Florida Power
CORPORATION

August 31, 1979

File: 3-0-3-a-3

POOR ORIGINAL

Mr. D. F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Identification and Resolution of Long-Term Generic Issues Related to the Commission Orders of May 1979

Dear Mr. Ross:

On August 23, 1979, Florida Power Corporation received your letter of August 21, 1979, identifying eight long-term issues related to the Order which must be resolved for Crystal River Unit 3 and the other B&W Operating Plants.

These eight (8) items were identified and briefly discussed in Enclosure 1 of your letter. In your discussion of Items 1, 4, [illegible] 7, and 8, you requested Florida Power Corporation to provide additional information and our schedule for resolution of these five (5) items by August [illegible].

In that regard, Florida Power Corporation hereby submits, as Attachment 1 to this letter, our response to your August 21, [illegible] request for additional information.

If you require further discussion concerning our response, please contact us.

Very truly yours,

FLORIDA POWER CORPORATION

G. C. Moore
Assistant Vice President
Power Production

GCMekcF06(D5)

Attachment

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ATTACHMENT 1

Response to Ross Letter of August 21, 1979

Item 1 - Failure Mode and Affects Analysis of the Integrated Control System

On August 17, 1979, B&W submitted to you for your review, copies of the report entitled "BAW--1564, Integrated Control System (ICS) Reliability Analysis". This letter is to advise you that this report is applicable to Crystal River Unit 3. Although this was a generic report developed by B&W, and there are differences in the secondary system designs at the various B&W plants, we feel that the conclusions reached in this report can be applied to Crystal River Unit 3. Florida Power Corporation is presently reviewing the recommendations listed in Section 3 of this report to determine what possible changes are necessary at Crystal River Unit 3 to enhance reliability and safety.

Item 4 - Auxiliary/Emergency Feedwater System Reliability Upgrade

This letter is to inform you of Florida Power Corporation's commitment to the AFW/EFW System Reliability Study proposed by B&W and discussed with you and your staff on July 19, 1979, and August 9, 1979. The draft report for Crystal River Unit 3 will be submitted by October 22, 1979, and the first report will be submitted by December 3, 1979.

Item 5 - Detailed Analysis of the Thermal-Mechanical Conditions in the Reactor Vessel During Recovery from Small Breaks With Extended Loss of All Feedwater

The above analysis will be submitted by December 21, 1979.

Item 7 - Small Break LOCA Analysis

The following is our schedule of response to the six (6) items contained in Attachment A of your letter:

1) A. Report will be submitted on December 1, 1979.
   B. Report will be submitted on September [illegible], 1979.

2) A. Report will be submitted on September [illegible], 1979.
   B. In response to this request, we are proposing three (3) options in preference of order:

   1) Provide a statement by September [illegible] 1979, that no small break with auxiliary feedwater will pressurize the system to the PORV setpoint.

   2) Provide by December 30, 1979, a qualitative assessment of the transient.

   3) Provide core analysis by February 1, 1980, using 0.01 $ft^2$ break with no AFW available.

   We are presently proceeding with option #1, unless otherwise notified by the NRC by September 7, 1979.

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## Table 4-5. (Cont'd)

| MODULE NO. | MODULE NAME | FAILURE MODE | EFFECT ON NSS | REACTOR TRIP | REMARKS |
|---|---|---|---|---|---|
| Functional: 2<br>ICS: 4-2-13 | Modified Turbine Header Pressure Error | High | The ICS pulser will send a continuous increase demand to the turbine EHC causing a throttle pressure decrease. The large pressure error detector transfers the turbine EHC to manual in ~5 seconds. The ICS assumes the tracking mode and the feedwater and reactor increase to meet the ~4% load increase. The erroneous modified throttle pressure error causes a mismatch between the NSS steam production and the turbine operation. The pressure decrease is limited at ~100 psi by the turbine initial pressure regulator. Reactor trip on high RC pressure is possible. | High RC Pressure | -No problem after reactor trip |
| | | Low | Essentially the same response as Failure Mode "High" except pressure rises and is terminated by turbine by-pass valve action. | High RC Pressure if power >~40%. | -No problem after reactor trip |
| Functional: 3<br>ICS: 3-6-1 | Turbine Control | | Failure is very similar to failure of functional block 2, above. | | |