



**Florida
Power**
CORPORATION

Crystal River Unit 3
Docket No. 50-302

May 28, 1992
3F0592-18

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

Subject: Licensee Event Report (LER) 92-04

Dear Sir:

Enclosed is Licensee Event Report (LER) 92-04 which is submitted in
accordance with 10 CFR 50.73.

Sincerely,

G. L. Boldt
Vice President
Nuclear Production

EEF:mag

Enclosure

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

010022

9206020022 920528
PDR ADOCK 05000302
S PDR

7E22

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 (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, 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 5
--	--------------------------------------	----------------------

TITLE (4)
Emergency Feedwater Block Valves Fail To Close Due To Degraded Valve Condition And Inadequate Motor Operator Capabilities

EVENT DATE (5)			LER NUMBER (8)			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										
0	4	2	8	9	2	0	0	4	0	0	0	5	2	8	9	2	N/A	0	5	0	0	0

OPERATING MODE (9) 1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §. (CHECK ONE OR MORE OF THE FOLLOWING): (11)										
POWER LEVEL (10) 0 6 5	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)							
	20.405(a)(1)(i)	50.38(c)(1)	50.73(a)(2)(v)	73.71(c)							
	20.405(a)(1)(ii)	50.38(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)(ix)									

LICENSEE CONTACT FOR THIS LER (12)											TELEPHONE NUMBER			
NAME W. A. Stephenson, Nuclear Safety Supervisor											AREA CODE 9 0 4 7 9 5 - 6 4 8 6			

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											
X	B	A	I	S	V	V	0	8	5	Y	X	B	A	8	4	L	2	0	0	Y
X	B	A	I	S	V	C	6	8	4	Y										

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 fifteen single-space typewritten lines) (16)

On April 24, 1992, Crystal River Unit 3 (CR-3) was in MODE 1 (POWER OPERATION) at 65.5% of Rated Thermal Power (RTP). During April 1992, Florida Power Corporation (FPC) engineering personnel were revising a test procedure associated with differential pressure (D/P) testing emergency feedwater (EFW) block valve EFV-14 in accordance with Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-10. The maximum D/P previously used in earlier testing and evaluations was determined to not represent worst case conditions. At that time, it was decided to close EFV-14 and a similar valve EFV-11 as an interim corrective action. On May 1, with CR-3 in MODE 5 (COLD SHUTDOWN) at 0% RTP, further testing revealed that none of the EFW block valve would fully close against the calculated worst case D/P. The root cause of the inability of the valves to close is attributed to valve condition due to normal wear. The affected EFW block valve/motor operator/cable combinations will be modified to enable the valves to meet the D/P test requirements.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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 (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, 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)		LER NUMBER (6)			PAGE (3) 0 2 OF 0 5
	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
	0 5 0 0 0 3 0 2	9 2	0 0 4	0 0		

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

EVENT DESCRIPTION:

On October 13, 1991, Crystal River Unit 3 (CR-3) was shut down in MODE 5 (COLD SHUTDOWN) at 0% of RATED THERMAL POWER (RTP) and 0 MW_e. During testing conducted on Emergency Feedwater (EFW) valve [BA, V] EFV-14, it was determined that the valve would stroke open satisfactorily, but would not stroke fully closed against 1445 pounds per square inch differential (psid) pressure across the valve. EFV-14 is the block valve isolating the motor [MO] driven EFW pump [P], EFP-1 discharge, to the "A" Once Through Steam Generator (OTSG) [SG]. The calculated worst case accident differential pressure (D/P) was 1367 psid across the valve (see Attachment 1, CR-3 Emergency Feedwater System). A report to the Nuclear Regulatory Commission (NRC) was not made at that time because the assumptions in the calculation were considered to be overly conservative and it was expected that the actual worst case D/P would be less than the 1367 psid value. The valve was considered operable because the valve had passed the D/P test at 1265 psid and the expected worst case D/P was expected to be much lower than this value. The revised March 6, 1992 calculation confirmed the worst case D/P would be less than 1367 (1219 psid). However while Florida Power Corporation engineering personnel were developing the procedure revisions to retest the valve a new problem was identified. Flow assumptions in the calculation were revised based on detailed review by several System Engineers and trial simulator scenarios. It was determined that the complex nature of the wide variety of transients prevented accurate analytical prediction of worst case D/P. It was therefore decided to bound the worst case D/P with total head of the motor driven emergency feedwater pump with recirculation flow. This revision of the calculation was issued April 28, 1992.

Historically, the standby position for EFW block valves has been fully open to help assure EFW flow to the steam generators if required. On April 24, 1992, as a result of the unverified results from the reevaluation of the maximum expected D/P calculation, a decision was made to close EFV-14 and similar valve EFV-11 until the upcoming refueling outage. EFV-11 is the EFW block valve from the turbine [TRB] driven EFW pump, EFP-2, to the "A" OTSG. Both valves would have stroked open automatically if required by the EFW Initiation and Control System (E-IC) [JB] or as needed in manual via operator action. Both valves had been cycled open, timed, and found to be well within the EFW actuation response time requirements listed in the plant Technical Specifications (T.S.). Additionally, EFV-14 had been tested in the open direction at 1445 psid and had performed satisfactorily. Maintaining EFV-11 and EFV-14 in a closed standby position was thus determined to negligibly reduce the reliability of EFW. It was decided at the same time to maintain EFV-32 and EFV-33, which were produced by a different manufacturer than EFV-11 and 14, in the normal standby position of open. EFV-32 is the EFW block valve isolating EFP-2 discharge from the "B" OTSG and EFV-33 is the EFW block valve isolating EFP-1 discharge from the "B" OTSG. In addition, the plant T.S. require a response time of 50 seconds for full EFW actuation on low OTSG level. The stroke time of EFV-32

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST 60.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, 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)		LER NUMBER (6)			PAGE (3) 0 3 OF 0 5
			YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	0 5 0 0 0 3 0 2	9 2	0 0 4	0 0		

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

and EFV-33 has been determined to be 60 seconds vice the 30 second stroke time exhibited by EFV-11 and EFV-14, therefore EFV-32 and EFV-33 could not have met the T.S. requirement if maintained in the closed position.

EFV-11 and EFV-14 are Chapman 6" gate valves actuated by Limitorque type SMB-0 motor operators. Each motor operator is rated at 25 foot-pounds (ft-lb,) torque, 1.805 motor horsepower, 1900 revolutions per minute (RPM) motor speed, with a unit ratio of 51.8:1. EFV-32 and EFV-33 are Velan 6" gate valves, also equipped with Limitorque type SMB-0 motor operators; however, the motor operator ratings are significantly different with each operator exhibiting 15 ft-lb, torque, 1.083 motor horsepower, 1900 RPM motor speed, and a unit ratio of 102.6:1.

On April 28, 1992, the bounding calculations associated with the reevaluation of maximum expected D/P across EFV-14 were finalized, resulting in a revision from 1219 psid up to 1501 psid. At that time, CR-3 was operating in MODE 1 (POWER OPERATION) at 65.5% of RTP and 522 MW_e, and was preparing to shut down for refueling. The event was then reported to the NRC at 1653 via the Emergency Notification System (ENS) per 10CFR50.72(b)(1)(ii)(B). On May 1, with the plant in MODE 5 (COLD SHUTDOWN) at 0% RTP, 0 MW_e and preparing to enter MODE 6 (REFUELING), EFW block valves EFV-11, EFV-32, and EFV-33 were stroke tested at full D/P, at which time it was determined that none of these valves would stroke fully closed against the revised worst case accident D/P of 1501 psid across the valves. The NRC was notified of the event via ENS at 0136 on May 2 per 10CFR50.72(b)(1)(ii)(B). This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B).

There are no requirements in the plant T.S. regarding either the low OTSG pressure EFW isolation or the OTSG overflow protection; thus, the EFW system was never considered inoperable as a result of the D/P testing.

CAUSE OF EVENT:

The inability of all four EFW block valves to stroke fully closed during the D/P testing has been attributed to normal degradation in the general condition of the valves due to wear and tear experienced during plant operation and previous surveillance testing. Recent industry information has revealed the operator's thrust ratings may not be sufficient for this application.

EVENT ANALYSIS:

These four EFW block valves function primarily to isolate the OTSGs from EFW sources when certain conditions are met. The two specific conditions that require isolation are low OTSG pressure and high OTSG level. EFIC design provides automatic closure for EFW block valves and the associated EFW control valves [FCV]

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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 (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, 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)		LER NUMBER (6)			PAGE (3) 0 4 OF 0 5
			YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	0 5 0 0 0 3	0 2	9 2	0 0 4	0 0	

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

for these conditions. The isolation on low pressure serves to prevent overcooling of the Reactor Coolant System (RCS) [AB] during accident scenarios such as a main steamline [SB] rupture, whereas the high level isolation provides OTSG overfill protection and thus prevents EFW carryover into the main steam lines. EFW block valves and control valves receive open/control signals on EFIC actuations provided the described conditions do not exist. The overfill condition does not produce high D/P conditions.

From the standpoint of potential valve performance during a plant casualty, it is important to note that EFV-14 and EFV-33 were successfully stroked to their fully closed positions against D/P values of 1265 psid and 1274 psid, respectively, during testing conducted in 1987. It is highly unlikely that any transient situation would result in such a high D/P condition, unless the calculated D/P value of 1501 psid. The failure to achieve isolation of the affected OTSG during the worst case steam line rupture would result in minimal overcooling of the RCS. There would be negligible effect on the consequences of a steam line break to the general public or site personnel as a result of EFW block valve performance.

CORRECTIVE ACTION:

The affected EFW block valves and/or their associated motor operators will be modified such that all four EFW block valves ultimately can be verified via testing to meet the worst case closing D/P requirement of 1501 psid.

Results of this testing and similar testing conducted at other nuclear facilities are being factored into FPC's GL 89-10 Motor Operated Valve Program to determine if similar problems exist with other valves. This will include an in-depth evaluation of calculations involved in determining closing D/P requirements. Preliminary results indicate that one additional valve will be modified prior to restart.

In addition, the motor operators of all valves are being modified in light of new information from the GL 89-10 program.

PREVIOUS SIMILAR EVENTS:

There have been no previous events where safety-related motor operated valves have been unable to meet design basis closing D/P requirements.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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 (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, 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)		LER NUMBER (5)			PAGE (3) 0 5 OF 0 5
			YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	0 5 0 0 0 3 0 2		9 2	0 0 4	0 0	

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

ATTACHMENT 1
CR-3 EMERGENCY FEEDWATER SYSTEM

