



**Florida
Power**

CORPORATION
Crystal River Unit 3
Docket No. 50-302

November 12, 1996
3F1196-10

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

Subject: Licensee Event Report (LER) 96-024-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-024-00 concerning an unanalyzed condition regarding a recent plant modification which could have prevented the fulfillment of a safety function involving emergency feedwater.

This report is submitted in accordance with 10 CFR 50.73. Florida Power Corporation intends to supplement this report by January 24, 1997 in order to provide the results of a comprehensive root cause analysis and identification of corrective actions.

Sincerely,

P.M. Beard, Jr.
Senior vice President
Nuclear Operations

PMB/TWC
Attachment

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

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EXPIRES 5/31/96

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 1 OF 0 4

TITLE (4) Plant Modification Creates Unanalyzed Condition Regarding Emergency Feedwater Availability

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)													
1	0	1	1	9	6	9	6	0	2	4	0	0	1	1	2	9	6	N/A	0	5	0	0	0
OPERATING MODE (9)		5		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)													
				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 T. W. Catchpole, Sr. Nuclear Licensing Engineer TELEPHONE NUMBER AREA CODE 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 NRPDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS

SUPPLEMENTAL REPORT EXPECTED (14) X YES (If yes, complete EXPECTED SUBMISSION DATE) NO DATE (15) 0 1 2 4 9 7

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

On October 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN). During a review of a diesel loading calculation, FPC engineers determined the calculation assumed the steam-driven Emergency Feedwater Pump (EFP) was running when the motor-driven EFP received an Engineered Safeguards (ES) automatic trip signal at Reactor Coolant System (RCS) pressure of 500 pounds per square inch gauge (psig) for Low Pressure Injection (LPI) actuation. This assumption is supported by Small Break Loss of Coolant Accident (SBLOCA) analyses. A plant modification had removed the automatic start signal from the "A" side of the Emergency Feedwater Initiation and Control system to prevent runout concerns with the steam-driven EFP during certain accident conditions when its flow control valves would fail open. This could have caused CR-3 to be in an unanalyzed condition with no emergency feedwater for the period of time between the 500 psig actuation signal and when RCS pressure is reduced below the LPI shutoff head (approximately 185 psig) when EFW is no longer required for residual heat removal. A four-hour notification was made to NRC and assigned Event Number 31141. The details of this event are being investigated by a root cause team. A supplement to this LER will be provided by January 24, 1997 after completion of the investigation.

EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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FACILITY NAME (1)

CRYSTAL RIVER UNIT 3 (CR-3)

DOCKET NUMBER (2)

LER NUMBER (6)

PAGE (3)

YEAR

SEQUENTIAL
NUMBERREVISION
NUMBER

0 5 0 0 0 3 0 2 9 6 -- 0 2 4 --- 0 0 0 2 OF 0 4

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

EVENT DESCRIPTION

On October 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN). During a review of the Emergency Diesel Generator [EK,DG] (EDG) loading calculation, FPC engineers determined the calculation assumed the steam-driven Emergency Feedwater Pump [BA,P](EFP-2) was running when the motor-driven emergency feedwater pump, EFP-1, received an Engineered Safeguards (ES) automatic trip signal at Reactor Coolant System [AB](RCS) pressure of 500 pounds per square inch gauge (psig). The automatic trip feature was installed in June, 1990 in order to prevent electrical loads on the EDG's from being greater than the 30-minute engine rating during certain design basis accident scenarios which require Low Pressure Injection [BP] (LPI) operation.

FPC engineers determined that the above assumption was invalidated by Modification Approval Record (MAR) 96-04-12-01, "ASV-204 EFIC Auto Open Removal," which was implemented in May, 1996 during Refueling Outage 10. The intent of the modification was to eliminate the automatic actuation of EFP-2 by removing the automatic opening of the redundant steam admission valve [SA,ISV](ASV-204) previously initiated from the "A" train of the Emergency Feedwater Initiation and Control [JB](EFIC) system in applicable transient conditions. This was determined to be necessary to address EFP-2 runout concerns following a loss of the "B" DC power train [EJ,BTRY] concurrent with a Loss of Offsite Power (LOOP) and Loss of Coolant Accident (LOCA). The loss of DC power condition would result in failure to control the "B" train emergency feedwater flow control valves which would fail open. EFP-2 would remain available but require operator action to cross-tie EFP-2 flow to the EFP-1 flow path controls which would have control power available during a loss of "B" DC Power train scenario.

The Emergency Feedwater System is required to assure an adequate emergency supply of feedwater to the Once-Through-Steam-Generators [AB,SG](OTSG) to ensure a heat sink to remove reactor decay heat until suitable conditions are attained to start the Decay Heat Removal System [BP](DH). The Decay Heat system functions as the LPI part of the Emergency Core Cooling System (ECCS) for LOCA's in the RCS. In the event of a small break LOCA, RCS pressure may not decrease to the point where the LPI pumps [BP,P] can inject water into the vessel [RPV]. Analyses exist indicating simultaneous operation of LPI and EFW is not required; however, there is presently no analysis to support operation without EFW between RCS pressures of 500 psig when the LPI pumps start, and the point at which RCS pressures are below the LPI shutoff head (approximately 185 psig). Therefore, with no credit taken for the manual operator action described above to cross-tie EFP-2 to EFP-1 flow path controls, between the time frame of startup from Refueling Outage 10 on May 17, 1996 and September 6, 1996 when the plant was taken to MODE 5 for a forced outage, CR-3 operated in an unanalyzed condition.

A Problem Report was initiated to describe the above conditions and was presented to the Shift Supervisor on Duty (SSOD) at 1750 hours on October 11, 1996. The SSOD then made a four-hour event notification at 1812 hours in accordance with

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (8)			PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
		0 5 0 0 0 3 0 2	9 6	0 2 4	

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

10CFR50.72(b)(2)(i) as an event found while the reactor was shut down that had it been found while the reactor was in operation, would have resulted in the plant being in an unanalyzed condition. This issue was assigned Event Number 31141.

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(A) to describe an unanalyzed condition that could have significantly compromised plant safety.

This event is the second reported issue related to the modification which removed the automatic open signal from ASV-204. LER 96-020 was provided as a voluntary report on October, 1996 to describe the impact of that modification on EDG loading. A supplement to this report will be provided by January 24, 1997 after completion of a comprehensive root cause analysis.

EVENT EVALUATION

The Emergency Feedwater (EFW) system provides secondary coolant to the Once Through Steam Generators (OTSG) in the event the Main Feedwater System [SJ] is rendered inoperable and is unable to perform this function.

For Small Break LOCA events, the EFW system will be initiated by an ES signal. Current procedures also require that the Reactor Coolant pumps [AB,P] be tripped when required subcooling margin is lost. Under these conditions, the EFIC system should raise the level in the steam generators to the natural circulation setpoint. This process should take approximately 30 minutes, unless EFW is further throttled to avoid exceeding cooldown rate limits. The purpose of raising the level is to assist in establishing steam condensation natural circulation (Boiler-Condenser Cooling) if voiding has occurred in the primary system. During this time, there is substantial EFW flow high in the OTSG which provides good heat transfer.

The circumstances described in this LER concerning a Small Break LOCA concurrent with a Loss of Offsite Power with a loss of the "B" DC power train and loss of emergency feedwater capability are outside existing analyses for CR-3 which assume continual High Pressure Injection [BQ] and Low Pressure Injection or Emergency Feedwater flow to mitigate accidents.

CAUSE

FPC management has classified this event as a significant condition adverse to quality and formed a root cause team under the direction of the Nuclear Safety Assessment Team (NSAT). FPC has recently adopted the methods developed by FPI International, a leading expert in prevention and investigation of organizational and programmatic failures. The root cause team will evaluate the event with respect to inappropriate actions, key activities, programs and work processes, and failure modes. The results of this investigation will be provided in a supplement to this LER.

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	0 5 0 0 0 3 0 2	9 6	0 2 4	0 0	0 4 OF 0 4

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

IMMEDIATE CORRECTIVE ACTION

Due to the EFW/EDG issues, and some other design-related issues, FPC management made a decision to remain shut down in the current forced outage until these issues are adequately addressed. The outage scope of work was provided to NRC by FPC Letter 3F1096-22 dated October 28, 1996 and includes a description of planned modifications to address this and other issues.

ADDITIONAL CORRECTIVE ACTION

Additional corrective actions will be defined upon completion of the root cause analysis described above.

ACTION TO PREVENT RECURRENCE

Actions to prevent recurrence will be defined upon completion of the root cause analysis described above.

PREVIOUS SIMILAR EVENTS

This appears to be the first event involving an unanalyzed condition concerning emergency feedwater.

There have been two previous events involving the emergency feedwater system reported in accordance with 10CFR50.73(a)(2)(v) in which the condition was determined to have prevented the fulfillment of a safety function. LER 85-027 reported a condition wherein the steam-driven EFP was disabled per procedure and the motor-driven pump was disabled due to a spurious EFIC actuation while calibrating EFIC instrumentation. A second spurious actuation occurred resulting in no EFW response. LER 88-008 reported a condition wherein three out of four EFIC level transmitters were found sufficiently out of tolerance to have prevented the actuation of emergency feedwater to the "B" OTSG.

LER's 94-006, 95-015, and 95-016 reported setpoints for EFIC system instrumentation determined to be non-conservative relative to revised analyses using new setpoint methodology which resulted in questioning the systems ability to perform its intended safety function.

On October 10, 1996, FPC provided a voluntary LER to describe an unreviewed safety question (USQ) involving the EDG loading calculation that was developed in support of the plant modification which removed the automatic open signal from ASV-204.

ATTACHMENT

None