August 26, 1985

Docket No. 50-302

DISTRIBUTION

| | |
|---|---|
| Docket File | RIngram |
| NRC PDR | HSilver |
| L PDR | BMozafari |
| ORB#4 Rdg | Gray File |
| HThompson | EBrach |
| OELD | HOrnstein |
| EJordan | WPaulson |
| BGrimes | GEdison |
| JPartlow | REmch |
| ACRS-10 | |

Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
P. O. Box 14042; M.A.C. H-3
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

We have been reviewing the information in your letters of July 17, 1980 and June 30, September 17, and November 1, 1982 with respect to testing of relief and safety valves (NUREG-0737, Item II.D.1) for Crystal River Unit 3. In order to permit us to complete our review, responses to the enclosed request for additional information are required.

Please provide the requested additional information within 45 days of receipt of this letter.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
As Stated

cc w/enclosure:
See next page

| | |
|---|---|
| ORB#4:DL | ORB#4:DL |
| HSilver;cr | JStolz |
| 8/20/85 | 8/22/85 |

REQUEST FOR ADDITIONAL INFORMATION

TMI ACTION NUREG-0737 (II.D.1)

FOR

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

OCTOBER 1984

SAFETY EVALUATION QUESTIONS

TMI ACTION NUREG-0737 II.D.1

FOR

CRYSTAL RIVER UNIT 3

Question Related To Selection Of Transients And Inlet Fluid Conditions:

1. The submittal does not explain how fluid transient cases other than the February 26, 1980 transient that occurred at Crystal River Unit 3 (CR-3) were considered in qualifications of the safety valve/PORV system. Since the February 1980 transient may not be the limiting transient, consideration of these other transients is important. The B&W Valve Inlet Fluid Conditions Report indicates that fluid conditions for overpressure transients in the CR-3 plant have been identified. Verify that the fluid conditions identified in the B&W report were determined through analyses of accidents and operational occurrences referenced in Regulatory Guide 1.70, Revision 2, as required by NUREG-0737 II.D.1. Specify the limiting fluid conditions expected for steam and liquid flow through the safety valves and PORV.

Questions Related To Valve Operability:

2. The EPRI report concerning examination and tests of CR-3 PORV and safety valves discusses performance of these valves during and after the February 26, 1980 transient. This report provides results for the case of saturated steam flow through the PORV and water flow through a safety valve, but this transient may not envelope all limiting transients identified under requirements of NUREG-0737 II.D.1. Thus, results from the EPRI test program are needed to complete qualification of the PORV and safety valves. The submittal does refer to several generic documents concerning this program and claims that these reports document successful performance of the CR-3 PORV and safety valves. To justify this evaluation, show that the fluid inlet conditions determined for limiting transients were enveloped in the EPRI tests. Show that the test results verify that the safety valves and PORV will open and close under expected flow conditions.

Along with other flow conditions, identify the expected total length of time of liquid flow through the safety valves for a transient such as that of February 1980 or other liquid flow transients (see Question 1). Provide assurance that the valves functionability will not be impaired by liquid flow of this duration. Demonstrate the PORV will operate properly over the range of fluid conditions expected for cold overpressurization events.

3. Crystal River 3 uses the Dresser Model 31739A safety valve, which was tested in the EPRI test program. According to results reported in the EPRI Safety and Relief Valve Test Report, this valve passed its rated steam flow on certain tests but not on others, depending on the ring settings and backpressures. The specific ring settings to be used at CR-3 and the expected plant backpressures were not identified in the submittal. The letter dated November 1, 1982, indicated that the safety valves were replaced with valves having ring settings consistent with the EPRI tests. It also stated that a plant specific evaluation of the discharge piping backpressure was in-process. This letter does not, however, identify the final ring settings or backpressure. To establish which EPRI tests are applicable to CR-3 and thereby demonstrate that the safety valves will pass rated flow, provide the final plant ring settings and expected backpressure.

4. In a letter dated March, 1976, from Dresser Industries (the manufacturer of the Crystal River PORV) to Metropolitan Edison Company, Dresser cautions that the PORV block valve should be kept closed when reactor coolant system pressure is below 1000 psig to avoid damaging the PORV disk and seat by steam wirecutting. Results from the EPRI test program indicate that this valve was successfully tested on water at pressures in the 500-900 psig range but was not tested on low pressure steam. Additionally, each test sequence was initiated with a valve where the disk and seat were in excellent condition, which may not be representative of the CR-3 PORV when placed in service. Thus, the available EPRI test data is evidently insufficient to demonstrate compliance with NUREG-0737 Item II.D.1. The Dresser recommendation indicates that precautions may be necessary

to avoid damage to the PORV disk and seat at pressures below 1000 psig. Following the Dresser recommendation to isolate this valve at these lower pressures would, however, seem to preclude the use of the PORV for cold overpressure protection. Explain what precautions will be taken at pressures less than 1000 psig to prevent damage to the PORV disk and seat or provide results from EPRI or other tests performed since March 1976 that demonstrate that precautions described in the March 1976 letter are not required to avoid such damage.

5. To verify that the EPRI tests adequately demonstrated stable operation of the plant safety valves, the EPRI Test Condition Justification Report indicated that the inlet pressure drop of the test piping must be at least as great as that for the plant. Provide a comparison between the inlet pressure drop for the tests and the expected pressure drop for the plant.

6. Bending moments are induced on the flanges of the safety valves and PORV during the time they are required to operate because of thermal expansion of the pressurizer tank and the piping and because of the fluid discharge loads. Provide assurance that these bending moments will not adversely affect operability of the valves.

7. To meet the block valve qualification requirement contained in Paragraph II.D.1.B of NUREG-0737, the submittal refers to a transmittal from R. C. Youngdahl of Consumers Power Company to Harold Denton, NRC, on June 1, 1982 concerning a block valve testing program conducted at the Marshall Steam Electric Station. This test program, however, did not include tests on the particular block valve used at CR-3. Additionally, these tests were limited to steam flow conditions, whereas NUREG-0737 requires demonstration of valve functionability for all fluid conditions for which the valve is required to operate. Provide a justification as to how results of the Marshall tests or other tests can be used to demonstrate operability of the CR-3 block valves for the required conditions. Account for differences between the CR-3 valve (and operator) and the test valve (and operator).

8. NUREG-0737, Item II.D.1 requires that plant-specific PORV control circuitry be qualified for design-basis transients and accidents. Please provide information which demonstrates that this requirement has been fulfilled.

Questions Related To Thermal Hydraulic Analysis:

9. To meet the NUREG-0737 requirement that the piping and supports associated with the safety and relief valves be qualified as well as the valves themselves, the submittal refers to an analysis that was performed on the February 26, 1980 transient. Relying on this analysis alone to qualify the piping and supports requires justification that the fluid conditions corresponding to this transient will produce maximum dynamic loading on the system. This is somewhat questionable since the safety valve discharged at a pressure of 2425 psig, while the valve set pressure is 2500 psig. Also, the transient involved only the discharge of one safety valve, whereas an actuation of both safety valves may produce worse loading. Provide a comparison between the peak pressures and pressurization rates with those expected for other limiting transients and discuss the effects that an opening of two safety valves rather than one would have on loading of the system. Justify, if possible, that the transient analyzed produces maximum loading on the piping system.

10. Further information is needed to evaluate the thermal hydraulic analysis. The RELAP4/MOD5 computer program was used to perform the thermal hydraulic analysis. Explain whether parametric studies or other verification studies were performed to assure that this program would generate accurate fluid forces for the transient analyzed. Also justify the use of a safety valve opening time of 0.04 seconds. The opening times observed in the EPRI tests on this valve for water discharges were typically significantly shorter than 0.04 seconds.

11. The THRUST computer code was used to generate fluid force histories from RELAP4/MOD5 output. Provide a detailed description of the methods used in this program and explain how the program has been verified for the type of transient analyzed.

12. The submittal does not provide some of the important details of the thermal hydraulic analysis on the PORV and associated piping. Identify the fluid conditions assumed including pressure, pressurization rate, temperature, flow rate and fluid type. Assure that the fluid conditions for cold overpressurization events were encompassed in the analysis.

Questions Related To Structural Analysis:

13. Further information is needed to evaluate the structural analysis. The submittal states that the PIPDYNII computer program was used to perform the structural analysis. Provide a description of the methods used in this program and explain how the program has been verified for the type of transient analyzed.

14. To adequately demonstrate structural integrity of the system, the loads due to fluid discharge transients must be combined with other loads such as seismic and operating. Identify the load combinations performed in the analysis together with the allowable stress limits. Explain the mathematical methods used to perform the load combinations, and identify all governing codes and standards used to determine piping and support adequacy. The submittal does mention that the ANSI B31.1 Code (year not given) was used to evaluate adequacy of the nozzle connection to the pressurizer.

15. The submittal states that incident transient loads were calculated only for the first four piping sections downstream of the pressurizer, claiming that these are the major contributors to the loads on the pressurizer nozzle. A problem with using this analysis to meet requirements of NUREG-0737 II.D.1 is that the NUREG requires a qualification of the safety valves and PORV and the associated piping

and supports. It must be verified that the fluid loads will not jeopardize operability of the safety valves or PORV, or structural integrity of the valve inlet piping as well as the pressurizer nozzles. Thus, provide a justification that an analysis has been performed where enough of the fluid transient loads, discharge piping, and pipe supports were included in the analysis to assure unimpaired operability of the valves and structural integrity of the valve inlet piping and pressurizer nozzles and to verify that pipe deformations will not block fluid flow anywhere in the system.

16. A letter included in the submittal dated July 3, 1980 discusses an analysis of the supports attached to the piping from the pressurizer to the reactor coolant drain tank. This letter suggests that at least two of these supports should be replaced to sustain loads that include steam and water hammer loadings. Provide a final evaluation of stresses in the piping and supports and identify any required modifications to the piping or supports.

17. One of the safety valves discharged water during the February 26, 1980 transient and the safety valves are expected to discharge water for extended operation of HPI events. When the Dresser safety valves discharged water in the EPRI tests, the valves typically fluttered through partial lift positions at high frequencies. These valve oscillations cause high frequency pressure oscillations in the valve inlet piping, which could potentially excite high frequency vibration modes in the piping. This excitation creates bending moments in the inlet piping that should be combined with moments from other mechanical loads. Provide one of the following: (1) a justification that these high frequency pressure oscillations will not occur or (2) a comparison between allowable bending moments with the bending moments induced in the plant piping by the dynamic motion and other mechanical loads.

Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

cc:
Mr. R. W. Neiser
Senior Vice President
and General Counsel
Florida Power Corporation
P. O. Box 14042
St Petersburg, Florida 33733

Nuclear Plant Manager
Florida Power Corporation
P. O. Box 219
Crystal River, Florida 32629

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Resident Inspector
U.S. Nuclear Regulatory Commission
Route #3, Box 717
Crystal River, Florida 32629

Regional Administrator, Revion II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. Ulray Clark, Administrator
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, Florida 32301

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Inverness, Florida 36250