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

August 6, 1996
3F0896-09

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

Subject: Licensee Event Report (LER) 96-017-01

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 96-017-01. This supplemental report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73.

This supplement provides clarifications regarding the sequence of operator actions after the May 31, 1996 Reactor Trip at Crystal River Unit 3. An error in editing the original LER resulted in a sentence structure which indicated that a reduction in pressure occurred after a review of post-trip data when in fact, the action occurred within one minute of the trip. A similar error indicated Emergency Feedwater Pump packing was loosened after entering Improved Technical Specification actions when in fact, the actions were entered as a result of loosening the packing to ensure a post maintenance test was accomplished.

Sincerely,

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

PMB/TWC

Attachment

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

IE22
1/1

NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	1 OF 0 7

TITLE (4) Reactor Trip on High Reactor Coolant Pressure During Turbine Testing Caused by Debris in Manual Isolation Valve

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
									N/A	0 5 0 0 0
0 5	3 1	9 6	9 6	— 0 1 7 —	0 1	0 8	0 6	9 6	N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	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	
T. W. Catchpole, Sr. Nuclear Licensing Engineer	3 5 2	5 6 3 - 4 6 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 fifteen single-space typewritten lines (16)

On May 31, 1996, Florida Power Corporation's Crystal River Unit 3 was operating at 100% power generating 879 megawatts. While testing the turbine trip devices, an unexpected isolation of the steam flow path to the Main Turbine resulted in a reactor trip due to high Reactor Coolant System pressure. The Reactor Protection System performed as designed and operators responded properly per Emergency Operating Procedures. Several anomalies were noted during the event: a Turbine Bypass Valve did not control properly; two Main Steam Safety Valves (MSSV) did not reseat properly; Emergency Feedwater Initiation and Control (EFIC) actuated on low steam generator level limits; and steam was noted coming from the inboard seal on Emergency Feedwater Pump EFP-1. The cause of the event was debris in a manual isolation valve preventing it from fully seating thus allowing Electro-Hydraulic Control (EHC) fluid to drain. A post-trip analysis addressed completion of main feedwater/ICS trip data reviews, EHC fluid flushes on all Turbine Trip Solenoid Valves, concerns regarding EFP-1 seal leakage and verification of lift setpoints for the MSSV's. Other actions will include enhancements to the EHC post-maintenance and turbine check procedures, a detailed evaluation of ICS response to low steam generator levels, and possible changes to turbine bypass and MSSV's.

EVENT DESCRIPTION

On May 31, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 100% power and generating 879 megawatts. During the performance of the "Turbine Generator Checks" procedure (PT-325), CR-3 experienced an unexpected isolation of the steam flow path to the main turbine [TA,TRB] which resulted in a reactor [AC] trip approximately 3 seconds later due to high Reactor Coolant System [AB](RCS) pressure. Prior to the reactor trip, all Integrated Control System [JA](ICS) hand/auto stations were in the automatic mode.

At 2123 hours, CR-3 operators were performing the Turbine Trip Solenoid Valve Test section of PT-325. This procedure is intended to test equipment such as Turbine Valves [SB,PCV], Electro-Hydraulic Control [TG](EHC) system components, and turbine trip devices [TA,20] while the plant is on line. After the operator properly performed prerequisite steps, the test switch associated with Turbine Trip Solenoid Valve [TA,20](TB-247-SV) was placed in the test position. The turbine throttle and governor valves [SB,PCV] then closed unexpectedly, isolating the main steam flow path and thereby increasing RCS temperature and pressure. The Reactor Protection System channels [JC,CH](RPS) tripped on high RCS pressure causing Control Rod Drive [AA,BKR](CRD) breakers to trip, resulting in a reactor trip. The operators performed the immediate actions of their Emergency Operating Procedures (EOP) for a reactor trip and noted several anomalies during the event as follows:

One of the four Turbine Bypass Valves [SB,PCV] MSV-11 remained closed following the trip. At 0300 hours on June 1, 1996, the Nuclear Shift Supervisor (NSS) entered Improved Technical Specification (ITS) 3.7.4 Required Action A.2 (verify operability of associated Atmospheric Dump Valve[SB,PCV]) and isolated MSV-11 for troubleshooting. Maintenance personnel subsequently corrected the response of MSV-11 by adjusting the positioner for the valve.

Two Main Steam (MS) Safety Valves [SB,RV] MSV-34 and MSV-36 did not reseat properly following the trip. Each of the four MS lines is equipped with four safety valves to provide over-pressure protection in the event of a load rejection or main turbine trip. Plant data indicates the valves initially lifted at the proper pressure of 1050 pounds per square inch gauge (psig). Within one minute of the reactor trip, operators reduced Main Steam pressure by 10 psig to reseat the valves. During the next two-hour period, the valves intermittently lifted and reseated at approximately 1000 psig and 960 psig, respectively. After completing actions per post-trip procedures, operators reduced pressure again by 25 psig, to maintain the valves closed. The 1000 psig is outside the maximum allowable +/- 3 percent tolerance for the 1050 psig setpoint and therefore, at 0300 hours on June 1, 1996, after a review of post-trip data, ITS 3.7.1 Required Actions A.1 and A.2 were entered. The actions

were exited on June 2, 1996 after setpoint adjustments were made to MSV-34 and MSV-36.

Emergency Feedwater Initiation and Control [BA](EFIC) actuation occurred due to low OTSG level following the trip. The ICS and Main Feedwater [SJ](MFW) systems could not respond quickly enough to prevent the EFIC actuation.

Operators observed steam coming from the inboard seal on the motor-driven Emergency Feedwater (EFW) Pump [BA,P] EFP-1 while the pump was running. After packing was loosened by mechanics, as a conservative measure, at 2300 hours the Nuclear Shift Supervisor declared the pump inoperable per (ITS) 3.7.5, Required Action B.1. This action was exited at 1710 hours on June 1, 1996 after a post maintenance test was performed. Subsequent investigation indicated that elevated packing temperature is expected for high speed pumps such as EFP-1.

After operators secured Emergency Feedwater, the "B" Startup Feedwater (FW) Control Valve [SJ,FCV] FWV-39 would not control OTSG level at Low Level Limits. Operators corrected this condition by deenergizing the Rapid Feedwater Reduction (RFR) circuit in the ICS consistent with shutdown procedures.

The event was reported to the Nuclear Regulatory Commission (NRC) at 2302 hours on May 31, 1996 via the Emergency Notification System due to an RPS actuation per the requirements of 10CFR50.72(b)(2)(ii) and assigned Event Number 30571. This report is submitted in accordance with 10CFR50.73(a)(2)(iv).

EVENT EVALUATION

The Reactor Protection System [JC](RPS) performed as designed by fully inserting all Safety and Regulating Control Rods [AA,ROD]. Neutron flux decreased into the Source Range approximately seven minutes after the trip, as expected. The safety related Emergency Feedwater [BA] system operated as designed in providing cooling water to the OTSG's.

Steam flow through the Main Turbine [TA,TRB] is controlled by the Throttle, Governor, Reheat Stop, and Reheat Intercept valves [SB,V]. There are sixteen hydraulically operated Turbine Control valves in all. Turbine trips are accomplished by dumping hydraulic fluid from the valve operators back to the EHC system reservoir. When actuated, the Turbine Emergency Trip Solenoid Valve TB-247-SV dumps oil from all 16 valves. This function is inhibited by the valve lineup during on-line testing, a modification installed in 1994 in response to the turbine overspeed event at the Salem plant (see NRC Information Notice 91-83). After the trip, operations supervisory personnel accompanied the operator who was assigned to perform PT-325 and verified that the valve alignment was correct. Therefore, personnel error was eliminated as a possible cause of the event.

The maximum RCS pressure during the transient was 2350 psig. Following the trip, RCS pressure decreased to 1770 psig. The pressure was sufficient to trip the Low RCS Pressure bistables [PS] in all four Reactor Protection System [JC](RPS) channels as designed. Pressure stabilized at 2155 psig following the trip.

Due to the cooldown, Pressurizer level reached a minimum of 35 inches. In accordance with EOP guidance, Letdown Flow Control Valve [CB,FCV] MUV-51 was closed and High Pressure Injection Flow Control Valve [BQ,FCV] MUV-24 was opened in order to maintain control of the PZR level. The operator initially opened MUV-24 to 50 percent but then opened the valve to 100 percent after observing that level was continuing to decrease. Pressurizer level then began to recover. The operator closed MUV-24 approximately four minutes after it was opened.

Turbine Bypass Valve (MSV-11) remained closed after the trip. The response of MSV-11 did not cause any control problems since the necessary steam flow was within the capacity of the remaining operating turbine bypass valve for that steam line. Maintenance personnel corrected the response of MSV-11 by adjusting the positioner for the valve.

The continuous lifting and reseating of Main Steam Safety Valves MSV-34 and MSV-36 caused Reactor Coolant System pressure, temperature, and inventory to decrease more than is normal following a reactor trip. RCS average temperature had increased to 583 degrees Fahrenheit (F) when the Reactor tripped. Due to the continuous lifting and reseating of MSV-34 and MSV-36, RCS temperature decreased to 545 degrees F following the trip. Operators stabilized RCS temperature at 546 degrees F. Investigation revealed that due to the close proximity of the Atmospheric Dump Valves [SB,PCV](ADV) to MSV-34 and MSV-36, the valves lifted repeatedly due to thermally induced setpoint drift as well as pressure fluctuations at their inlet nozzles. The ADV's are used to control OTSG pressure and primary system temperature when the main condenser [SG,COND] is not available. Maintenance personnel successfully adjusted the lift setpoint of the two MSSV's, indicating no mechanical damage to the valves had occurred. Engineering personnel investigating the repeated lifting of these valves found no evidence to determine if the initial lift had been low and determined they lifted within their allowable tolerance.

The "A" Steam Generator reached Low Level Limits at 2126 hours. The "B" OTSG reached Low Level Limits four seconds later. At this point, Main feedwater response appeared to be normal. Steam Generators reached their Low Level alarm setpoints at 2127 hours. At this point, the Reactor Operator observed that the "B" Startup Feedwater control valve and the "A" Startup and Low load Feedwater control valves were being demanded open. The operator then observed that the Main Feedwater pumps [SJ,P] were not providing the required 80 pounds per square inch (psi) differential pressure across the Feedwater control valves. The operator took the Feedwater Pump control stations to manual and began increasing

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	05000302	96	017	01	05 OF 07

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

pump demand. Despite these actions, the EFIC system actuated due to low OTSG level at 2127 hours, four minutes and three seconds after the Reactor trip. Engineering determined that closure of the Throttle and Governor valves at the beginning of the transient increased OTSG pressure, resulting in reduced Feedwater flow before the Reactor tripped. Once the Startup Control Valves started opening, the differential pressure across the valves went high which caused the demand for the feedwater pumps to stay low. Due to these plant conditions, the non-safety related ICS and MFW systems could not respond quickly enough to prevent the EFIC actuation. It appears that the release setpoint for the Rapid Feedwater Reduction circuit which was added to ICS in 1987 to prevent overcooling after a reactor trip, may have contributed to the slow response of ICS in controlling the steam generators on low level limits. Steam Generator levels reached a minimum of 8 inches following the Reactor trip.

As noted in the Event Description section, elevated packing temperatures for pumps such as EFP-1 should be expected; therefore, there were no adverse consequences associated with the steam coming from the inboard seal.

Although this event raised questions concerning ICS control of Main Feedwater, no safety concerns were raised by this event. All protective safeguards systems performed as expected when challenged. There was no release of radioactive material, hence this event did not compromise the health and safety of the general public.

CAUSE

The cause of the unexpected isolation of steam flow path to the turbine was debris between the plug and the seat of a manual isolation valve [TG,ISV] preventing the valve from fully seating in the closed position thereby allowing high pressure EHC fluid to flow to the turbine trip drain header when TB-247-SV was stroked open per PT-325. Some debris could have entered into the EHC system when maintenance was performed on the turbine trip solenoid test blocks during the just completed Refuel 10 Outage. System post maintenance flushing of this and the other turbine trip line may not have removed particulate material that may have broken loose during outage activities. CR-3 had started up from a refueling outage on May 17, 1996. A contributing cause is the lack of sufficient instruction in the maintenance procedure to address post outage flushing of the main turbine trip line as the manual isolation valve and the turbine trip solenoid valve remain closed during the flushing activity.

IMMEDIATE CORRECTIVE ACTION

Plant conditions were stabilized in accordance with the immediate actions of EOP-2 "Vital System Status Verification," and EOP-10, "Post-Trip Stabilization".

ADDITIONAL CORRECTIVE ACTION

Note: Unless otherwise noted as being completed, the following actions are intended to be completed within six months of the issuance of this report.

1. Prior to startup from the subject reactor trip on June 2, 1996, the Plant Review Committee (PRC) reviewed the results of the analysis compiled in accordance with Administrative Instruction AI-704 "Reactor Trip Review and Analysis". The AI-704 package includes several actions that were completed, including a review of the main feedwater/ICS trip data for any operability concerns. In addition, the PRC reviewed the results of checks for seat leakage and EHC fluid flushes performed on all Turbine Trip Solenoid Valves, resolution of concerns regarding EFP-1 pump seal leakage and verification of proper lift setpoints for MSV-34 and 36.

2. A Request for Engineering Assistance (REA) has been initiated with reference to the repeated lifting of MSV-34 and MSV-36 to evaluate the impact of pressure disturbances affecting valve lift.

3. An evaluation will be conducted of the calibration method used for main steam turbine bypass valves (MSV-9 and MSV-10; MSV-11 and MSV-14), to provide better response between the pairs of valves.

ACTION TO PREVENT RECURRENCE

NOTE: The following actions are intended to be completed within six months of the issuance of this report.

1. A study will be conducted to address the long term resolution of the ICS/level control valves response to steam generator level after RFR initiation.

2. Maintenance Procedure MP-218C "Electro-Hydraulic System Flush" will be revised to enhance flushing of EHC Emergency trip and Overspeed trip fluid lines and attendant components.

3. Performance Test Procedure PT-325 "Turbine Generator Checks" will be reviewed to determine the need to insert intervening steps which may enable the operator to have greater assurance that manual isolation valves are properly seated. PT-325 will be revised as required based on this review.

PREVIOUS SIMILAR EVENTS

This appears to be the first event involving a reactor trip resulting from on-line turbine trip testing. There have been several previous RCS high pressure automatic reactor trips, a majority of which were caused by feedwater transients. The most recent automatic reactor trip was described in LER 93-09 which discusses a September, 1993 event involving a flux/flow/imbalance trip of 3 RPS channels. During the September, 1993 event, one of the main steam safety valves did not reseat due to a release nut that had rotated out of position.

ATTACHMENT

None