NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	1 OF 0 3

TITLE (4)
FEEDWATER TRANSIENT CAUSES REACTOR COOLANT SYSTEM HIGH PRESSURE TRIP

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)
0 8	2 0	8 5	8 5	— 0 1 6 —	0 0	0 9	2 6	8 5	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 2 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)	X 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)	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)
W. K. Bandhauer, Nuclear Safety Supervisor	9 0 4 7 9 5 - 6 4 8 6

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
X	J A	P D C	B 0 4 5	Y	A	S J	F C V	F 1 3 0	Y
E	J A	A I C	B 0 4 5	Y					

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 August 20, 1985, Crystal River Unit 3 was operating at 20% reactor power while conducting a plant startup. With the exception of the Turbine Generator, all major control stations of the Integrated Control System were in the manual control mode. At 1823 while raising plant power, a slight Reactor Coolant System (RCS) heat removal cyclic transient developed due to sluggish feedwater control valve response. When the feedwater control valve and reactor control stations were placed in automatic control, a reactor power oscillation developed due to incorrect control module settings. The feedwater control valve and reactor control stations were returned to manual control. However, the RCS pressure and temperature perturbations reappeared. Subsequent operator attempts to manually control the oscillations were not successful and the reactor tripped on High RCS Pressure.

The sluggish response of the feedwater control valve was caused by a crimped air line and a mechanically misaligned air control valve within its pneumatic controller. The air line was replaced and the air control valve was aligned. The reactor control module settings were adjusted to the recommended values. Upgraded administrative controls will be implemented for the control module settings.

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NRC Form 366A
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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	8 5	— 0 1 6	— 0 0	0 2 OF 0 3

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

EVENT DESCRIPTION

On August 20, 1985, Crystal River Unit 3 was operating at twenty percent reactor power while conducting a plant startup. With the exception of the Turbine Generator (TA,TG) all major stations of the Integrated Control System (ICS) (JA) were in the manual mode of control. Automatic control of the main feedwater pump (SJ,P) had been erratic. Feedwater control valve (SJ,FCV) response was sluggish.

At 1823 manual escalation of unit power resulted in a small cyclic mismatch of Reactor Coolant System (RCS) (AB) heat removal. In response to oscillations of 6^{o}F RCS temperature and 130 psig RCS pressure, the operator placed all major ICS control stations except the feedwater pump in the automatic control mode. When the reactor control station (JA,JIK) was placed in automatic control, the control rods (AA,ROD) began cycling. Reactor power oscillations of 2.5% occurred and began to slowly increase in amplitude. The feedwater control valve response remained sluggish.

At 1838, the reactor and feedwater valves ICS control (JA,PDC) stations were returned to the manual control mode to stop the reactor power oscillations. However, the RCS temperature and pressure oscillations returned. Operator attempts to control the perturbations were unsuccessful and the reactor automatically tripped at 1842 on high Reactor Coolant System pressure.

Because the feedwater control stations were in the manual mode of control, the feedwater flow rate was not reduced at the normal rate after the reactor trip. This led to lower than expected RCS temperature (547^{o}F) and pressurizer level (32 inches). RCS temperature and pressurizer levels were returned to normal post-trip values when feedwater flow rates were sufficiently reduced by the operator.

CAUSE

The RCS pressure and temperature transients are attributed to the sluggish main feedwater control response. The feedwater control valve air line was found to be partially crimped and the movement of its air control valve was being restricted due to mechanical misalignment. Thus a corresponding larger feedwater flow error signal was required to cause feedwater valve movement. The reason for the crimped air line and misaligned valve could not be determined. This condition was probably caused by inadvertent bumping of the controller and air line during maintenance or operational activities. Although the automatic control feature of the feedwater pump was erratic, it was not a major contributor to the cause of the event. However, if satisfactory automatic control of the feedwater pump had been available, the magnitude of the transient may have been reduced.

The reactor power transient is attributed to improper settings on a reactor control module. The settings were probably established during the plant power "coast-down" prior to the 1985 refueling and modification outage. The discovered settings would be appropriate to counter the expected plant oscillations which occur at sixty percent reactor power due to steam generator flow resonances.

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

SAFETY CONSIDERATIONS

The Reactor Protection System (RPS) (JC) responded as designed by automatically tripping the control rods when the high Reactor Coolant System pressure trip setpoint was sensed.

The reactor core was cooled normally through the secondary plant. Use of engineered safety features other than RPS was not required.

Due to the already existing small amount of steam generator tube leakage, a radioactive release was made via the main steam safety (SB,RV) and atmospheric dump (SB,PCV) valves. This release did not cause any radioactive limits to be exceeded.

No safety equipment was damaged or otherwise made inoperable during this event.

CORRECTIVE ACTIONS

The crimped air line for the feedwater control valve was replaced and the air control valve was properly aligned. The defective feedwater control module was replaced, and the reactor control module settings were adjusted to the recommended values. Since repair of these control modules is designed to consist of module replacement, no further action is planned.

PREVIOUS SIMILAR EVENTS

Since reducing the reactor trip setpoint to 2300 psig in 1979, CR-3 has experienced sixteen RCS high pressure automatic reactor trips. Three of these trips have been directly related to feedwater or reactor control malfunctions. Although none of these previous trips has the same set of causes, the defective feedwater flow control module may have been the cause of erratic feedwater pump automatic control at low feedwater flow rates that has been observed throughout plant life.

Florida Power
CORPORATION

September 26, 1985
3F0985-21

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Licensee Event Report No. 85-016-00

Dear Sir:

Enclosed is Licensee Event Report (LER) No. 85-016-00 which is submitted in accordance with 10 CFR 50.73.

Should there be any questions, please contact this office.

Sincerely,

G. R. Westafer

G. R. Westafer
Manager, Nucler Operations
Licensing and Fuel Management

AEF/feb

Enclosure

cc: Dr. J. Nelson Grace
Regional Administrator, Region II
Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, GA 30323

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