Docket File

APR 2 7 1973

Docket No. 50-302

Florida Power Corporation
ATTN: Mr. J. T. Rodgers
Assistant Vice President &
Nuclear Project Manager
P. O. Box 14042
St. Petersburg, Florida 33733

POOR ORIGINAL

Gentlemen:

On the basis of our continuing review of the Final Safety Analysis Report for the Crystal River Unit 3 Nuclear Generating Plant, we find that we need additional information to complete our evaluation. The specific information required is listed in the enclosure.

In order to maintain our licensing review schedule, we will need a completely adequate response by June 8, 1973. Please inform us within seven (7) days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests, it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to retruning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Br. No. 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: See next page

LB

8003160125 A

cc: w/encl

Florida Power Corporation
ATTN: Mr. S. A. Brandimore
Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

DISTRIBUTION
AEC PDR
Local PDR
Docket File
RP Rdg
RCDeYoung
PWR Branch Chiefs
RWKlecker
BCBuckley
EIGoulbourne (2)
PWR-4 Rdg
JHendrie
AKenneke
RWKlecker
OGC
RO (3)

POOR ORIGINAL

OFFICE ▶	PWR-4	L:C/PWR-4				
SURNAME ▶	BCBuckley:kmf	ASchwencer				
DATE ▶	4/ /73	4/ 27 /73				

REQUEST FOR ADDITIONAL INFORMATION

FLORIDA POWER CORPORATION

CRYSTAL RIVER, UNIT 3

DOCKET NO. 50-302

5.0 CONTAINMENT SYSTEMS AND SPECIAL STRUCTURES

5.34 Discuss how the instrument lines penetrating the reactor building conform to regulatory position E2 of the supplement to Regulatory Guide 11.

6.0 ENGINEERED SAFETY FEATURES

6.12 Provide a breakdown of the static elevation head, the friction head loss in the suction piping, the vapor pressure of the fluid and the reactor building pressure, in feet of water, utilized in the post-accident NPSH calculations reported in Table 6-11A for the reactor building spray system pumps. In addition, provide the sump temperature that was utilized in the vapor pressure determination that was utilized in the vapor pressure determination and discuss the relationship of this sump temperature to the sump temperature provided in Figure 14-61, the time building spray is initiated and the time suction is switched from the borated water storage tank to the reactor building sump.

6.13 Describe the reduction in available building pressure utilized in the NPSH calculation of Table 6-11A considering maximum reactor building safety features (e.g. 3 air coolers and 2 spray systems).

14.0 SAFETY ANALYSIS

14.20 For the evaluation of reactor building response following a design basis LOCA, it is not apparent that the internal vent valves will preclude steam venting through the steam generators for a cold leg, pump suction break. We will need the results of your reactor building pressure transient analyses for a spectrum of cold leg breaks Include the effect of post-blowdown energy sources, such as core stored energy and decay heat, primary system metal stored energy, and steam generator stored energy. The analyses should be extended through the initial blowdown, reflood, and post-reflood phases of the postulated accidents.

14.21 Provide a detailed description of the core reflood model that is used following reactor coolant system blowdown, include the assumptions used to develop the model, e.g., the hydraulic modeling of the reactor coolant system, the resistances of components (reactor coolant pump, steam generator, piping, reactor core, and internal vent valves), the method used to compute boiloff of the reflooding water and energy sources (core stored energy, decay heat, thick and thin metal stored energy, and steam generator stored energy.

14.22 For a cold leg break resulting in the highest calculated reactor building pressure, provide data in the form of tables of mass release rate to the reactor building and enthalpy of the mass released throughout the blowdown and reflood phases of the accident. Include a graph of core inlet velocity as a function of time for the reflood phase of the accident.

14.23 The FSAR indicates that the 8.5 ft^2 hot leg break results in the highest reactor building pressure. The analyses for a spectrum of hot leg breaks should be reanalyzed utilizing applicable methods and assumptions used in the cold leg break.

14.24 Provide a description of the analytical methods, computer programs and general parameters utilized in the hot leg rupture analyses if they differ from that presented in the FSAR.

14.25 For a hot leg break resulting in the highest calculated reactor building pressure, provide tables of mass release rate to the reactor building and enthalpy of the mass released as functions of time throughout the blowdown and reflood phases of the accident.

14.26 If the heat sink film coefficients given in Table 14-40 of the FSAR are utilized in the reanalysis of the reactor building pressure, provide justification for the condensing film coefficents from the reactor building atmosphere to the steel heat sinks and for heat transfer to unpainted concrete surfaces.

14.27 Analyze the reactor cavity and the steam generator compartments for the pressure response considering a homogeneous steam-water-air mixture with appropriate correlations for sonic flow through the vents. A vent discharge coefficient of 0.5 should be used, and reactor blowdown calculations should assume a discharge coefficient of 1.0. A range of break sizes should be considered for each subcompartment.

14.28 Describe the analytical model, including assumptions and appropriate bases, used in calculating the subcompartment pressure response.

14.29 Provide a flow diagram showing the free volume and the vent area of all subcompartments and compartment subdivisions, and the flow interconnections considered in the subcompartment pressure analysis. In addition, provide justification for the selection of the subdivisions of each compartment.

14.30 Provide the mass and energy blowdown rates as functions of time and location used in each subcompartment analysis.

14.31 Discuss the results of analyses used to justify that the break locations selected result in the highest calculated subcompartment pressures. Include a description of the correlations used for determining subcooled blowdown rates.

14.32 Provide the method and results of analysis of the jet forces which can impinge on the reactor cavity and steam generator structures.

14.33 Discuss how the reactor coolant system blowdown analyses for the loss-of-coolant accidents were modified to assure a conservative calculation of the mass and energy releases to the reactor building for the reactor building and subcompartment pressure transient analyses (e.g. maximum emergency core cooling should be assumed and maximum cladding surface film coefficients should be utilized).

14.34 Provide the continuous purging rate and duration utilized in arriving at the whole body and thyroid dose calculations in the FSAR Table 14B-2.1 for the MHA Regulatory Guide 1.7 (formerly Safety Guide 7) case.

14.35 Figure 14B-5.8 of the FSAR indicates that an air compressor will be rented in the event that purging is necessary. Provide justification as to the assured availability and capacity of the required air compressor under such an arrangement.

14.36 The installed equipment and piping of the combustible gas control system should be designed to Class I seismic requirements, quality Group B standards and should provide necessary capacity considering a single active failure. Identify and justify any deviations from these requirements.

14.37 Describe the provisions made to ensure mixing of hydrogen with the reactor building subcompartments. Consider the long-term (i.e. several days following the rupture) evolution of hydrogen resulting from radiolysis in the core. Show that sufficient mixing would occur within each compartment to preclude high local concentrations of hydrogen in excess of the recommended limits specified in Regulatory Guide 1.7.