Florida Power
CORPORATION

April 2, 1986
3F0486-02

Mr. Harold R. Denton, Director
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
Snubber Optimization Approval Request

Dear Sir:

Attachment 1 to this letter references thirteen documents, transmittals, and meetings since the Fall of 1984 related to the Florida Power Corporation (FPC) plans to optimize the reactor coolant (RC) snubber arrangement for Crystal River Unit 3 (CR-3). These plans are based on the leak-before-break (LBB) concept as applied to the primary system of pressurized water reactors which is soon to be formalized as a revision to General Design Criterion 4 (GDC-4).

At the February 27, 1986 meeting (Reference 13), ten questions raised by the NRC staff reviewers were discussed among representatives of NRC, FPC, Babcock and Wilcox (B&W), Brookhaven National Laboratory, and Gilbert Commonwealth. These ten NRC questions and the elements of response to each question as discussed in the February 27, 1986 meeting are shown in Attachment 2. All but questions 1, 2, 3, and 5 were considered resolved at the meeting. Subsequent review by the NRC of a Franklin Institute report resolved questions 1 and 2. Question 3 was resolved after an NRC review which indicated that the approach used by B&W appeared to be the same as used by other nuclear steam supply system manufacturers. An additional study/analysis was performed by B&W to provide supplementary information on Question 5. Results of this study and analysis are included as Attachment 3. On March 19, 1986, the NRC requested additional information on questions 9 and 10 which is included as Attachment 4.

We believe the information provided the NRC since the Fall of 1984 and further supplemented by this transmittal should constitute a sufficient technical basis for NRC's completion and issuance of the Safety Evaluation Report. As of April 1, 1986, installation of the CR-3 optimized snubber/link bar configuration has been completed for RC pumps A and B and is underway for RC pumps C and D.

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With this transmittal, FPC has completed its response to all outstanding questions from the NRC related to the pump support configuration planned for use at CR-3. We request NRC endorsement of these plans by April 30, 1986 to provide an orderly startup for CR-3.

Sincerely,

G. R. Westafer

G. R. Westafer
Manager, Nuclear Operations
Licensing and Fuel Management

EHD/feb

Attachments

ATTACHMENT 1

LIST OF REFERENCES

ATTACHMENT 1

List of References

1. B&W Report, BAW 1847, dated October 1984, subject the B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS.

2. FPC letter to NRC, Westafer to Denton, dated February 1, 1985 (3F0285-02), subject Request for Exemption from a Portion of 10 CFR 50, Appendix A, General Design Criterion 4 (GDC-4).

3. Meeting on August 5, 1985 among NRC, Babcock and Wilcox, and FPC representatives to present FPC plans and define NRC staff needs for information to support the FPC request of Reference 2 above.

4. FPC letter to NRC, Westafer to Denton, dated August 30, 1985 (3F0885-24), subject Re-evaluation of CR-3 Reactor Cooling System Loads Utilizing Leak-Before-Break Concept to Remove Reactor Coolant System Main Loop Pipe Break Protective Devices; transmitted B&W Report prepared for FPC, subject Evaluation of Reactor Coolant System Loads and Component Support Margins Resulting from Optimized Reactor Coolant Pump Support Configuration.

5. FPC letter to NRC, Simpson to Denton dated September 27, 1985 (3F0985-26), subject Transmittal of Report Related to Request for Exemption from a Portion of 10 CFR 50, Appendix A, General Design Criterion 4 (GDC-4); transmitted B&W Report, Document ID 51-1159048-00, prepared for FPC, subject Safety Balance Assessment for Elimination of Reactor Coolant System Main Loop Pipe Break Protective Devices.

6. B&W Report, BAW 1847, Rev. 1, dated October 7, 1985, same subject as in Reference 1; revised report issued to address comments and questions raised by NRC staff reviewers.

7. FPC letter to NRC, Westafer to Denton, dated October 29, 1985 (3F1085-13), subject Transmittal of Report Related to Request for Exemption from a Portion of 10 CFR 50, Appendix A, General Design Criterion 4; transmitted report prepared by FPC, subject Assessment of CR-3 RC Leak Detection System, File: SP 83-133, dated October 25, 1985.

8. Meeting on October 31, 1985 among NRC, B&W, and FPC representatives to discuss partial exemption from GDC-4.

9. NRC summary issued by H. Silver dated November 13, 1985 of the reference (8) meeting.

10. FPC letter to NRC, Westafer to Denton, dated January 13, 1986 (3F0186-12), subject Additional Information Regarding Request for Partial Exemption from General Design Criterion 4.

11. FPC letter to NRC, Westafer to Denton, dated January 16, 1986 (3F0186-18), subject Technical Specification Change Request No. 142; proposes to remove the tabular list of snubbers from the Technical Specifications in accordance with NRC guidance provided in Generic Letter 84-13.

12. FPC letter to NRC, Simpson to Denton, dated January 21, 1986, subject Snubber Optimization Approval Request.

13. Meeting on February 27, 1986 among NRC, FPC, B&W, and Gilbert Commonwealth, Lynchburg, Va., to discuss ten questions raised by NRC.

ATTACHMENT 2

Meeting Among NRC, FPC, B&W and Gilbert Commonwealth

Lynchburg, Virginia

ELEMENTS OF RESPONSES TO NRC QUESTIONS OF 2/21/86

February 27, 1986

NRC QUESTION 1

The reactor coolant system analysis is based on a one-half system structural model with appropriate boundary conditions. B&W has stated that this approach has been verified by comparing satisfactorily the results from a half-system model to a model containing both loops. However, a diagram of the full system indicates that the symmetry is only approximate. The concerns is whether the half-system model of the reactor coolant system and its boundary conditions provides correct results when compared to the full system analysis.

ELEMENTS OF RESPONSE TO QUESTION 1

o B&W Report 32-1103808 considered the CR-3 RCS piping arrangement.

o Conclusion - A half loop model can be used to determine the dynamic response of the full system.

o CR-3 RC pumps have supports with:

  - Differing Orientations
  - Spring Rates
  - Lengths

o Differences have been modelled.

o Both the North and South half loops have been analyzed.

NRC QUESTION 2

The analysis of the RCS is based on a model which combines the reactor building interior concrete and the reactor coolant system. The concern is whether the properties of the reactor building have been properly determined and are properly reflected in the combined model.

ELEMENTS OF RESPONSE TO QUESTION 2

- O Reactor building interior concrete model properties used by B&W have been reviewed recently by Gilbert Commonwealth.

- O A 10.6% weight difference was identified.

- O Weight difference results in approximately a 7% change in seismic stress at the highest stressed location.

- O Results in an approximate 1/2% increase in the total design stress at the highest stressed location.

NRC QUESTION 3

The analysis of the RCS subjected to dead weight, and other distributed loading, is performed by imposing concentrated forces at the mass joints. The equivalent moments (known as fixed-end moments) appear to be neglected in this approach. No justification is provided for this approach.

ELEMENTS OF RESPONSE TO QUESTIONS 3

- O RCS structural model includes local deflection of the RV and OTSG nozzles.

- O End moments are accounted for by modelling the nozzle flexibility.

- O Lumped masses at discrete locations along the piping length produce conservative moments.

NRC QUESTION 4

The damping value for the RCS components was taken as 2% of critical, per RG 1.61. However, for the pumps, a total of 32 large bore snubbers were replaced by a total of four smaller snubbers and four struts, thus reducing significantly the physical sources of damping in the structure. Justification is needed for not choosing a lower damping value for the optimized configuration than that specified in RG 1.61.

ELEMENTS OF RESPONSE TO QUESTION 4

O Damping of the RCS piping system has been reduced by support removal.

O RC pump and motor assembly is a complex arrangement with:

- Rotating Elements
- Various Attached Piping Systems
- Gaped Interfaces
- Bolted Connections
- Complicated Structural Members

O 2 percent damping is considered appropriate for the pump, motor stand and motor assembly per RG 1.61.

NRC QUESTION 5

An evaluation is needed on the structural stability of the optimized RC pump support as-built configurations when subjected to compressive loading. Uncer certain types of such loading, the proposed configurations appear to be structurally unstable indicating that the supports may not be effective under actual loading, thus causing the allowable stresses to be exceeded.

ELEMENTS OF RESPONSE TO QUESTION 5

- The RCS structural analysis has pump support elements modelled as pinned bars.
- No moments or perpendicular loadings are transferred by these elements.
- Support members are positioned so they complement existing support already provided by the piping.
- The RCS structural model is a 3 dimensional model with a 3 dimensional loading.
- Masses are input at centerline locations representative of the actual structure.
- Earthquake inputs are calculated for 3 directions.
- The piping primarily supports the pumps; constant support hangars provide additional support for dead weight.
- Allowable stresses are not exceeded.

NRC QUESTION 6

Clarification is needed on the method of determining the composite damping values and the seismic response spectra for structural models with different damping through the structure.

ELEMENTS OF RESPONSE TO QUESTION 6

- O Composite damping values are calculated based on a mass and mode shape weighted technique.

- O Method referred to as strain energy weighted.

- O Input seismic response spectra is from the FPC Environmental and Seismic Qualification Guide Specifications and Data SP-5095.

NRC QUESTION 7

Clarification and justification are needed for the generation of different flexibility matrices for the seismic analyses of different earthquake components.

ELEMENTS OF RESPONSE TO QUESTION 7

- Two flexibility matrices are generated for seismic because of half loop model.

- Boundary conditions are different in the half loop models.

- For X and Y direction earthquakes:

  - RV and wall centerlines are fixed for Z translation and rotation of about the X and Y axes.

- For Z direction earthquake:

  - RV and wall centerlines are fixed for X and Y translation and Z rotation.

NRC QUESTION 8

Clarification is needed on the boundary conditions used for the thermal analysis of the half-system model.

ELEMENTS OF RESPONSE TO QUESTION 8

o Boundary conditions maintain vertical RV centerline

o RV and wall centerlines are:

- Fixed for Z direction translation
- Fixed for X direction rotation

o RV rotates and transmits loadings to base.

NRC QUESTION 9

Clarification is needed that steady-state hydraulic loads include flow induced vibration due to pump operation (NUREG/CR-1319, "Cold Leg Integrity Evaluation"), and that these loads are appropriately combined with dead weight and thermal loads.

ELEMENTS OF RESPONSE 9

o Stress is 1.7% of operating primary stress allowable.

o Stress amplitude is well below the endurance limit.

o Stresses are negligible in the ASME Code fatigue analysis.

  - Usage factor would be increased by less than 1%

NRC QUESTION 10

Clarification is needed that local stresses in the RC pump casing resulting from the attachments of the supports to the casing have been included in the stress evaluation of the casing.

ELEMENTS OF RESPONSE TO QUESTION 10

- RC pump supports are not integral attachments to the pump casing.

- The pump supports are attached to a restraint ring attached to the motor stand.

ATTENDEES: 2/27/86

Name	Organization
E. H. Davidson	FPC, Licensing
Larry Tittle	FPC, Engineering
Santo Ferrarello	Gilbert/Commonwealth Engrng.
Paul Schmitzer	Gilbert/Commonwealth Engrng.
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Harley Silver	NRC
Jim Canning	B&W
Bob Allen	B&W
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Mark Hartzman	U.S. NRC