50-270

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Docket Nos. 50-270
and 50-287

APR 27 1972

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
Production and Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Our continuing review of your Oconee Units 2 and 3 application indicates the need for additional information. The information required is described in the enclosure.

In order to maintain the review schedule, we need your complete reply by May 25, 1972. Please inform us within seven (7) days after receipt of this letter as to the date when you will be able to submit the requested information to us so that we may revise our schedule if necessary.

Sincerely,

Original Signed By
K. R. Goller
for

Richard C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
Request for Additional Information

cc w/encl:
William L. Porter, Esq.
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

OFFICE ▶	DRL:PWR-4	DRL:PWR-4	DRL:AD/PWRs		
SURNAME ▶	IAPeltier;jkm	ASchwencer	RCDeYoung		
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REQUEST FOR ADDITIONAL INFORMATION

OCONEE NUCLEAR STATION UNITS 2 & 3

DOCKET NOS. 50-270/287

1.0 GENERAL

1.1 Since the 1970 AEC review of this station, there have been a number of changes in the information contained in the FSAR for Oconee 1, 2 and 3. There also have been a number of events which may result in changes to the plant and/or their operation.

Likewise the AEC has issued new or modified old criteria contained in 10 CFR 50 and issued a number of Safety Guides in this time period.

1.1.1 Please provide information regarding the extent to which experience with Oconee Unit 1 will affect the design and operation of Units 2 and 3.

1.1.2 Please provide information regarding the extent to which Units 2 and 3 will be affected by the current Appendix A of 10 CFR 50, General Design Criteria for Nuclear Power Plants.

1.1.3 Please provide information regarding the extent to which Units 2 and 3 will be compatible with the intent of current issued Safety Guides with particular emphasis on Guides 1, 4, 7, 13, 16, 20, 21.

2.0 SITE AND ENVIRONMENT

2.1 We understand that a minimum of one year of onsite meteorological data have been accumulated after the filling of Lake Keowee with water and that wind calibration problems with the data previously presented for the period from June 19, 1968 through June 18, 1969 have been resolved. Provide the joint frequency distribution of wind direction and wind speed by stability class for annual period(s) of record after the filling of Lake Keowee. Use vertical temperature difference measurements in currently acceptable classes to define atmospheric stability.

2.2 Discuss the effect of heated water discharges from the three nuclear units on the diffusion climatology of the site.

2.3 Section 2.1 of the FSAR, indicates that the population statistics are based on the 1960 census. Update this section to include the 1970 census data.

2.4 Figure 2-2 of the FSAR, Plot Plan and Site Boundary, shows the one-mile radius exclusion area boundary. Provide a revised figure which clearly shows the boundary which will be used for establishing effluent release limits. [See the AEC's recently published "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" dated February 1972, Section 2.1.2.2.] Show the nearest location suitable for dairying.

2.5 Figure 2-2 of the FSAR shows a highway passing in a northerly direction through the one-mile radius exclusion area boundary. Provide an analysis of a truck transportation incident involving toxic chemicals, explosive materials, and flammable materials on the [illegible]e operation of the Oconee Nuclear Power Station. If there are any oil or gas pipelines which pass near the reactor facility, the effect of potential accidents on the safe operation of the nuclear facility should be evaluated. If gaseous chlorine is stored onsite, describe the consequences of accidental release of chlorine on the reactor control room personnel. Describe any protective devices used to protect control room personnel during such a postulated incident.

2.6 Describe, if any, all industrial activities within a five mile radius of the site. If industrial facilities are located within a five mile radius of the plant, provide a description of the products manufactured, stored, or transported to indicate the maximum quantities of hazardous material likely to be processed, stored or transported.

3.0 REACTOR

3.1 Describe your plans for vibration monitoring and detection of loose parts in Units 2 and 3.

3.2 Describe your procedures for preventing control rod damage due to dry trip.

4.0 REACTOR COOLANT SYSTEMS

4.1 Describe any design modifications, operational limits and/or additional test procedures that may result from experience gained during hot functional test in Oconee Unit 1.

4.2 Regarding the fracture toughness data obtained for all pressure-retaining ferritic materials of the reactor vessel, state the degree of compliance with the acceptance criteria for the recently revised ASME Code Section III fracture toughness rules (Code Case 1514). These rules require determination of the following for reactor vessel plates, forgings, and qualification welds:

a. NDT temperature obtained from drop weight (DWT) tests, and

b. Temperature, at which "weak" direction Charpy V-notch specimens exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs absorbed energy.

4.3 For the materials of the reactor vessel beltline (including welds) provide the initial upper shelf fracture energy levels, as determined by Charpy V-notch tests, if available, in both directions.

4.4 Provide proposed operating limitations during startup and shutdown of the reactor coolant system using as a gudie Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code Section III fracture toughness rules (Code Case 1514).

4.5 For the predicted NDT temperature shift of 250°F (FSAR, page 4-24) at least five capsules are required by the AEC proposed "Reactor Vessel Material Surveillance Program Requirements," 50.55a, Appendix H, published in the Federal Register on July 3, 1971. Each of these surveillance capsules should include specimens from the base metal, heat-affected zone and the weld metal, as recommended in ASTM E-185, Section 3.3 Section 4.4.6 of the FSAR refers to the report BAW-10006 for the description of the surveillance program consisting of six capsules, only three of which contain weld metal specimens. In effect, the proposed surveillance program consists of only three capsules containing the required number and type of impact specimens.

Describe the steps that will be taken to provide five surveillance capsules, each of which contain specimens per ASTM E-185.

4.6 Regarding preoperational mapping of the reactor vessel by ultrasonic examination, to meet the requirements of IS-232 of Section XI of the ASME Code, state the acceptance standards that were used to establish acceptability of the vessel for service.

5.0 CONTAINMENT SYSTEMS AND OTHER SPECIFIC STRUCTURES

5.1 Section 5 of the FSAR contains procedures and instrumentation for the structural testing of the Unit No. 1 containment. However, the extent to which these procedures and instrumentation will be applied to the testing of Units 2 and 3 cannot be ascertained from the section. Provide this information and, in addition, indicate the extent to which the procedures and instrumentation proposed are compatible with Safety Guide No. 18.

5.2 Reword Section 5.6.2.1 of the FSAR to make it clear (under "Integrated Leak Rate Test") that a leak rate test will be performed at the end of the ten year period also (three tests per 10 year period). Delete Item C under "Integrated Leak Rate Test."

6.0 ENGINEERED SAFETY FEATURES

6.1 Provide analysis to justify that the ECCS and containment spray pumps will have adequate net positive suction head.

6.2 Provide information on how failures in the engineered safety features will be detected during operation.

6.3 Identify any field run piping used for the engineered safety systems and the manner in which such runs, if any, are checked against design predictions.

7.0 INSTRUMENTATION AND CONTROL

7.1 Provide information on the measures being taken to prevent the type of fire which occurred in the control rod drive system transfer panel for Unit 1 on March 9, 1972.

8.0 ELECTRICAL SYSTEMS

8.1 Provide information on the Performance Discharge Tests to be performed in accordance with IEEE Standard 308 - 1970 on the station batteries.

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.1 Provide information to support the fact that the RIA-44 monitoring device to be installed in the unit vents will have the required sensitivity for measuring the anticipated levels either for a continuous or instantaneous release. Discuss iodine plate out.

11.2 Verify that the charcoal to be used in the RIA-44 monitor is impregnated to assure collection of both elemental and non-elemental forms of iodine. Provide information as to the frequency at which the charcoal will be changed and tested.

12.0 CONDUCT OF OPERATION

12.1 Expand the job description of the Shift Supervisor and the Assistant Shift Supervisor to describe their responsibilities during Units 2 and 3 operation (FSAR Section 12.1.2.5 and 12.1.2.6).

12.2 The minimum qualifications listed for the key supervisory positions do not meet the minimum requirements as specified in ANSL N.18.1 - 1971. Justify (FSAR Section 12.2).

12.3 Describe what provisions have been made for the review of both operating and emergency procedures by the operating crew during the life of the plant, to enhance operator preficiency (FSAR Section 12.3).

12.4 State whether agreements with those offsite organizations which will be called upon to perform emergency functions or services are in writing. [FSAR Sections 12.3.2.2 and 12.3.2.3(c)].

12.5 Define what is meant by "periodic" in relation to emergency drills for the training of plant personnel. State at what frequency simulated drills involving offsite agencies will be conducted [FSAR Section 12.3.2.3(e)].

12.6 The doses listed for initiation of protective measures for people in the low population zone are higher than appropriate. We suggest that you consider 2.5R for external exposure and 500 XMPC for internal exposure as possible action levels for notification of offsite support groups (FSAR Section 12.3.2.7.2).

12.7 Define "periodically" (1) regarding the plant operations review by the Station Review Committee and (2) regarding the audit of station operations by the General Office Review Committee. (FSAR Section 12.5).

12.8 The shift complement for the Units 2 and 3 operation is not acceptable. (Figure 12-4B FSAR). (See the letter from P. A. Morris to Duke Power Company dated February 13, 1970).

12.9 Define the role of the Staff Engineer (see page 12A-6 FSAR) in the station organization (Figure 12-4B FSAR).

12.10 State how many nuclear engineers and mechanical engineers with nuclear training and experience are employed in the Mechanical Engineering Section at the present time. (FSAR Section 12A-6).

12.11 Provide resumes for all individuals selected for the positions of Shift Supervisor and Assistant Shift Supervisor.