UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 15, 1970

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Attention: Mr. Austin C. Thies
Vice President
Production & Operation

Gentlemen:

In our continuing review of your application for Provisional Operating Licenses for the Oconee Nuclear Station, Units 1, 2, and 3, we need the additional information described in the enclosure.

Please contact us if you desire any discussion or clarification of the material requested by this letter.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

Distribution:

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PWR-2/DRL	PWR-2/DRL	PWRs/DRL	DRL	DRL
ASchwencer:pt	CGLong	RCDeYoung	FSchroeder	PAMorris
4/14/70	4/14/70	4/14/70	4/13/70	4/15/70

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REQUEST FOR ADDITIONAL INFORMATION

14.5 *REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT

Provide a qualitative description of the transients caused by a reactor coolant pump locked rotor for each of six possible combinations (i.e., 1 case for 4-pump operation, 2 cases for 3-pump operation, 2 cases for 2-pump operation, and 1 case for 1-pump operation).

For the worst of the above cases, provide the results of calculations of reactor core and coolant leg flows, power, primary system pressure, fuel and clad temperatures, and DNB ratios. Describe the computational procedure and show that conservative assumptions were used for moderator temperature coefficient, initial power, initial temperature, initial pressure, minimum shutdown margin with a stuck rod, hot channel factors, core heat transfer, gap conductance, steam generator heat transfer, and pressurizer response.

14.6 REACTOR COOLANT PUMP SHEARED SHAFT ACCIDENT

Provide the same information on the sheared-shaft-accident as requested for the locked-rotor-accident above.

14.7 OPERATION WITH LESS THAN FOUR REACTOR COOLANT PUMPS RUNNING

a. Calculate and discuss the flows and temperatures for the reactor core, the two steam generators, and the six primary coolant legs for these modes of partial loop operation: three pumps, two pumps in one loop, one pump in each loop, and one-pump operation. Include subcases corresponding to isolation or nonisolation of one steam generator.

b. Describe the measurements that will be made during the startup program to verify these flows and temperatures.

c. Describe your evaluation of accidents and operational transients which might be initiated during partial-loop operation, especially during single-loop operation.

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* Requests 14.1 through 14.4 were made by letters of February 13 and March 3, 1970.

d. For each mode of partial-loop operation, evaluate the potential for cooling of the loops by the once-through steam generator system. Provide a discussion of the operation of the integrated control system for each mode of partial-loop operation and each mode of control: automatic, manual, load tracking, and startup.

e. For each mode of partial-loop operation, discuss the potential for cold water transients resulting from inadvertent startup of an inactive pump or pumps. Provide an analysis of the consequences of the worst case. Make conservative assumptions such as instantaneous acceleration to full pump flow, most negative moderator temperature coefficient, minimum 1% hot shutdown reactivity margin, minimum stagnant loop temperature, and high initial pressurizer level. Describe the calculational method and give values of all input parameters.

14.8 RESTART OF A TRIPPED PUMP

a. Provide an analysis of the worst cold water transient which could result if subsequent to the tripping of a coolant pump, operator and integrated control system actions reduced power and restarted the tripped pump. Make conservative assumptions, especially for secondary side flows and heat transfer.

b. Describe the measurements to be made during the startup program to verify the system behavior and consequences of this transient.

14.9 STARTUP ACCIDENT

For the maximum reactivity ramp insertion rate which is slow enough to cause a high pressure reactor trip before a high neutron flux level trip [about 2×10^{-4} $(\Delta k/k)$/sec, see Figure 14-3], provide curves of pressurizer level and pressure versus time and compare the maximum expansion rate of the primary system with the relief capacities of the pressurizer safety valves.