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THRU: R. C. DeYoung, Chief
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Division of Reactor Licensing
OCONEE NUCLEAR STATION -- REQUEST FOR ADDITIONAL INFORMATION

50-269, -270

As requested in your memorandum of March 23, 1967, we have reviewed the Preliminary Safety Analysis Report for the Duke Power Company's Oconee plant. Attached is a list of additional information requested concerning the reactor coolant system.

Attachment:
As stated above

cc: R. C. DeYoung
S. Levine
R. S. Boyd
D. J. Skovholt
B. Grimes

Distribution:
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S. S. Pawlicki

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SURNAME ▶	SSPawlicki:re	RCDeYoung				
DATE ▶	4/14/67	4/24/67				

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REACTOR COOLANT SYSTEM

Please provide the following information concerning the design adequacy of components in the reactor coolant system (reactor vessel, steam generators, piping, pump casings and pressurizer):

a. We understand that a number of steps are specified in the design, quality control, and fabrication of these components above and beyond the requirements of the applicable ASME and ASA codes. Please provide a list of these specifications which improve the quality and provide a greater safety margin for each component.

b. Discuss the provisions incorporated in the design to detect and locate the source of leakage from each of these components during operation. State the criteria for corrective action. Estimate the sensitivity of the leakage detection methods employed.

c. State your criteria in terms of stresses or deflections for the combined hypothetical earthquake and accident loadings. This should include internal pressure, seismic forces, blowdown reaction forces and rapid temperature variations.

d. Describe the provisions to allow for in-service inspection of inner and outer surfaces of the reactor coolant system components. Indicate the test method to be used (i.e., UT, dye penetrant, etc.) for each area, and discuss the ability to detect flaws in the material.

Outline the proposed frequency of in-service inspection of the various areas of the reactor coolant system.

Please provide a drawing showing vertical cross-section of the reactor vessel, insulation and the primary shield. Particularly, show the arrangement of these materials around the reactor vessel nozzles, and indicate which parts are removable.

Reactor Vessel

Please provide a summary of the reactor vessel stress analysis listing:

a. The design basis
b. Loading conditions
c. Methods of analysis used
d. Distribution (preferably graphical) of calculated stresses in the vessel shell, nozzle area and the vessel head, both

at the inner and the outer surfaces of the vessel, during normal operation of the plant.

Provide the highest stress intensity in the nozzle area of the reactor vessel and state the assumed loading conditions.

State the maximum stress intensity at the base of the control rod housings on the vessel head due to pressure and seismic loading.

Please state if Table 4-7 includes transient loading due to safety injection of cold water. Is the number of Miscellaneous Transients (10), assumed in Table 4-7, sufficient to cover the periodic hydrostatic testing of the reactor vessel?

Provide the maximum cumulative fatigue usage factor and the area of the vessel of which it occurs. Estimate the percentage contribution to this usage factor from each type of the transient cycles listed in Table 4-7.

Recent conclusions, drawn from the results of tests on the irradiated and non-irradiated specimens, indicate that neutron irradiation tends to reduce the low-cycle fatigue life. Did the fatigue analysis of the vessel consider this effect?

Provide the results and method of analysis which show that the reactor vessel is designed to accommodate without failure the transient loading, close to the end of its design life, due to safety injection of cold water up to the level of the main coolant nozzles. Assume maximum deluge rate starting from an empty vessel. State the initial vessel temperature used in the calculations and the assumed failure criterion. Also, estimate the limit of the initial vessel temperature which could cause vessel failure upon injection. Relate this temperature to the maximum delayed injection time before vessel wall temperature could reach the limit.

The description of the neutron flux calculations (PSA, page 3-18) states that the results from 34-group P3MG calculations show that reduction of the flux above 1 Mev by the thermal shield is about a factor of 4 greater than that computed from the 4-group calculations. Was the P3MG analysis performed for the Oconee reactor arrangement? If not, please provide the appropriate reference.

Please clarify the basis for selection of the factor of 2 by which the calculated neutron flux was multiplied to obtain the predicted time-integrated neutron flux on the reactor vessel wall.

State if the material irradiation surveillance program will comply with all the recommendations of the recently revised ASTM Specification E185-6X.

With respect to the neutron irradiation of the reactor vessel, please provide the following additional information:

a. Type of neutron flux monitors, which will be used in the irradiation capsules.
b. Method which will be employed to determine neutron flux of the irradiation samples.
Anticipated difference in neutron spectra at the sample location and the vessel wall.
d. Estimated accuracy of the neutron flux calculation, and the experimental basis for the estimate.

Steam Generator

Discuss the probability that failure of one steam generator tube would cause other tube failures.

Provide the results of a detailed analysis of the consequences of the rupture of a single steam generator tube.

Provide the margin to failure of the steam generator tubes and tube sheets.

Provide the scheduled date for completion of the mechanical vibration tests of the 37-tube model of the once-through steam generator.