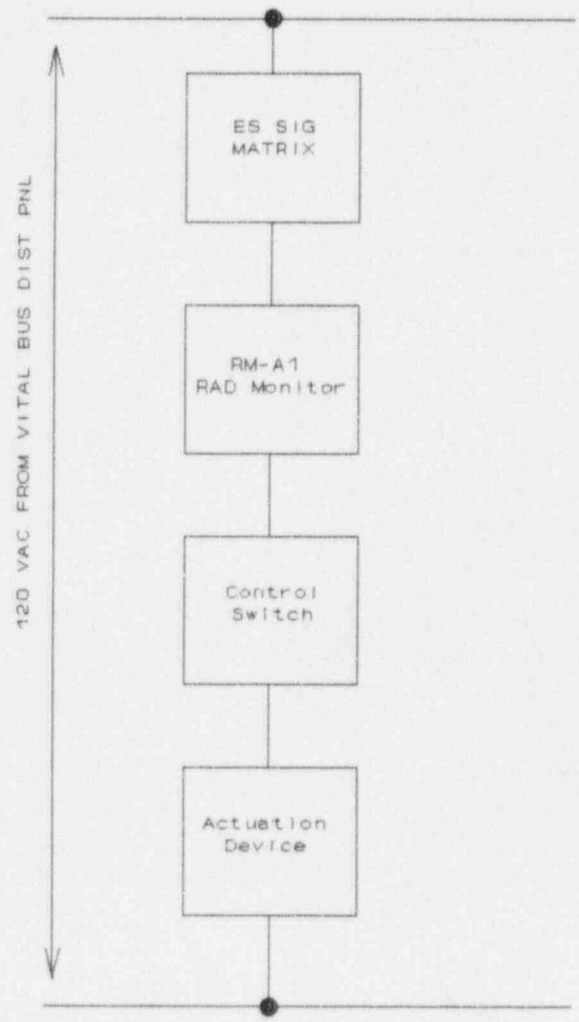

SIMPLIFIED

ELECTRICAL DRAWING

FIGURE 2