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REPORT TO AEC REGULATORY STAFF

ADEQUACY OF THE STRUCTURAL CRITERIA FOR CRYSTAL RIVER UNITS 3 AND 4 NUCLEAR GENERATING PLANT

Florida Power Corporation

(AEC Docket Nos. 50-302 and 50-303)

by

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and

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INTRODUCTION

This report concerns the adequacy of the containment structures and components, reactor piping and reactor internals for the Crystal River Units 3 and 4 Nuclear Generating Plant, for which application for a construction permit has been made to the U. S. Atomic Energy Commission (AEC Dockets No. 50-302 and 50-303) by the Florida Power Corporation. The facility is to be located in the northwestern portion of Citrus County, Florida, on the Gulf of Mexico between the mouths of the Withlacoochee and Crystal Rivers, and approximately 7½ miles NW of Crystal River, and 70 miles N of Tampa, Florida. Specifically this report is concerned with the evaluation of the design criteria that determine the ability of the containment system, piping, and reactor internals to withstand a design earthquake acting simultaneously with other applicable loads forming the basis of the design. The facility also is to be designed to withstand a maximum earthquake simultaneously with other applicable loads to the extent of insuring safe shutdown and containment. This report is based on information and criteria set forth in the Preliminary Safety Analysis Report (PSAR) and supplements thereto as listed at the end of this report. We have participated in discussions with the AEC Regulatory Staff, and the applicant and its consultants, in which many of the design criteria were discussed in detail.

DESCRIPTION OF THE FACILITY

The Crystal River Nuclear Generating Plant is described in the PSAR as a pressurized water reactor nuclear steam supply system furnished by the Babcock and Wilcox Company and designed for an initial power output of 2452 MWt

(855 MWe net) for each unit. The reactor coolant system consists of the reactor vessel, coolant pumps, steam generators, pressurizer, and interconnecting piping. For each reactor there are two steam generators. The reactor vessel will have an inside diameter of about 14.3 ft., a height of 41.7 ft., is designed for a pressure of 2500 psig and a temperature of 650°F, and is made of SA-302, Grade B, steel clad with type 304 austenetic stainless steel.

The containment for this plant consists of two systems as follows: (1) the reactor building which provides biological and missile shielding, and which contains the energy and material that might be released by an accident; and (2) the engineered safeguards systems which limit the maximum value of the energy released by an accident.

The reactor building, which encloses the reactor and steam generators, consists of a steel lined concrete shell in the form of a reinforced concrete vertical cylinder with a flat base and a shallow dome roof. The cylindrical structure of 130 ft. inside diameter has side walls rising 157 feet from the top of the foundation slab to the spring line of the dome roof. The concrete side walls of the cylinder and dome will be approximately 3 ft. 6 in. and 3 ft. 0 in. in thickness, respectively. The foundation mat will be approximately 9 ft. thick with a 2 ft. thick concrete slab over the bottom liner plate. The foundation slab will be reinforced with conventional steel reinforcing. The cylindrical walls will be prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof will be prestressed utilizing a three-way post-tensioning system. The inside surface of the reactor building will be lined with a carbon steel liner 3/8 in. thick for the cylinder and dome and 1/4 in. thick for the base. The reactor building is essentially the same as the containment buildings for the Turkey Point, Oconee, and Three Mile Island plants.

Personnel and equipment access hatches are provided for access to the reactor building. In addition there are other penetrations for piping and electrical conduits.

The engineered safeguards for each nuclear unit consist of the emergency core injection system and the reactor building atmosphere cooling and washing system.

Other Class I components and systems whose design must include consideration of seismic effects are listed in Appendix 5A and include such items as the spent fuel cooling system and shutdown cooling system, reactor control room and equipment, and the post-incident air filtration system. Some of these items are located totally or partially outside of the reactor building.

The facility includes a cooling water intake and pump structure located at the foot of the intake canal about 400 ft. from the reactors.

The bedrock at this site is located approximately 20 ft. beneath the present ground surface. The surface overburden consists in the upper layers of approximately 3 to 5 ft. of surface fill, followed by the natural soil cover consisting of deposits of thinly laminated organic sandy silts and clays interspersed with marine deposits, and in turn overlying a residual limy soil unit derived from the decomposition of the underlying bedrock. The bedrock consists of biogenic carbonates of Tertiary Age. The uppermost bedrock member is that identified as the Inglis member which is characterized by a cream-colored to an occasionally tan, porous, granular, biogenic limestone and dolomite deposited in a shallow marine environment.

The closest evidence of possible faulting occurs at a distance of three miles to the east of the site. Studies of the site show no evidence of existence of subsurface faults.

SOURCES OF STRESSES IN REACTOR BUILDING AND CLASS I COMPONENTS

The reactor building is to be designed for the following loadings: dead load; live loads (including roof loads, pipe penetration reactions, and crane loads); internal pressure due to loss of coolant accident of 55 psig; test pressure of 63.3 psig; negative pressure of 2.5 psig; accident temperature of 281°F and operating temperature of 110°F; wind loads corresponding to roof line load of 35 psf; tornado wind loading (300 mph tangential wind velocity, external vacuum of 3 psig. and missile loading); prestressing loads; and seismic loading as described next.

The seismic design of the reactor building is based on the response to a maximum horizontal ground acceleration of 0.05g. Also, the design is to be checked to insure no loss of function for an earthquake based on a maximum horizontal ground acceleration of 0.10g.

The piping, internals, and vessel support design procedure is outlined in answer to question 9.11 of Supplement No. 1. Therein it is noted that these items will be designed for various loading combinations as listed in Table 1, including the design load, the design earthquake and pipe rupture loads. In addition a discussion of modes of deformation of reactor internals, and the allowable deformations are presented in Table 2.

As noted in Appendix 5A, all Class I structures, components and systems will be designed for primary steady-state stresses combined with the appropriate seismic stresses, and where applicable, in accordance with the appropriate codes. In the case of primary steady-state stress combined with the seismic stress resulting from the maximum earthquake, the response is to be limited so that the function of the component, system, or structure shall not be impaired to prevent a safe and orderly shutdown.

COMMENTS ON ADEQUACY OF DESIGN

Foundations

The applicant has proposed to found the mat foundation for Crystal River Units 3 and 4 on a structural fill composed of crushed limestone. The base of the structural fill is planned to be at about elevation 73 and will extend up to about elevation 80. Quality control of the crushed limestone fill and the 98 percent maximum Modified Density (ASTM Test Designation D1557-66-T) requirement as noted in Ref. 3(a) will be adequate to assure a structural fill with satisfactory stress-strain properties.

Because the exploratory investigation revealed the presence of both open and filled solution cavities in the limestone bedrock beneath the site, the applicant proposes to undertake consolidation grouting beneath the reactor building to about elevation 30 and beneath other structures to about elevation 60. From the information presented in the foundation grouting report on Unit 2 (Ref. 3(b)) and the report on the test grouting program for Crystal River Units 3 and 4 (Ref. 3(c)), it appears that the modified split-spaced hole procedure utilized on Unit No. 2 will be adequate for the foundation of Units 3 and 4. The effectiveness in providing a curtain wall around the area to be grouted is illustrated quite clearly by Fig. 5 of Ref. 3(b) which shows a graph of hole order versus unit grout take. The graph illustrates that the grout takes approach reasonable limits in the Tertiary and Quaternary holes. It is understood that the grouting specifications for the grouting contract are flexible to the extent that the decision on the hole order at which grouting will be stopped is to be decided by the field engineer. It would be our recommendation in the application of this procedure that the unit grout takes be reduced to 0.5 to 1.0 cubic feet per lineal foot f hole before grouting is stopped. From all the available data for the site thus far (Refs. 3(b) and 3(c)) it appears as if this result will be accomplished on either the Tertiary or Quaternary consolidation grout holes at 8 ft. and 4 ft. spacings, respectively, if a curtain wall is first established around the area to be grouted. We believe the proposed structural fill and grouting program will be adequate to prevent excessive differential supplement of the reactor buildings and appurtenant structures.

Seismic Design

All structures, components, and systems classified in Class I are to be designed for a design earthquake based on a maximum horizontal ground acceleration of 0.05g. Such items are also to be designed for a maximum earthquake based on a maximum horizontal ground acceleration of 0.10g so as not to impair or prevent a safe and orderly shutdown of the plant. These desig levels are in agreement with those proposed by the U. S. Coast and Geodetic Survey (Ref. 4) and we concur in these design criteria.

The response spectra to be employed in the design are given in Fig. 3 of Appendix 2I. The response spectrum shown is for five percent gravity, the design earthquake, and we concur in the use of the spectra as shown on the assumption that at periods greater than 1.0 sec. (not shown) the spectra do not drop sharply but remains essentially at the spectral velocity levels at which the present plot is cut off. We assume that the response spectra for use in design for the maximum earthquake loading condition will be twice the values of the spectra just described.

The vertical component of earthquake excitation will be taken as two-thirds of the horizontal component and will be assumed to occur simultaneously with the horizontal component. We concur in this criterion.

It is noted in Section 6 of Appendix 5A that "The respective vertical and horizontal seismic components at any point on the shell will be added by summing the absolute values of the response (i.e., stress, shear, moment, or deflection) of each contributing frequency due to vertical motion and adding the resultants to the corresponding absolute values of the response of each contributing frequency due to horizontal motion." The seismic stresses are then to be added directly to the dead load, live load, operating loads, and accident (pressure and temperature) loading conditions in accordance with the loading expressions presented in Appendix 5B. From this one can infer that the seismic stresses are added linearly and directly with the other applicable stresses, and on the basis of this assumption we concur in the design approach.

The damping values to be employed in the dynamic analysis are given in Section 5 of Appendix 5A. These values are to be employed for both the maximum and design earthquake. As noted in answer to Question 9.3 of Supplement 1, a damping value of 5 percent of critical will be used for both the design and maximum earthquake for rocking effects for the foundation. We concur in the use of these values in the design.

The general method of dynamic analysis will be either a modal analysis or will be carried out in accordance with the procedure outlined briefly in Section 6 of Appendix 5A. The discussion presented in Section 6 suggests that for systems such as piping systems which are highly complex geometrically, that the analysis may be carried out as for a single-degree-of-freedom system. We do not corcur in this approach in general, and it is our recommendation that a formal dynamic analysis be performed for Class I structures equipment, piping, and reactor internals as appropriate, especially for those systems which are vital to safety of the plant.

Further information on the dynamic piping analysis is included in the answer to Question 9.12 and provides some clarification to the discussion presented in Section 6 of Appendix 5A. However, it is noted that the description given only applies to the dynamic analysis of piping systems supported at fixed points. The applicant is requested to provide additional information concerning the methods of dynamic analysis that will be employed for the piping systems. Additional comments on the analysis of piping systems appear later in this report.

The method of analysis to be employed for the reactor building is described in Section 2.2 of Appendix 5C and we concur in the approach as outlined there.

All structures and components classified as Class II are to be designed for a ground acceleration of 0.05g in accordance with the procedures of the Uniform Building Code. We are in agreement with this approach.

General Design Provisions

The load combination equations to be employed in the design of the reactor building are presented in Section 1.3 of Appendix 5B. We are in general agreement with the combinations to be employed with one exception, namely that of load expression "c" wherein it is our belief that a term reflecting the accident pressure load is missing. Clarification of this point by the applicant is requested.

The design stress criteria for the reactor building are presented in Appendix 5B and 5C. It is noted therein that the load deformation behavior of the structure is one of elastic, low strain response. The building will be checked for the factored loads and load combinations, compared with the yield strength of the structure, and the load capacity is to be defined as the upper limit of the elastic behavior of the effective load carrying structural materials. The deformation of the structure is to be such that the compressive strain in the steel liner does not exceed 0.005 in/in. nor to cause average tensile strains to exceed that corresponding to the minimum yield stress. Membrane tension will be limited to $3\sqrt{f'_c}$ and it is noted further that when principal flexural tension exceeds $6\sqrt{f'_c}$ due to thermal gradients through the wall, non-prestressed reinforcing will be added to resist thermal stresses. It would be our recommendation that no net membrane tension be permitted in the containment shell but on the assumption that the latter statement refers to the combination of membrane tension combined with flexural tension arising from pressure or thermal effects we concur in the general design provisions noted.

The reinforcing steel to be employed in the plant will consist of either ASTM A-15, A-408, A-431, or A-432. It is noted in Appendix 5B that arc welding for reinforcing splices will not be employed and that Cadweld splices will be used when required. We are in agreement with this approach.

The liner is to be designed so that the critical buckling stress will be greater than the proportional limit of the steel. Present analysis, according to the PSAR indicates that the basic accident conditions produce a strain of approximately 0.002 in/in. in the liner. The liner is to be analyzed as a flat plate and the liner anchors, which will be vertical angles, are to be spaced horizontally at 18 in. center to center. The liner anchors are to be designed such that the welds connecting the anchors to the liner will fail before the liner is breached. Generally we concur in this design approach for the liner, although it is not clear how this type of attachment may affect the buckling strength and long-term service performance of the liner.

A discussion of the general design criteria for handling differential settlements and relative motions under seismic response is presented in Appendix 5A and we are in agreement with the general concepts presented there.

The post-tensioning stressing system to be employed will consist of either the S.E.E.E. or the BBRV system. In general, the design concepts to be employed in the prestressing are similar to those employed in other plants designed by Gilbert Associates such as Turkey Point, Oconee and Three Mile Island. The reactor tendons, which are unbonded, will be protected from corrosion by insertion of a protective coating in the tendons. The steel portions of the plant will be connected electrically to provide protection against stray currents. It is noted in the PSAR that the tendon inspection program could be made if it appeared desirable. It is our recommendation that a reasonable inspection program be implemented, especially in view of the location of this plant near a salt water environment.

Piping, Reactor Internals, Reactor Vessel and Vessel Supports

The design approach to be employed for the piping, and reactor internals, which also would include for the most part the design of the engineered safeguard system, are to be designed for general criteria as outlined in the PSAR, namely in accordance with applicable ASME codes and procedures outlined in AEC Publication TID-7024. A further more detailed discussion of the design approach is presented in answer to Question 9.11 of Supplement 1.

The possible modes of deformation of reactor internals are summarized in Table 2 of the answer to Question 9.11 and involve values labeled "allowable" and "no loss of function." It is noted in the discussion preceding the table that the "no-loss-of-function" deformations could cause safety problems, and that the "allowable" deformations are those that are used as design limits. It is not clear whether these design limits refer to those associated with the design earthquake loading condition or the maximum earthquake loading condition or even a combination of seismic loading with other applicable loadings. Clarification of this point by the applicant is requested, in order that a better judgment on the margin of safety inherent in the design can be made.

The approach to be employed for the piping appears to be patterned after that presented in ASME applicable codes and in Westinghouse Electric Corporation Report WCAP-5890 Rev. 1, 1967. However, the approach presented is limited in that it relates solely to the margin of safety with regard to stress levels and does not provide information on the margin of safety with regard to permissible strain or deformation. With regard to the presentation encompassing possible strain hardening, no information is presented to form a judgment as to whether the stress analysis conforms to real property materials, and moreover whether localized stresses or deformations are included in the analysis. Further information concerning the design criteria to be employed for the piping, particularly with respect to the maximum earthquake loading condition, is requested.

Instrumentation and Controls

The design of the control instrumentation for seismic effects is discussed in answer to Question 9.13 of Supplement 1. Therein it is noted that "the components in the reactor protection system and safeguard evacuation system will suffer no loss of function at accelerations of 0.1g horizontal and 0.067g in vertical condition." A similar comment is given concerning the batteries and battery mounts. We can not concur in this approach, for an analysis may show that the instrumentation can be subjected to larger accelerations. Also, will the instrumentation function under conditions of moderate tilting?

Flooding

Information concerning possible flooding of the site is presented in Appendix 2C and in answer to Question 9.12 of Supplement 1. The protection provided against flooding appears adequate to us.

Cranes

The polar crane in the reactor building is a Class I component and is noted in Appendix 5A that the design will be made to insure stability during an earthquake. It is noted in answer to Question 9.10 that other handling bridges which are not considered Class I equipment are also provided with anti-derailing devices. The design criteria for the cranes are acceptable to us.

Penetrations

It is noted in Section 7 of Appendix 5B that the penetrations will be designed for the load combinations applicable to the reactor building and will be analyzed by using the finite element technique developed by the Franklin Institute Research Laboratories. Smaller penetrations will be designed in accordance with published and accepted procedures as noted in the discussion presented in Appendix 5B. We are in general agreement with the design approaches outlined briefly in Section 7 of Appendix 5B.

CONCLUSIONS

In line with the design goal of providing serviceable structures and components with a reserve in strength and ductility, and on the basis of information presented, we believe the design criteria outlined for the containment and other Class I components including the reactor internals, and piping, vessels and supports, can provide an adequate margin of safety for seismic resistance. However, in arriving at this conclusion we have noted, in the report several items for which additional information is required from the applicant, namely information concerning the analysis of the piping under dynamic loading, stress criteria for piping, design criteria for the reactor internals, design of instrumentation and controls, and clarification of the load combination expressions.

REFERENCES

1. "Preliminary Safety Analysis Report, Vols. 1, 2, 3, and Appendices," Crystal River Units 3 and 4 Nuclear Generating Plant, Florida Power Corporation, 1967.

2. "Preliminary Safety Analysis Report, Amendments 1 and 2," Crystal River Units 3 and 4 Nuclear Generating Plant, Florida Power Corporation, 1968.

3. "Preliminary Draft Reports (to be filed with AEC)

   (a) "Foundation Investigation -- Proposed Nuclear Power Plant -- Florida Power Corporation," Woodward, Clyde, Sherard and Associates, February 7, 1968.

   (b) "Foundation Grouting Report - Unit 2," Crystal River Plant, Florida Power Corporation, by Gilbert Associates, Inc., Report No.1657, January 30, 1968.

   (c) "Test Grouting Program - Units 3 and 4," Crystal River Units 3 and 4, Florida Power Corporation, Gilbert Associates, Inc., Report No. 1658, January 30, 1968.

4. "Report on the Seismicity of the Crystal River Site," U. S. Coast and Geodetic Survey, Rockville, Maryland,____________.