Docket No. 50-302 August 7, 1989

Mr. W. S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Operations
Licensing
P. O. Box 219-NA-2I
Crystal River, Florida 32629

Dear Mr. Wilgus:

SUBJECT: CRYSTAL RIVER UNIT 3 - CORRECTION TO AMENDMENT NO. 117 (TAC NO. 54445)

On May 31, 1989, the Commission issued Amendment No. 117 for the Crystal River Unit 3 Nuclear Generating Plant. The amendment provided additional requirements for decay heat removal redundancy in the Crystal River Unit 3 Technical Specifications. The staff has subsequently been informed of administrative errors in the amendment.

Page IV of the index was previously changed in Amendment No. 115 by adding the toxic gas systems. However, when page IV was issued as an overleaf page in Amendment No. 117, this previously approved change was not included. Enclosed is the corrected page IV, as well as the overleaf page.

In addition, due to an administrative error, page B3/4 4-1 was not included in a number of reproduced copies that were distributed. Therefore, enclosed is page B3/4 4-1, as well as the overleaf page.

Sincerely,
Original signed by

Herbert N. Berkow FOR

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

[AMEND/CR-3 TAC 54445]

LA:PDII-2	PE:PDII-2	PM:PDII-2	D:PDII-2
DMiller	GWunder/jd	HSilver	HBerkow
08/07/89	08/04/89	08/04/89	08/7/89

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cc

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PDR ADOCK 05000302
P PNU

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Crystal River Unit No. 3 Nuclear
Generating Plant

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DATED: August 7, 1989

AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-72-CRYSTAL RIVER UNIT 3

Docket File
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D. Miller
H. Silver
OGC-WF
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B. Grimes, 9/A/2
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Wanda Jones, P-130A
J. Calvo, 11/F/23
J. Miller, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
M. Sinkule, R-II

cc: Plant Service list

OF01
1/1

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one loop, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE, ensuring that the DNBR will be maintained above 1.30 at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR equal to 22%, whichever is more restrictive.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two Reactor Coolant loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or DHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE.

The operation of one Reactor Coolant Pump or one DHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration changes in the Reactor Coolant System. The reactivity change rate assoicated with boron change will, therefore, be within the capability of operator recognition and control.

3/4.4.2 RELIEF VALVES - SHUTDOWN

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 317,973 lbs per hour of saturated steam at the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 RELIEF VALVES - OPERATING

The power operated relief valve (PORV) operates to relieve RCS pressure below the setting of the pressurizer code safety valves. This relief valve has a remotely operated block valve to provide a positive shutoff capability should the PORV become inoperable.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a pressurizer low level or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Engineered Safety Feature Actuation System as a result of a reactor trip. The high level limit is based on maximum reactor coolant inventory assumed in the safety analysis.

The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensures that the structual integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant