Florida Power
CORPORATION

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Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
NUREG-0737, Item II.D.1
Pressurizer Safety/Relief Valve Operability

Dear Mr. Eisenhut:

Pursuant to NUREG-0737, Item II.D.1 (as revised by your letter of September 29, 1981), Florida Power Corporation (FPC) hereby submits our assessment of the EPRI PWR Safety and Relief Valve Test Program and a discussion of efforts which have been or are being taken with respect to the results of that assessment as it relates to Crystal River Unit 3 (CR-3).

As indicated in our submittal to you of March 31, 1982, the following documents have been transmitted to you by Mr. David Hoffman of Consumers Power Company on behalf of the participating PWR utilities and are incorporated by reference herein as part of this submittal:

- o Valve Selection/Justification Report
- o Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves for B&W 177-FA and 205-FA Plants
- o EPRI Test Condition Justification Report
- o Safety and Relief Valve Test Report

Based on the above, Items II.D.1.A(1), II.D.1.A(2), II.D.1.A(3), and II.D.1.B are addressed as follows:

1. Item II.D.1.A(1) - The Safety and Relief Valve Test Report provides data which shows the abililty of the Pressurizer Safety and Relief valves to open and close under all non-ATWS transient conditions. Specifically, the report documents the successful performance of the CR-3 as-installed relief valves, and, in our

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engineering judgement, the successful performance of the as-installed safety valves. Detailed justification of our engineering judgement will be presented following completion of evaluations to be performed pursuant to Items II.D.1.A(2) and II.D.1.A(3).

2. Item II.D.1.A.(2) - The applicability of valve tests performed by EPRI at the C-E, Marshall Steam Electric Station, and Wyle Norco facilities are direct in nature based on the following:

   a) A valve directly applicable to the CR-3 safety valves (i.e., Dresser Model 31739A) was tested at the C-E facilities.

   b) A valve directly applicable to the CR-3 power-operated relief valve (PORV) (i.e., Dresser Model 31533VX-30) was tested at both the Marshall and Wyle facilities.

   c) Inlet piping configurations at all three (3) facilities model the CR-3 short inlet (i.e., pressurizer nozzle) configuration.

   d) Inlet fluid test conditions for the safety and relief valves, as documented in the "EPRI Test Condition Justification Report," are either comparable or more conservative than those conditions presented in the "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves for B&W 177-FA and 205-FA Plants."

   e) Discharge piping configuration effects on relief valve performance have been shown to be minimal. Effects observed on safety valves with regard to CR-3 specific backpressure effects are presently being evaluated by an independent consultant to determine what, if any, piping changes may be necessary to ensure that backpressure effects do not affect valve performance. Preliminary indications are that no changes are anticipated due to the fact that the only common point of discharge piping for each safety and relief valve occurs in the Reactor Coolant Drain Tank volume.

3. Item II.D.1.A(3) - Results of testing of the power-operated relief valves indicate, as documented by the Marshall and Wyle test reports, that the performance of the CR-3 PORV (Dresser Model 31533VX-30) is within criteria established for successful performance.

   Results of safety valve performance have been shown by the C-E testing to be dependent on ring settings. In that ring settings affect accumulation, lift, blowdown, and general stability, each will be addressed as follows:

   a) Safety Valve ring settings will be evaluated following completion of negotiations with appropriate parties to perform this effort. However, based on the past performance of the CR-3 safety valves under transient water conditions with no evidence of instability or subsequent damage, it is felt that as-installed ring settings provide justification for the continued operation of CR-3 with current ring settings. Our judgement indicates that any changes to ring settings which are recommended will only serve to further optimize already proven valve performance.

b) Blowdown through the safety valves in excess of ASME Code specified values is being evaluated by the NSSS vendor (Babcock & Wilcox). Preliminary assessment by B&W indicates that blowdown in excess of that seen at the C-E test facility can be safely and reasonably accommodated by the CR-3 NSS.

Discharge piping and supports have been analyzed pursuant to the February 26, 1980, transient. Results of this previous evaluation, as submitted by letter (P. Y. Baynard to R. W. Reid) dated July 17, 1980, showed piping and support acceptability under transient water conditions; therefore, no further evaluation is deemed necessary for CR-3.

4. Item II.D.1.B - Block Valve Qualification was previously addressed by the PWR Owners as an issue not important to safety. This position is based on the successful response of the Reactor Coolant System to an unisolable leak as a result of a stuck open PORV (i.e., small break LOCA) and the low probability of this event (based on relief valve operability testing results). However, testing done at the Marshall Steam Electric Station on representative block valves has been documented and transmitted by Mr. R. C. Youngdahl of Consumers Power Company to Mr. Harold Denton, NRC, on June 1, 1982. Though not specifically warranted, FPC hereby incorporates this transmittal into our present submittal to close the aforementioned NUREG-0737 sub-item.

Based on the above, it is concluded that the EPRI test program has demonstrated with reasonable assurance that Safety and Relief Valves will operate as intended under all postulated transient events. With regard to the evaluations which FPC now has under way relative to ring settings, safety valve back pressure, and safety valve blowdown, it is anticipated that the results or definitive schedules for resolution will be provided by September 15, 1982.

If you require any further information, please contact Mr. D. G. Mardis of my staff at (813) 866-4283.

Very truly yours,

Patsy Y. Baynard

Dr. P. Y. Baynard
Assistant to Vice President
Nuclear Operations

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