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

June 27, 1997
3F0697-24

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

Subject: Licensee Event Report (LER) 97-011-00

Gentlemen:

Please find the enclosed Licensee Event Report (LER) 97-011-00. This report is submitted by Florida Power Corporation, in accordance with 10 CFR 50.73, to report a condition outside the design basis involving pressure locking of emergency feedwater system isolation valves resulting in the potential for interruption of emergency feedwater flow to a steam generator.

Sincerely,

J. J. Holden

J. J. Holden, Director
Nuclear Engineering and Projects

JJH/twc

Attachment

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

JE22 1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104

EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Potential design basis Condition from Pressure Locking of Emergency Feedwater Isolation Valves Caused by Insufficient Oversight/Review of Calculations

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 NAME	DOCKET NUMBER
05	29	97	97	-- 011 --	00	06	27	97	FACILITY NAME	DOCKET NUMBER

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	N		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	000		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)	50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)	73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T. W. Catchpole, Senior Licensing Engineer	(352) 563-4601

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED 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
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 29, 1997, Florida Power Corporation's Crystal River Unit 3 was in MODE 5 (COLD SHUTDOWN). A reportability determination was made based on a condition previously discovered on March 15, 1997 during the system readiness review of the Emergency Feedwater System (EFW). A re-evaluation of EFW isolation valves EFV-32 and EFV-33 determined they are potentially susceptible to pressure locking during a very specific set of conditions in which a steam environment is caused by a pipe break and the valves cannot reopen after they close on a steam generator [AB,SG] overfill signal. If pressure locking did occur, EFW would not be able to provide a continuous emergency supply to the steam generators, possibly placing the plant in a condition outside its design basis. This condition was caused by an inadequate calculation prepared to assess the potential for occurrence of pressure locking and thermal binding (PLTB). The planned installation of cavitating venturis in the EFW pump discharge lines will reduce the maximum flow rate and increase the time to overfill the steam generators. EFV-32 and EFV-33 will be modified or replaced prior to plant restart. The PLTB calculation will be reviewed by October 26, 1997. The procedure for preparation of calculations has been revised to ensure system engineers and plant operations review assumptions and results.

EVENT DESCRIPTION

On May 29, 1997, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN). A reportability determination was made based on a condition previously discovered on March 15, 1997 during the system readiness review of the Emergency Feedwater System [BA](EFW). Reviews are being conducted of selected systems to provide assurance of conformance with design and licensing basis requirements in preparation for restart of the unit. On March 15, 1997, it was identified that FPC's pressure locking and thermal binding (PLTB) evaluation may not have adequately accounted for both closing and re-opening of the emergency feedwater isolation valves [BA,ISV], EFV-32 and EFV-33, by automatic Emergency Feedwater Initiation and Control [JB](EFIC) actuations. A precursor card was generated and the condition was determined to be a possible design basis issue requiring further evaluation. EFV-32 and EFV-33 are flexible wedge, motor operated gate valves. The re-evaluation of these valves determined EFV-32 and EFV-33 are potentially susceptible to pressure locking during a very specific set of conditions in which a steam environment is caused by a pipe break and the valves cannot reopen after they close on a steam generator [AB,SG] overfill signal. If pressure locking did occur, emergency feedwater would not be able to provide a continuous emergency supply to the steam generators to remove reactor decay heat.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B) as a condition outside the design basis.

EVENT EVALUATION

The conditions for an event which could cause the postulated pressure locking are as follows. Refer to Figure 1.

1. A pipe rupture occurs in a steam line or main feedwater pipe associated with the 'A' steam generator. The pipe rupture must take place in the Intermediate Building [VK] and create a high temperature environment in the emergency feedwater pump room in excess of 400 degrees Fahrenheit (F). Depending on the break size, the steam generator will either isolate quickly (large breaks) with the temperature in the building returning to below 150 degrees F in approximately 15 minutes, or cause the steam generator to blow down for up to 20 minutes (small breaks) until operator action is implemented to isolate the break.

2. Main feedwater [SJ](FW) is isolated and emergency feedwater actuates on low steam generator pressure (less than 600 pounds per square inch gauge).

3. The 'A' steam generator is isolated by a feed-only-good-generator (FOGG) signal or by manual operator action.

4. One of the two emergency feedwater control valves [BA,FCV] (EFV-55 or EFV-57) to the 'B' steam generator fails open. The control valve failure is the single failure in this scenario;

therefore, both EFW pumps continue to function providing secondary heat removal via the 'B' steam generator.

5. The flow path with the failed control valve continues to fill the 'B' steam generator to the overfill setpoint which causes both isolation valves EFV-32 and EFV-33 to close.

6. Boil-off in the 'B' steam generator to the overfill reset point takes greater than five minutes and both isolation valves are assumed to pressure lock due to the high temperature environment created by the line break.

7. After boil-off in the steam generator to the overfill reset point, the isolation valves attempt to reopen but can not due to the actuator being incapable of overcoming the additional thrust requirements resulting from pressurized fluid in the valve bonnet caused by thermal expansion of the confined fluid.

Existing plant procedures and training call for close monitoring of EFW operation during an event of this nature. Therefore, there is a high probability the operator would note the uncontrolled flow of the EFW train and stop it prior to overfill. The operator could do this by stopping the associated pump or closing the associated isolation valve before the automatic overfill causes closure of both isolation valves. Safety-related power supplies and manual override of the EFIC system are provided for EFV-32 and 33 to accomplish this. However, based on a simulator run performed on May 26, 1997, there is approximately five to ten minutes before the steam generator water level reaches overfill. Therefore, it was determined that operator action may be required prior to ten minutes after the event. Per NRC guidance, this action cannot be assumed until ten minutes. There are numerous conditions and associated steam generator levels that could be postulated concurrent with this event. The inability to confirm that every event would provide for a full 10 minutes for operator action prior to automatic closure of EFV-32 and 33 resulted in the determination that the above specific conditions constituted a design basis issue.

This set of conditions results in a loss of main and emergency feedwater. This event is adequately addressed within CR-3's emergency operating procedures, although the mitigation steps rely on a combination of safety related and non-safety related equipment. Based on inadequate heat transfer conditions, the emergency operating procedures require initiation of high pressure injection [BQ](HPI) flow and opening the power operated relief valve [AB,RV](PORV) to reduce reactor coolant pressure while establishing feedwater using auxiliary feedwater pump [SJ,P] FWP-7. This auxiliary source of secondary cooling was installed in April 1993. The purpose of this pump and associated components was to provide an additional, non-safety related source of secondary heat removal to the steam generators in emergency conditions only if main feedwater and emergency feedwater are unavailable. This satisfied NRC Generic Issue 124 with respect to emergency feedwater reliability criterion. In addition, with offsite power available, main feedwater could be reinitiated after operators were able to isolate the affected line. The OTSGs could boil dry; however, the reactor coolant system [AB](RCS) does not overheat.

The emergency feedwater isolation valves, EFV-11 and EFV-14, in the flow path to the 'A' steam generator were replaced in 1992 with valves equipped with an integrated relief from the valve bonnet to the upstream side. Therefore, a similar set of conditions involving a line associated with the 'B' steam generator would not provide the same results described in the above scenario.

This postulated event would not have compromised the health and safety of the general public.

CAUSE

Calculation M94-0003, "Pressure Locking/Thermal Binding Evaluation," was issued in May 1994 to evaluate PLTB concerns defined in NUREG 1275, Volume 9, "Pressure Locking and Thermal Binding of Gate Valves." The calculation stated that once the EFW isolation valves are closed, they are not required to reopen to perform a safety function. This evaluation failed to account for potential single failure of an EFW control valve during a pipe rupture and the possible affect on pressure locking. The cause was insufficient oversight of the calculation created by an inadequate procedure which did not identify the need for review of calculation assumptions and results by personnel most knowledgeable of emergency feedwater system and controls. An interdisciplinary review involving the system engineer and plant operations personnel was not obtained by the personnel performing the PLTB evaluation.

IMMEDIATE CORRECTIVE ACTION

There is no concern in MODE 5 since pipe ruptures are not postulated in MODE 5 and emergency feedwater is not required to be available in MODE 5 per the Technical Specifications.

ADDITIONAL CORRECTIVE ACTION

FPC determined the need to install cavitating venturis in the EFW pump discharge lines due to concerns regarding pump runout and net positive suction head (NPSH). This was a corrective action described in event reports 96-024-01 and 97-001-00. The installation of the cavitating venturis will reduce the maximum flow rate and increase the time to overfill the steam generators, thereby aiding in the resolution of the condition described in this event.

ACTION TO PREVENT RECURRENCE

EFV-32 and EFV-33 will be modified or replaced by October 26, 1997, to preclude the possibility of pressure locking.

Calculation M94-0003 will be reviewed by October 26, 1997, to confirm the function of each valve where credit was taken for the valve position.

Nuclear Engineering Procedure NEP-213, 'Design Analysis/Calculations,' has been revised to include a requirement for a calculation review form to direct the review of input, assumptions and

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

results by the system engineer, plant operations, and other organizations, as applicable. Calculation M94-0003 would be subject to these reviews if generated under the current procedure requirements.

PREVIOUS SIMILAR EVENTS

This is the first reported event by CR-3 involving pressure locking and thermal binding.

ATTACHMENTS

Figure 1, Emergency Feedwater System

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)