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

November 17, 1995
3F1195-06

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

Subject: Licensee Event Report (LER) 95-022-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 95-022-00. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73.

Sincerely,

B J Hickle

B. J. Hickle, Director
Nuclear Plant Operations

JAF:ff

Attachment

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

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PDR ADOCK 05000302
S PDR

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

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 8

TITLE (4) System Flow Balancing Identifies Low Flow to Components Resulting in Operation Outside Design Basis

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)
									N/A	0 5 0 0 0
1 0	1 8	9 5	9 5	— 0 2 2	— 0 0	1 1	1 7	9 5	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)		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	TELEPHONE NUMBER (AREA CODE)
J. A. Frijouf, Sr. Nuclear Regulatory Specialist	9 0 4 5 6 3 - 6 4 8 6

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 October 18, 1995, Florida Power Corporation's Crystal River Unit 3 was in MODE ONE (POWER OPERATION), operating at 100% RATED THERMAL POWER and generating 877 megawatts. During the performance of the "B" train of Decay Heat Closed Cycle Cooling System (DC) flow balance, two components were found to have measured flow less than the flow required in the Enhanced Design Basis Document. The motor cooler for the Decay Heat Service Raw Water Pump had a DC cooling water flow rate of 20 gallons per minute (gpm) when it was required to have a flow rate of 24 gpm. Decay Heat Removal Heat Exchanger 1B had a DC cooling water flow rate of 2925 gallons per minute when it was required to have a flow rate of 3000 gpm. When instrument error was considered, the DC cooling water flow rate could be less than an analyzed flow rate of 2918 gpm.

This event was determined to be a condition outside the design basis of the plant and was reported to the Nuclear Regulatory Commission in accordance with 10 CFR 50.72. The DC system was determined to be inoperable, and a 72 hour Action Statement was entered. The primary cause of this event was an inadequate procedure. Corrective actions include procedure revision and rebalancing the DC system.

EVENT DESCRIPTION

On October 18, 1995, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 100% REACTOR THERMAL POWER (RTP) and generating 877 megawatts. During the performance of a flow balancing procedure, FPC personnel discovered two components having less than required cooling water flow rates.

In a prior event, on June 20, 1995, FPC personnel discovered that cooling water flow to a Makeup Pump [CB,P](MUP 1-A) decreased when the cooling water source was switched from the normal Nuclear Services Closed Cycle Cooling System [KE](SW) to the alternate Decay Heat Closed Cycle Cooling System [BP](DC). This condition was evaluated as a design basis issue and was subsequently reported to the NRC as LER 95-010-00. As one of the corrective actions for this condition, FPC committed to flow balance both the SW and DC systems by the end of Refuel 10, April 30, 1996.

On October 18, 1995, during the performance of the "B" train of DC flow balance in accordance with Performance Testing Procedure PT-136B, "DC System Flow Balance and EGDG KW Loading", two components were found to have measured flow less than the flow required in the Enhanced Design Basis Document (EDBD) (see Figure 1, Decay Heat Closed Cycle Cooling System). The motor cooler [HX] for the Decay Heat Service Raw Water Pump [BS,P](RWP-3B) is required to have a flow rate of 24 gpm and was found to have a DC cooling water flow rate of 20 gallons per minute (gpm). Further, the Decay Heat Removal Heat Exchanger [BP,HX](DHHE-1B) is required to have a flow rate of 3000 gpm per the EDBD and the DC cooling water flow rate was found to be 2925 gpm. When flowmeter instrument error (+/- 2%) was considered, the DC cooling water flow rate could be less than an analyzed flow rate of 2918 gpm.

This event was determined to be a condition outside the design basis of the plant. The event was reported to the Nuclear Regulatory Commission at 1357 on October 18, 1995 via the Emergency Notification System (ENS) per the requirements of 10 CFR 50.72(b)(1)(ii)(B). The event was assigned the NRC Event Number 29470. This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B).

EVENT EVALUATION

Both RWP-3B and DHHE-1B are safety-related, and require DC cooling water flow to perform their safety function. The DC system removes decay heat released by the reactor core via the DH system (DHHE-1A and DHHE-1B)as well as heat from several essential thermal loads (pumps and motors, including RWP-3B). The system is designed to operate during normal shutdown operations and for design basis accidents, including a Loss of Coolant Accident (LOCA). Figure 1, "Decay Heat Closed Cycle Cooling System" illustrates the "B" train of the DC system. The DC system has two independent, 100% capacity trains. The redundant intermediate heat transport loops that form the DC system transfer decay heat from the contaminated

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

Reactor Coolant System (RCS) to the Ultimate Heat Sink (UHS) (Gulf of Mexico), via the Decay Heat Raw Water System. This prevents the release of radioactivity to the environment by acting as a barrier between the primary coolant and the UHS. During normal operation, the DC system removes Decay Heat (DH) released by the reactor core after a normal shutdown and maintains the RCS at or near a constant temperature during shutdown conditions. The DC system also provides cooling to various components.

An operability assessment was conducted in accordance with Compliance Procedure CP-150, "Identifying and Processing Operability Concerns." This procedure provides a structured, organized approach toward determining the OPERABILITY of plant components required for accident mitigation and safe shutdown of the plant and provides guidelines to ensure no loss of plant system or component safety function.

The "B" DC train was conservatively declared inoperable at the time the event occurred. This condition placed CR-3 outside Improved Technical Specification (ITS) Limiting Condition for Operation (LCO) 3.7.8, which requires two DC trains to be OPERABLE. Action "A" of this ITS requires the inoperable DC train be restored to OPERABLE within 72 hours, or be in MODE THREE within 6 hours and be in MODE FIVE within 36 hours. The 72 hour Action statement was entered at 1050 on October 18, 1995. The 1-hour non-emergency report to the NRC was made within one hour of the completing the "as-found" data portion of the procedure at 1310 on October 18, 1995 and after assessing this data against design basis requirements. The event was subsequently reported to the Nuclear Regulatory Commission at 1357.

In accordance with ITS, it was established that the CR-3 plant operation could continue at 100% RTP while in the 72 hour Action statement. As part of PT-136B, the DC flows to the DC cooled components were adjusted and "as-left" data obtained. These actions restored the flow to RWP-3B and DHHE-1B to ensure minimum design basis values including instrument accuracy, were achieved. The procedure was being worked while the evaluation was being conducted. The 72 hour action statement was exited at 0330 on October 20, 1995, when all the DC flow rates to all "B" train DC cooled components were within their acceptance criteria.

The Probabilistic Safety Assessment (PSA) has identified that this plant condition marginally increases the core melt frequency. Further, the Probabilistic Safety Assessment Monitor (PSAM) displays a "5.5 Yellow" indication relative to this condition. PSAM is a computerized system which provides a quantitative value associated with plant risk as well as a color indicative of the level of risk. A Yellow color indicates that the condition is risk significant and operators need to assure all adequate precautions are identified and the condition must be monitored. A Green Low Risk Significant PSAM level occurs below 4.7, a Red High Risk Significant level occurs above 47.

During the entire evolution, the redundant "A" train of DC was OPERABLE and available. The "A" train is capable of supplying 100% of required DC cooling capacity. Additionally, a preliminary review of the conditions outlined above indicated that even with the reduced DC flow to DHHE-1B given the current UHS temperature of approximately 72 degrees Fahrenheit (F), DHHE-1B would have performed adequately since the flows specified in the EDBD are for a design maximum temperature of 95 deg. F.. Therefore, based on the foregoing evaluation of plant conditions, this event did not compromise the health and safety of the general public. A similar evaluation is being undertaken to determine if RWP-3B would have been able to adequately perform its safety function during a design basis accident.

CAUSE

The primary cause of this event was an inadequate procedure. Relative to DHHE-1B, the procedure, PT-136 Rev 0, was last performed in December, 1987. The acceptance criteria states "NOTE: The flow to DHHE-1B is assumed to be 3200 gpm", but procedure step 7.4.2 states that the flow to DHHE-1B should be approximately 3000 gpm. The reason for this discrepancy is unknown.

Relative to RWP-3B, the procedure did not provide any flow margin for component flow rates to account for possible degradation in flow rate of the DC Pump [BP,P](DCP-1B) or instrument error.

IMMEDIATE CORRECTIVE ACTION

1. A Problem Report was issued, documenting the current condition.

2. An Operability Assessment was conducted in accordance with CP-150, "Identifying and Processing Operability Concerns."

ADDITIONAL CORRECTIVE ACTION

Previously identified DC system discrepancies were reported in LER 95-010-00, and several corrective actions were taken. These corrective actions also apply to the current discrepancies. Some of these corrective actions that are applicable include:

1. Procedure PT-136 was revised and divided into two procedures, one for the SW system (PT-136A) and one for the DC system (PT-136B). The procedure revisions addressed the causes stated above. The DC system procedure PT-136B was issued on October 16, 1995.

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2. New ultrasonic, non-intrusive flow measurement instrumentation was acquired. This instrumentation is capable of sensing flow in lines from 1/2 inch to 24 inches with greater accuracy then previously possible. This will aid in more accurately measuring system flow and assuring adequacy of margins of safety.

ACTION TO PREVENT RECURRENCE

1. Calculations will be performed to determine if the initial "as-found" flow rates for DHHE-1B and RWP-3B were adequate to cool the respective components during a design basis event at maximum UHS temperature. This task will be completed by January 15, 1996.

2. In order to provide defense in depth, standards will be reviewed for setting throttle valve positions in a consistent manner to ensure adequacy. This task will be completed by June 15, 1996.

3. Procedure PT-136B was performed to adjust DC system flow rates as necessary to increase cooling to DHHE-1B and RWP-3B.

4. Procedure PT-136B will be performed for the "A" train of the DC system to verify all cooling water flow rates are within acceptance criteria. This task will be completed prior to January 5, 1996. This schedule is acceptable because UHS temperatures are expected to be low enough to assure adequate margins to safety on the "A" train during this time interval.

PREVIOUS SIMILAR EVENTS

There have been three previous reportable events involving low flow. LER 89-009 reported decay heat pump continuous flow less than minimum required; LER 89-030 addressed the receipt and installation of an incorrect raw water pump impeller resulting in low flow; and LER 95-010 addressed low cooling water flow to a makeup pump.

ATTACHMENTS

Figure 1 - Decay Heat Closed Cycle Cooling System

Attachment 1 - Abbreviations, Definitions and Acronyms

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TEXT CONTINUATION

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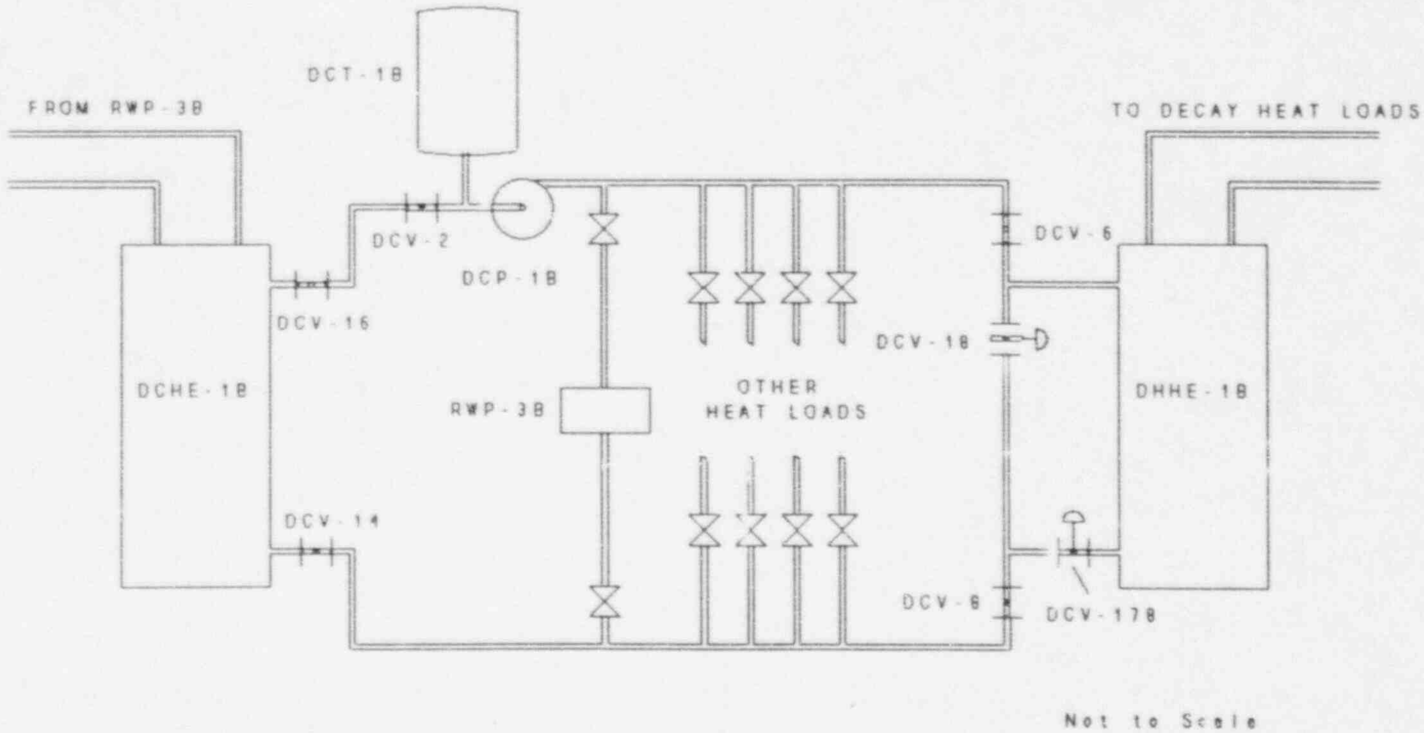
TEXT (If more space is required, use additional NRC Form 366A's) (17)

Figure 1



DECAY HEAT CLOSED CYCLE COOLING SYSTEM

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

CP-150	Procedure "Identifying and Processing Operability Concerns"
CR-3	Crystal River Unit 3
DC	Decay Heat Closed Cycle Cooling System
DHHE-1B	Decay Heat Removal Heat Exchanger 1-B
EDBD	Enhanced Design Basis Document
EGDG	Emergency Diesel Generator
ENS	Emergency Notification System
FPC	Florida Power Corporation
GPM	Gallons per Minute
ITS	Improved Technical Specifications
KW	Kilowatt
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
MODE ONE	Power Operation
MODE THREE	Hot Standby
MODE FIVE	Cold Shutdown
MUP	Makeup Pump
NRC	Nuclear Regulatory Commission
OPERABILITY	Capable of Performing its Design Function
PSA	Probabilistic Safety Assessment
PSAM	Probabilistic Safety Assessment Monitor

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PT-136	DC and SW System Flow Measurements and EGDG-1A KW Loading Due to ES Pumps (procedure)
PT-136A	SW System Flow Balance and EGDG KW Loading (procedure)
PT-136B	DC System Flow Balance and EGDG KW Loading (procedure)
RCS	Reactor Coolant System
REFUEL 10	Refueling Outage currently scheduled to begin February 29, 1996
RTP	RATED THERMAL POWER
RWP-3B	Decay Heat Service Raw Water Pump -3B
SW	Nuclear Services Closed Cycle Cooling System
UHS	Ultimate Heat Sink

NOTES: ITS defined terms appear capitalized in LER text (e.g. MODE ONE)

Defined terms/acronyms/abbreviations appear in parenthesis when first used (e.g. Reactor Building (RB)).

EIIS codes appear in square brackets (e.g. Makeup Tank [CB,TK])