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IN REPLY REFER TO:
Docket Nos. 50-302
and 50-303

Florida Power Corporation
101 Fifth Street South
St. Petersburg, Florida 33701

Attention: Mr. J. T. Rodgers
Nuclear Project Manager

Gentlemen:

We have completed our initial review of your Preliminary Safety Analysis Report on Crystal River Units 3 and 4. The material that you have submitted does not fully meet our requirements for the contents of applications, as specified in 10 CFR Part 50 and elsewhere. We will need to defer continuation of our review until you amend your application to provide the necessary information.

The proposed Part 50 requires coverage as fully as available information permits on the preliminary design of the facility, including the principal design criteria, the design bases and the relation of the design bases to the principal design criteria and information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. Our Guide for the Organization and Contents of Safety Analysis Reports gives further guidance as to the depth of detail required.

Specifically, the site description does not contain the results of the consolidation grouting program, essential to the evaluation of foundation worthiness and determination of structural adequacy to seismic disturbances. The analysis of site response to the maximum probable hurricane is incomplete in the areas of wave run-up (model description), sea-water drawdown (minimum water level), inlet structure details, and service water pump locations.

The reactor description is deficient in the areas of in-core detectors, core-barrel check valves, the primary pump anti-reverse rotation device, and core design. Your design does not meet, in some areas, the recently published AEC Supplemental Criteria for ASME-III Vessels. We need additional information on the significance of the criteria with respect to your design and construction. Justification for lack of full compliance should be presented where applicable.

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The containment design describes grouted tendons; however in a recent meeting you have discussed a different design. In your clarification of this point, as well as for other applicable matters, you may, if you desire, reference other applications.

We understand that your present emergency core cooling system design departs substantially from that now described in your PSAR. Appropriate updating is required.

The instrumentation and control system is different from previous cases, since it does not provide any direct method for measuring primary flow, either absolutely or relative to a nominal value. Justification of this lessening in plant protection, relative to the Dockets 50-269/270/287/289 (Oconee and Three-Mile Island Unit), will require submittal of detailed design information. Based on the present information, we cannot conclude that there is adequate safety instrumentation for protection against certain loss-of-flow accidents. Specific questions are provided in the attachment to give guidance as to the type of information needed.

This description of the electrical system (Chapter 8) is not adequate because the description of the engineered safety features load distribution is lacking, the description of the off-site power connection to the emergency busses is inadequate, and justification of automatic diesel cross-connection was not provided.

Only one emergency feedwater pump is provided. This is not considered acceptable, because the single-failure criterion is not met. Insufficient details have been included in your application in regard to justifying the turbine stop valves as steam line isolation valves.

We note that no radiation interlocks are provided in the liquid radioactive waste discharge lines. We believe that they should be provided to prevent inadvertent release of radioactive material to the environment.

The safety analysis section is incomplete in the areas of loss-of-flow analysis, fuel handling incidents, steam generator failures, and spectrum-of-break analyses. The incomplete areas are generally attributable to obsolete descriptions, relative to similar nuclear steam supply systems.

Other items are listed in the attachment. This list is not intended to be complete, but does illustrate the kind of information needed. If there are any questions regarding details of these subjects which were not clarified in previous discussions, please contact us.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Attachment:
Additional Information Required

ADDITIONAL INFORMATION REQUIRED
FLORIDA POWER CORPORATION
DOCKETS 50-302/303

1.0 GENERAL

1.1 Describe the extent and manner in which the in-core detector assembly will be used as a tool in determining maximum-to-average ratios during plant operation. Provide the basis for continued plant operation in the event of in-core instrument malfunctions.

1.2 Identify those items that will eventually be classified as technical specifications that now affect plant design.

1.3 Describe how your design complies with General Design Criterion No. 11.

1.4 Update the description of your research and development program status.

1.5 Describe the manner in which Unit 4 will be constructed as related to operation of Unit 3; consider possible blasting, Unit 4 reactor building pressure-test, utilization of heavy equipment in and around the existing auxiliary building, control room, and other shared areas.

1.6 Provide additional justification for the assumption that the 1- to 5-mile zone will not become much more populated during the life of the plant. Will the 5-mile radius be considered as the low-population zone?

2.0 REACTOR AND COOLANT SYSTEM

2.1 We require assurance that the DNBR will not be less than 1.3 (W-3 correlation) at design overpower with consideration given to undetected loss of one or more core barrel check valves. Include the instrumentation response available to the operator that will indicate such valve defects.

2.2 Submit additional design data on the primary pump anti-reverse rotation device. We understand that there has been essentially no experience with the proposed device at high speeds such as will be encountered. Also, it appears that performance testing cannot be accomplished during operation. Finally we understand that it is your position that safety was not a consideration in the decision to provide the device. Provide as confirmation your present position on the above statements.

3.0 CONTAINMENT

3.1 State the criteria for dividing the auxiliary building design into both Class I and Class II zones. Indicate whether the components required for safe shutdown can withstand loss of the Class II components (collapse during the maximum probable earthquake).

3.2 Your design provides capability for reactor building purge during operation. What building pressures would result if the purge valves were open during a LOCA? Is there a break size small enough to prevent (with purge valves open) pressure buildup to 4 psig (isolation pressure) but large enough to incur clad failures? Justify not using ECCS actuation as a building isolation signal.

4.0 ENGINEERED SAFETY FEATURES

4.1 Provide the following information on the core flooding tanks:

(a) method of adding water
(b) immunity of the pair of tanks to a single failure of the N_2 pressurization system
(c) use-rate of N_2 during normal operation
(d) estimated sampling frequency for boron concentration
(e) projected frequency of a full-scale discharge test of a CF tank into the primary system
(f) leak characteristics of relief valves.

4.2 Indicate whether routine testing of the reactor building spray system will include opening of the sodium thiosulfate tank outlet valves.

5.0 INSTRUMENTATION AND CONTROL

5.1 The PSAR states that primary motor status monitors are designed to serve as flow monitors, and that no direct flow measuring devices are provided. In this light, provide:

5.1.1 Design details of a pump monitor, including its mode of operation, range of alarm sensitivity to abnormally high and low currents, independence from other monitors, and response to loss of one of the three phases of motor voltage, change in pump power for range of temperatures, and response to trip of another pump.

5.1.2 A summary of PWR experience or previous designs wherein this design has been used.

5.1.3 Proposed method of determining at the plant startup that design flow rates have been achieved.

5.1.4 Procedures for verifying during the lifetime of the plant that flow rate is not degraded below design values.

5.2 Provide an analysis on the likelihood and consequences of a failure of one in-core instrument tube at the pressure vessel.

5.3 Justify the use of a single bus to energize all control rod clutches. Show how this satisfies the IEEE Proposed Standard.

5.4 The PSAR does not adequately describe the preliminary design of the power-to-flow instrumentation. Please provide a more detailed description and justify the combining of protection and control functions.

5.5 Provide a description of the preliminary design of the operational test system and procedures for the Protection and Engineered Safety Feature channels.

5.6 Clarify the use of the same temperature instrumentation for both protection and control.

5.7 List all protection system channels which provide bypasses and show that the design complies with IEEE Proposed Standard. List all protection system channels which contain variable trip settings and show that the design complies with the IEEE Proposed Standard.

5.8 Provide further details on your radiation monitoring system. For guidance as to appropriate details refer, for example, to the Metropolitan Edison PSAR (Docket 50-289), Supplement 1, Questions 10.1 through 10.8.

5.9 Indicate the means of assuring that those instrumentation and control items that must survive part or all of the LOCA environment have prior qualification performance tests.

5.10 Provide your position on diversification of sensing devices for actuation of the ECCS.

5.11 Will the part-length out-of-core ion chambers provide a signal useful in detection of axial xenon oscillations? If so, provide details such as sensitivity and projected utilization.

6.0 <u>RADIOACTIVE WASTES</u>

6.1 Provide your plans, pre- and post-operational, for survey of marine ecology, More details are needed on environmental monitoring programs.

6.2 Analyze the likelihood and consequences of a failure of the relief valve on one of the waste gas storage tanks.

7.0 <u>CONDUCT OF OPERATIONS</u>

7.1 Summarize emergency procedures planned for the first hour after a major loss-of-coolant accident (LOCA).

7.2 Provide a summary of emergency plans, to include:

(a) shift responsibilities
(b) alarm systems
(c) communication systems
(d) environmental monitoring equipment (portable)
(e) notification of and liaison with authorities
(f) medical facilities
(g) critical actions to be performed prior to evacuation
(h) initial assessment of damage plans
(i) evacuation plans.

7.3 Provide an estimate of accumulated radiation to the operating staff during and after a major LOCA. Include radiation while in the control room, ingress and egress, and possible missions to the turbine building, auxiliary building, and borated water storage tank.

7.4 Provide an outline of the preoperational testing of the engineered safety features that will ensure that design criteria have been met or exceeded.

7.5 State the means by which safety-oriented design or construction changes will be implemented for the time period after construction permit but before operation. Outline the internal review process and the decisional line of authority.

8.0 SAFETY ANALYSIS

8.1 Provide an analysis of the whole body dose rate from noble gases in the control room following the design basis accident.

8.2 Submit a thermal performance analysis of a control rod assembly (CRA) following a LOCA. Include energy deposition rates, heat transfer modes, melting point of control alloy, and cladding performance following fusion of control alloy.

8.3 Provide additional details on the restrictive device on the crane that limits the height of fuel elements during a fuel transfer.

8.4 Provide a flow loss analysis based on this sequence.

(1) The reactor is operating at 74% power; four primary pumps are operating.

(2) A pump with a failed anti-reverse rotation device trips.

(3) The other pump on the same steam generator sends flow back through the failed pump.

(4) Primary flow drops, perhaps below 50% (provide complete details of your analysis).

(5) Reactor power stays at 74% (as no trip will detect this incident).

Provide the minimum DNBR vs time, number of rods (if any) going through DNB, actions by the ICS, and comparison of this incident to the PSAR-expressed thermal design criteria for flow loss. Indicate the relative merits of the reverse rotation device. What means are available to indicate to the operator that this incident has occurred? What would terminate this incident? Is reactor protection sensitive to the rotational inertia of the pump-motor combination. Consider other flow-loss combinations also, as related to the thermal protection criteria.

8.5 Indicate the size of break in the primary coolant system such that the normal makeup system could maintain volume and a normal shutdown/cooldown could be achieved.

8.6 Provide an analysis of chronic iodine release through the combination of 1% failed fuel, leaky steam generator tubes, and leaky safety valves. Include justification for assumed leak rates. Indicate your applicable concentration limits at the site boundary. How will the release be monitored so as to stay within applicable limits?

8.7 Provide an analysis for the accident involving the double-ended rupture of one steam generator tube. To be consistent with other loss-of-coolant accidents, assume coincident loss of off-site power (and thus loss of circulating water to the condenser).

8.8 During the course of a major LOCA the sump water may be hotter than the building atmosphere. Provide justification for your position that the iodine will be picked up in the circulating spray under this condition. Also indicate your plans for spray water cooling, should your research indicate the need.

8.9 Expand your spectrum-of-breaks analysis, for response of ECCS to loss-of-coolant accidents, for break sizes less than 0.4 ft^2.

8.10 Provide an analysis of the dose consequences at the exclusion boundary for the refueling accident. Use as input parameters: 56 fuel rods damaged (from PSAR), 20% of noble gases released, 10% of fuel rod iodine inventory released, 90% iodine retention in the pool water, and 90% iodine retention on the charcoal adsorbers in the building exhaust.

8.11 Provide a thermal shock analysis for the response of the pressure vessel to action of the ECCS. In particular, provide for the fracture mechanics approach:

(1) critical stress intensity factor (K_{IC}) actually used

(2) initial crack geometry and size assumed

(3) equations used to correlate crack size with stress intensity.

9.0 SEISMIC DESIGN

9.1 The foundations for the proposed units receive considerable attention in the PSAR. It is noted on page 2-1 that the principal structures will be founded on limerock. On page 2G-8 it is stated that the reactor building mats or spread foundations will rest on lean concrete or grouted structural fill replacing incompetent rock or soil above the base competent rock. In view of the nature of the cap rock and overlying sediments, as well as the discussion of bedrock solution studies, a detailed description of the locations and types of foundations to be employed for the containment vessels and auxiliary buildings is required.

The additional information supplied concerning the foundations should include a discussion of the steps taken to preclude differential settlement and tilt under both static and dynamic loading conditions.

9.2 With reference to the table of damping values on page 5A-3, confirm that these values will be used in the analysis for both the design and maximum earthquake conditions.

9.3 From a description of the proposed design, it appears that the containment structures will undergo rocking on their foundations. What value of damping is to be employed for the rocking of the structure on the foundation for both the design and maximum earthquakes?

9.4 The method of analysis to be employed in the seismic design is described briefly in Appendix 5C. A more detailed description of the method of analysis is required, and should include a description of the manner in which the containment structure is modeled for the analysis.

9.5 On page 5A-3 it is stated that the vertical and horizontal components (of seismic motion) are assumed to occur simultaneously and their effects added algebraically. It is recommended that the effects associated with the horizontal and vertical earthquake excitation be added directly and linearly as appropriate for the item under consideration, and moreover added directly to the applicable deadload, liveload and operating loads. Clarification of the manner in which the seismic loading effects will be combined with other loadings is required. The load factors for the various inputs should be defined for all components.

9.6 On page 5-16 of the PSAR there is an indication that the tendons will be grouted. No other discussion of this point was found in the PSAR. Further information on the details of grouting and the long term surveillance program for the prestressed tendons is required.

9.7 The liner is noted in the PSAR to consist of 3/8-inch steel plate in the cylinder and dome and 1/4-inch thickness in the base. Additional information concerning the fastening of the liner is required. Additional information must be provided in the manner in which the liner is to be attached to the shell, the stresses under which buckling may occur, and the design provisions that are made to ensure that the buckling can occur without distress or difficulties that will endanger the function of the liner.

9.8 The provisions for carrying shear in the concrete containment vessel are discussed on pages 5c-4 and 5. Further discussion is required of the manner in which the code provisions in Chapters 17 and 26 of ACI 318-63 will be applied to the containment structure, in view of the fact that the containment structure is not an element of the type for which the code was originally written.

9.9 The design of the penetrations receives limited description in the PSAR. Provide additional information on the design technique to be employed with discussion as to how secondary effects arising from thermal loadings, secondary bending, etc., will be handled in the design.

9.10 Cranes in the structure are noted to be Class I components. Additional information is required concerning the design provisions that will ensure that the cranes will not topple during an earthquake or otherwise cause damage which could endanger the safety of the plant.

9.11 With regard to the piping, reactor internals, reactor vessel, and vessel supports, little information is noted in the PSAR concerning the design of these items for seismic and other loadings. For each of these items provide detailed information concerning (a) the loadings that are applicable to the design, and the manner in which the loads are to be combined; (b) the method of analysis to be followed; (c) stress and/or deformation limits for normal operating conditions as well as those conditions involving earthquake loadings; (d) a discussion of the basis for installation of snubbers and/or dampers and their locations; (e) the location of critical valves and their design to preclude difficulties under seismic and other loadings.

9.12 In view of the possible concern about flood conditions that might exist at the plant site, information is required concerning the location of any critical pumps and motors that might be necessary for safe shutdown of the plant and the relationship to the possibility of flooding that might render them inoperable. Also of concern is the seismic design adequacy of the interconnecting line from the service water intake.

9.13 A review of the control instrumentation section reveals no mention of the operation of critical controls under seismic loadings. Provide information concerning the design provisions that are taken to ensure that the critical controls can operate to ensure safe shutdown under seismic loading.

If a battery system is required for emergency shutdown, describe the design of the battery support system, and the provision incorporated to ensure that no damage will occur during an earthquake.

9.14 The cooling water intake structure and associated piping are critical to the safe shutdown of the plant. Additional information is required as to the design of these items, particularly under dynamic loading conditions, and for conditions associated with high water and hurricanes.

9.15 With regard to the prestressed containment building and other critical components, describe in detail the long term surveillance program that is planned to ensure the continuing adequacy of the facility.