MAY 3 1 1972

Richard C. DeYoung, Assistant Director for PWR's, Directorate of Licensing

OCONEE NUCLEAR STATION, UNITS 2 AND 3, DOCKET NOS. 50-270 AND 50-287

Adequate responses to the enclosed request for additional information are required before we can complete our review of the subject application. These requests, prepared by the L Mechanical Engineering Branch, concern the reactor coolant pressure boundary, reactor internal structures, safety related mechanical systems, seismic design criteria and pipe whip criteria submitted in Volumes I through IV of the FSAR. This request supersedes our previous request dated 4-27-72 and reflects the results of the reevaluation discussed in our letter dated 5-27-72.

The applicant's description of seismic qualification testing for instrumentation and equipment (Section 7A.2) appears to be extracted from B & W Topical Report BAW-10003, "Qualification Testing of Protection System Instrumentation" (March 1971). The additional information required to complete our review of this report was forwarded in our request of March 13, 1972 for the Three Mile Island Station Unit No. 1, Docket No. 50-289. Our review of the Oconee 2/3 application can therefore be expedited if the applicant will agree to reference BAW-10003 and provide the requested information.

The applicant has referenced (Section 14) B & W Topical Report BAW-10008, "Reactor Internals Stress and Deflections due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake" (June 1970). The sections of the report applicable to Oconee are currently under review by the MEB and additional information may be required prior to completion of this review.

Original Signed by
R. R. Maccary

DKT #50-287

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Form AEC-318 (Rev. 9-53) AECM 0240

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REQUEST FOR ADDITIONAL INFORMATION

OCONEE NUCLEAR STATION UNITS 2 & 3

DOCKET NOS. 50-270/287

3.6 CRITERIA FOR PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH A LOCA

1. Provide a more detailed description of the measures that have been used to assure that the containment liner and all essential equipment within the containment, including components of the primary and secondary coolant systems, engineered safety features, and equipment supports, have been adequately protected against blowdown jet forces, and pipe whip resulting from a loss-of-coolant accident. The description should include:

   a. Pipe restraint design requirements to prevent pipe whip impact.

   b. The features provided to shield vital equipment from pipe whip.

   c. The measures taken to physically separate piping and other components of redundant engineered safety features.

   d. A description of the type of pipe whip restraints and the location of all restraints.

2. Describe the dynamic system analysis methods and procedures that were used to confirm the structural design adequacy of the reactor coolant system (unaffected loop) and the reactor internal loadings. The following information should be included:

a. the locations of the postulated double ended pipe rupture on which dynamic analyses were based.

b. the rupture type(s), such as circumferential and/or longitudinal break(s), for each postulated rupture location.

c. the description of the forcing functions used for the pipe whip dynamic analyses. The function should include direction, rise time, magnitude, duration and initial conditions. The forcing function should adequately represent the jet stream dynamics and the system pressure differences.

d. a description of the mathematical model used for the dynamic analysis.

e. analyses performed to demonstrate that unrestrained motion of ruptured lines will not sever adjacent impacted piping or pierce impacted areas of containment liner.

3.7.2 SEISMIC SYSTEM ANALYSIS

1. Confirm the validity of a fixed base assumption in the mathematical models for the dynamic system analyses by providing summary analytical results that indicate that the rocking and translational responses are insignificant. A brief description should be included of the method, mathematical model and damping values (rocking vertical, translation and torsion) that have been used to consider the soil-structure interaction.

2. Describe the method employed to consider the torsional modes of vibration in the seismic analysis of the Category I building structures. If static factors are used to account for torsional accelerations in the seismic design of Category I structures, justify this procedure in lieu of a combined vertical, horizontal, and torsional multimass system dynamic analysis.

3. The use of both the modal analysis response spectrum and time history methods provides a check on the responses at selected points in the station structure. Submit the responses obtained from both of these methods at selected points in the Category I structure to provide the basis for checking the seismic system analysis.

4. Provide the dynamic methods and procedures used to determine Category I structure overturning moments. Include a description

of the procedures used to account for soil reactions and vertical earthquake effects.

5. Provide the analysis procedure followed to account for the damping in different elements of the model of a coupled system. Include the criteria used to account for composite damping in a coupled system with different structural elements.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

1. Provide the criteria for combining modal responses (shears, moments, stresses, deflections. and/or accelerations) when modal frequencies are closely spaced and a response spectrum modal analysis method is used.

2. With respect to Category I piping buried or otherwise located outside of the containment structure, describe the seismic design criteria employed to assure that allowable piping and structural stresses are not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures.

3. Describe the evaluation performed to determine seismic induced effects of Category II piping systems on Category I piping.

4. Provide the criteria employed to determine the field location of seismic supports and restraints for Category I piping, piping system components, and equipment, including placement of snubbers and dampers. Describe the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

3.7.4 CRITERIA FOR SEISMIC INSTRUMENTATION

Provide the following information with respect to the use of the seismic instrumentation for the facility:

1. Discuss the seismic instrumentation provided and compare the proposed seismic instrumentation program with that described in AEC Safety Guide 12, "Instrumentation for Earthquakes." Submit the basis and justification for elements of the proposed program that differ substantially from Safety Guide 12.

2. Provide a description of the seismic instrumentation such as peak recording accelerographs and peak deflection recorders, that will be installed in selected Category I (Class 1 Seismic) structures and on selected Category I (Class 1 Seismic) components. Include the basis for selection of these structures and components, the basis for location of the instrumentation, and the extent to which this instrumentation will be employed to verify the seismic analyses following a seismic event.

3. Describe the provisions that will be used to signal the control room operator the value of the peak acceleration level experienced in the tendon access gallery of the reactor containment structure to the control room operator within a few minutes after the earthquake. Include the basis for establishing the predetermined values for activating the readout of the accelerograph to the control room operator.

4. Provide the criteria and procedures that will be used to compare measured responses of Category I (Class 1 Seismic) structures and the selected components in the event of an earthquake with the results of the system dynamic analyses. Include consideration of different underlying soil conditions or unique structural dynamic characteristics that may produce different dynamic responses of Category I (Class 1 Seismic) structures at the site.

3.7.5 SEISMIC DESIGN CONTROL MEASURES

Describe the design control measures (as specified in Appendix B - of 10 CFR Part 50 - "Quality Assurance Criteria for Nuclear Power Plants") implemented to assure that appropriate seismic input data (as derived from seismic system and subsystem analyses including any necessary feedback from such analyses) are correctly specified to the manufacturer of Category I components and equipment to constructors of other Category I structures and systems. The responsible design groups or organizations that will verify the adequacy and validity of the analyses and tests employed by manufacturers of Category I components and equipment and constructors of Category I structures and systems should be identified. A description of the review procedures employed by each group or organization should be included.

3.9.1 DYNAMIC SYSTEM ANALYSIS AND TESTING

1. Paragraph I701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to prevent excessive vibration under startup and initial operating conditions. Submit a discussion of your vibration operational test program which will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than the reactor coolant pressure boundary that will result in significant vibration response of reactor coolant pressure boundary systems and components.

2. Discuss the testing procedures used in the design of Category I mechanical equipment such as fans, pumps, drives, valve operators and heat exchanger tube bundles to withstand seismic, accident and operational vibratory loading conditions, including the manner in which the methods and procedures employed will consider the frequency spectra and amplitudes calculated to exist at the equipment supports. Where tests or analyses do not include evaluation of the equipment in the operating mode, describe the bases for assuring that this

equipment will function when subjected to seismic and accident loadings.

3. Provide a brief description of the dynamic system analysis methods and procedures used to determine dynamic responses of reactor internals and associated Class I components of the reactor coolant pressure boundary (e.g., analyses and tests). The discussion should include the preoperational test program elements described in Safety Guide 20, Vibration Measurements on Reactor Internals. In the event elements of the program differ substantially from the requirements of Safety Guide 20, the basis and justification for these differences should be presented.

4. Provide a discussion of the preoperational analysis and testing results that will be used to augment the LOCA dynamic analysis methods and procedures, i.e., barrel ring and beam modes, guide tube responses, water mass and compliance effects, damping factor selection, etc.

3.9.2 ASME CODE CLASS 2 AND 3 COMPONENTS

1. The FSAR states that faulted operating condition categories have been applied to certain reactor coolant system components. Identify any other components or systems that are not a part of the reactor coolant pressure boundary for which the design stress limits associated with faulted conditions were applied. If faulted conditions are used for such cases, then provide justification for applying such conditions, including the bases for the loading conditions and combinations, and associated design stress limits which were applied.

   In addition, for all components and systems comparable to ASME Code Class 2 and 3, provide the design condition categories (normal, upset or emergency), the associated design loading combinations and the design stress limits which will be applied for each loading combination. This information may be submitted in tabular form as suggested below:

System and/or Component	Design Loading Combinations	Design Condition Categories (Normal, Upset, or Emergency)	Design Stress Limits

2. If any design stress limits allow inelastic deformation (or are comparable to the faulted condition limits defined in ASME Section III for Class I components) then provide the bases for the use of inelastic design limits by demonstrating that the component will maintain its functional or structural integrity under the specified

design loading combination. Include a brief description of the methods and design procedures that were used in such cases.

3. Describe the design and installation criteria applicable to the mounting of the pressure-relieving devices (safety valves and relief valves) for the overpressure protection of systems with Class 2 components. In particular, specify the design criteria used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event all the valves are required to discharge. Indicate the provisions made to accommodate these loads.

5.2.1 DESIGN CRITERIA, METHODS AND PROCEDURES (REACTOR COOLANT PRESSURE BOUNDARY)

1. Categorize all transients or combinations of transients listed in Table 4-8 of the FSAR with respect to the conditions identified as "normal", "upset", "emergency", or "faulted" as defined in the ASME Section III Nuclear Component Code. In addition, provide the design loading combinations and the associated stress or deformation criteria.

2. Table 4-20 of the FSAR includes faulted condition stress limits for pump casings. Describe the criteria employed to assure that active* components will function as designed in the event of a postulated piping rupture (Faulted Condition) within the reactor coolant pressure boundary (e.g., allowable stress limits established at or near the yield stress calculated on an elastic basis). Where empirical methods (tests) are employed, provide a summary description of test methods, loading techniques and results obtained including the bases for extrapolations to components larger or smaller than those tested.

3. The design criteria which was used to account for full discharge loads (i.e., thrust, bending, torsion) imposed on safety and relief valves and connected piping in the event all valves are required to discharge, including the provision made to accommodate these loads should be specified.

*Active components of a fluid system (e.g., valves, pumps) are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the reactor coolant pressure boundary.

5.2.2 <u>OVERPRESSURIZATION PROTECTION</u>

1. To facilitate review of the bases for the pressure relieving capacity of the reactor coolant pressure boundary, submit (as an appendix to the FSAR) the "Report on Overpressure Protection" that has been prepared in accordance with the requirements of the ASME Section III Nuclear Power Plant Components Code or, if the report is not available, indicate the approximate date for submission. In the event the report is not expected to be available until either the Operating License review or late in the construction schedule for the plant, provide in the FSAR the bases and analytical approach (e.g., preliminary analyses) being utilized to establish the overpressure relieving capacity required for the reactor coolant pressure boundary.