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

November 3, 1995
3F1195-03

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

Subject: Licensee Event Report (LER) 95-021-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 95-021-00. 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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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)

Inadequate Pipe Supports Allow Stress On Motor Cooler Nozzle Resulting In Operation Outside The 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	0 4	9 5	9 5	0 2 1	0 0	1 1	0 3	9 5	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 9 7	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
J. A. Frijouf, Sr. Nuclear Regulatory Specialist	AREA CODE 9 0 4 5 6 3 - 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	

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 4, 1995, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 97% RATED THERMAL POWER (RTP) and generating 848 megawatts. As a result of an evaluation of certain pipe supports (or hangers), FPC engineering personnel determined that stresses induced on piping and connections to a motor cooler may have placed CR-3 in a condition outside its design basis. An operability assessment was conducted which concluded that the Building Spray Pump associated with the motor cooler was OPERABLE but degraded.

On October 4, 1995, the event was reported to the Nuclear Regulatory Commission per the requirements of 10 CFR 50.72(b)(1)(ii)(B) for operation outside the design basis of the plant. The primary cause of this event was personnel error by contract Architect and Engineering personnel responsible for the original plant analysis and design. Completed corrective actions includes completion of the evaluation of the current deficient motor cooler and the inspection of known similarly configured plant motor coolers. A review of other safety related motor coolers will be performed.

EVENT DESCRIPTION

On October 4, 1995, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 97% RATED THERMAL POWER (RTP) and generating 848 megawatts. As a result of an evaluation of certain pipe supports (or hangers), FPC engineering personnel determined that stresses induced on piping and connections to a motor cooler may have placed CR-3 in a condition outside its design basis.

In response to a Request for Engineering Assistance (REA) FPC engineering personnel were investigating two pipe supports [H], located on a Decay Heat Closed Cycle Cooling Water System [BI](DC) pipe, which appeared to be pried away from their mounting surfaces. The pipe supplies cooling water to the motor cooler [HX] for Reactor Building Spray Pump [BE,P](BSP) BSP-1B. The investigation revealed that the two supports (DCH-N-816 and DCH-N-802) were installed prior to 1985, and consequently do not meet the more stringent pipe support criteria currently required by plant procedures. The existing piping analyses, which includes the piping supported by these hangers, does not include support points at the location of these two hangers. The hangers were apparently installed as small bore pipe supports and later documented on the FPC Large Bore Piping Isometric (305) drawings and support drawings in accordance with the NRC IE Bulletin 79-14 "Seismic Analyses For As-Built Safety-Related Piping Systems" program.

FPC personnel were unable to retrieve any analysis or design data relative to these hangers. In an effort to determine the support load requirements a new piping analysis model was developed. The preliminary results of the analysis resulted in questions pertaining to the adequacy of the supports as well as the nozzles at the BSP-1 motor cooler, located adjacent to the hangers in question (a nozzle is the point of connection of the process piping to the vendor supplied component). Allowable nozzle loads at the motor cooler were not available so a modified piping analysis model was developed in an effort to ascertain the stress level seen by the motor cooler's header and tube sheet. Since specific information pertaining to the header and tubing materials and wall thicknesses were not available, several assumptions were made.

The preliminary analysis determined that the copper tubing at the header sees a worst case stress level, in excess of Code allowances, but less than the assumed yield stress of the tubing, thereby placing CR-3 outside its design basis. Although not yet formalized and issued, the Preliminary Analysis Verification has been completed. The findings of the Verification were that no substantial differences exist between it and the preliminary analysis.

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.

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CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	9 5	0 2 1	0 0	0 3 OF 0 7

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

At 1641, on October 4, 1995, the event was reported to the Nuclear Regulatory Commission (NRC) via the Emergency Notification System (ENS) per the requirements of 10 CFR 50.72(b)(1)(ii)(B) for operation outside the design basis of the plant and was assigned the Event Number 29419. This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B).

EVENT EVALUATION

The Reactor Building [NH](RB) Spray System has no normal duty function, but serves only in an Emergency Safeguards [JE](ES) capacity as part of the ES system. A high RB pressure signal of 30 pounds per square inch gauge (psig) from the ES system actuates RB spray operation. The two pumps (BSP-1A and BSP-1B) start and take suction from the Borated Water Storage Tank [BP,TK](BWST) through the Low Pressure Injection [BP](LPI) system piping. In the event of a major Loss of Coolant Accident (LOCA) the system sprays the RB atmosphere to remove the post accident energy and Iodine. Each train of Building Spray must be able to deliver 1200 gallons per minute (gpm) of flow into the RB within 60 seconds in order to be considered OPERABLE. The Improved Technical Specification (ITS) definition of OPERABLE includes a requirement that if cooling water is required for a component to perform its safety function then the cooling water source must be "capable of performing its related support function." Therefore, for BSP-1B to perform its maximum post accident mission time of 60 hours, a reliable source of cooling water must be available.

Each BSP (BSP-1A and BSP-1B) is powered by a 250 horsepower totally enclosed electric motor [MO] cooled by a water to air motor cooler (heat exchanger [HX]). The cooling water is supplied to the motor cooler by the Decay Heat Closed Cycle Cooling Water System [BI](DC). The motor cooler's function is to ensure the environment of the motor is acceptable for operation of the motor driving the BSP. It is therefore imperative that adequate cooling water pass through the motor cooler during pump operation. Failure of the piping could result in a reduction of water flow through the cooler. The postulated failure could be the result of local buckling of the supply or discharge tubing or the rupture of one or more tubes allowing a portion of the coolant to escape. Either failure could result in inadequate motor cooling and subsequent degradation of motor function.

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 a safe shutdown of the plant and provides guidelines to ensure no loss of plant system or component safety function.

Some of the factors considered in reaching a conclusion included the following:

1. The indicated stresses seen by the tubing are greater than those recommended by ANSE/ASME B31.1, "Power Piping." The B31.1 allowable value of 6,000 pounds per square inch (psi) is exceeded by the 15,000 psi actual value determined in the analysis. However, this 15,000 psi actual value is less than the 22,000 psi yield stress value associated with the material assumed in the analysis. This data indicates that the condition is unacceptable for long term operations but it is reasonable to assume this tubing will not experience any permanent deformation should an event resulting in full design loading occur prior to bringing the tubing/piping into full code compliance.

2. The potential for the worst case event necessary to create the stress concerns occurring during the operation of the BSP is extremely small.

3. A visual inspection of the condition of the tubing, header and piping by FPC engineering personnel did not indicate any structural distress. The tubing appears to be in the as-manufactured configuration. The tubing has been in service, with intermittent motor cooler operation, for the past seventeen years of plant operation.

4. Probabilistic Safety Assessment (PSA) has determined that the unavailability of BSP-1B will not increase the core melt frequency since Building Spray does not provide core cooling. Additionally, the CR-3 Probabilistic Safety Assessment Monitor (PSAM) displays a "green" indication relative to this condition, thereby indicating low risk level.

Based on analysis and inspection results as well as engineering judgement, the concerns related to this condition did not prevent BSP-1B from performing its safety function and FPC concluded that BSP-1B could be considered as OPERABLE but degraded. Therefore, this event did not compromise the health and safety of the general public.

CAUSE

The primary cause of this event was personnel error by contract Architect and Engineering personnel responsible for the original plant analysis and design. The qualification of nozzle loads was not consistently documented.

A secondary cause involves the original acceptance requirements versus those currently in effect. Since FPC personnel were unable to retrieve the original motor cooler design documentation, current, more rigorous requirements were applied to the equipment qualification efforts. The attachments to the motor coolers were not supported to meet these requirements.

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

Subsequent to the time this condition was discovered, additional investigation and analysis has been performed. The original analysis applied thermal loads that were overly conservative. It has now been concluded that the piping and supports are within code allowables. However, stress on the equipment nozzles is still outside of code allowable but well within yield stress. The analytical overstress condition on the equipment nozzles is due to the seismic load component.

The seismic adequacy of this equipment has been evaluated under the USI A-46 Program by Engineers qualified under the requirements of the Seismic Qualification User's Group (SQUG). Engineers performing this evaluation included both FPC Engineers and Consulting Engineers from Programmatic Solutions, Inc. The equipment nozzles of concern are part of the SQUG "rule of the box" for horizontal pumps (the rule of the box extends to the first support).

When SQUG criteria was applied, it was determined that the equipment is seismically adequate. Since seismic was the only load of concern, FPC now concludes that the equipment, as presently configured, is within established design basis.

ACTION TO PREVENT RECURRENCE

Other plant motor coolers known to have similar configurations (Makeup Pumps, Decay Heat Pumps and Building Spray Pumps) have been inspected. We have determined that these components are also in compliance with our design basis using the same SQUG methodology discussed above. Prior to closing out this issue, a review of other plant safety related motor coolers will be performed. This action is required to make certain that any other equipment of similar configuration is also evaluated. To accomplish this, it is necessary to gain access to the Reactor Containment Building. This action will be completed during the Refuel 10 outage scheduled for the spring of 1996.

PREVIOUS SIMILAR EVENTS

There have been four previous reportable events involving loose or damaged hangers. LER 85-044-00 addressed defective Control Complex Ventilation Duct hanger deficiencies, LERs 88-13-01 reported defective control air tubing supports, LER 86-003-00 addressed loose or damaged supports on Decay heat Pump [BP,P](DHP) 1-B and LER 92-24 reported hangers with a Factor of Safety Less than 2.0.

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

ANSE/ASME B31.1	Power Piping (US Standard Code for Pressure Piping)
BSP	Building Spray Pump
BWST	Borated Water Storage Tank
CMIS	Configuration Management Information System
C-150	Procedure "Identifying and Processing Operability Concerns"
CR-3	Crystal River Unit 3
DC	Decay Heat Closed Cycle Cooling Water System
ENS	Emergency Notification System
ES	Engineered Safeguards
FPC	Florida Power Corporation
GPM	Gallons per Minute
ITS	Improved Technical Specifications
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MODE ONE	POWER OPERATION (Greater Than 5 Percent Rated Thermal Power)
NRC	Nuclear Regulatory Commission
OPERABLE	Capable of Performing its Design Function
PSA	Probabilistic safety Assessment
PSAM	Probabilistic safety Assessment Monitor
PSI	Pounds Per Square Inch
REA	Request for Engineering Assistance

RTP RATED THERMAL POWER

IE 79-14 NRC IE Bulletin 79-14 "Seismic Analyses For As-Built Safety-Related Piping Systems"

305 Dwgs FPC Large Bore Piping Isometric Drawings

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])