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Docket Nos: 50-390
50-391

Mr. N. B. Hughes
Manager of Power
Tennessee Valley Authority
830 Power Building
Chattanooga, Tennessee 37401

Dear Mr. Hughes:

SUBJECT: REQUESTS FOR INFORMATION AND POSITIONS ON WATTS BAR

Enclosed are requests for information and staff positions on the Watts Bar thermal-hydraulic design.

Responses are requested by April 17, 1979 to continue our review.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosure:
As stated

cc: See next page

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OFFICE	DPM:LWR #4	DPM:LWR #4			
SURNAME	CStahle:tl	SVarga			
DATE	2/5/79	2/6/79			

Tennessee Valley Authority

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ccs:
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Mr. David Lambert
Tennessee Valley Authority
303 Power Building
Chattanooga, Tennessee 37401

ENCLOSURE

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221.0 Reactor Analysis Section

221.11 The information presented in Section 4.4 of Watts Bar 1&2 FSAR is not sufficient to demonstrate the required thermal-hydraulic stability of the core.

Watts Bar 1&2 uses the HYDNA code to analyze the flow stability characteristics of the core and references WCAP-7240 (P) and WCAP-7966 to describe the effect of open channel flow on thermal-hydraulic flow instability. However, the referenced reports do not describe the HYDNA code and its use in the referenced reactor calculations. The reports provide experimental data intended to show that simulated fuel assemblies without an enclosing shroud will provide a larger stability margin than would fuel assemblies with an enclosing shroud. While such data are useful as background information, they are not sufficient to support a conclusion that the HYDNA code conservatively predicts the onset of flow instability in the core. To support such a conclusion, either (1) provide a complete description of the HYDNA code and its use in the analysis; or (2) provide a discussion excluding the HYDNA code which supports the contention that the core is thermal-hydraulically stable.

221.12 The response to Question 221.2 discussed Westinghouse experience with regard to crud deposition in operating reactors. The following additional information is required to resolve this issue:

1. A list of all instrumentation available to the operator to detect changes in core flow;

2. A description of the procedures which would be used to detect and quantify changes in core flow;

3. A description of the corrective action which would be taken.

221.13 The response to question 221.10 provides functional design information about the loose parts monitoring system. This information provides an acceptable description of the system. However, no information has been provided relative to the operation of the system. Therefore, provide the following information:

1. A description of how alert levels will be determined;

2. A description of the diagnostic procedures to be used to confirm the presence of a loose part;

3. A description of the precautions to ensure acquisition of quality data;

4. A description of plans for a signature analysis during initial startup testing;

5. A description of your personnel training program.

222.0 System Analysis Section

222.1 The transient analysis methods used to evaluate the various transients and accidents are presented in Section 15. In this regard, we are conducting a review of the LOFTRAN computer program including its experimental verification. The amount of applicable experimental data available at this time is limited, and additional data is needed to complete this review. To obtain this information, you are requested to provide an outline for a series of carefully planned transients tests including a discussion of instrumentation requirements. The type of tests to be performed should include the following:

1. Turbine trip

2. Loss of Feedwater Flow

3. Complete loss of primary system flow

4. Partial loss of primary system flow

Test results from similar facilities can be applied when applicable. The transients tests can be performed at less than full power. However, the tests should be sufficiently severe to provide a reasonable test of the analysis methods.

222.2 Describe any difference between the steam line break analysis methods and results reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases", with those reported in the Watts Bar Nuclear Plant FSAR.

222.3 Describe any difference between the feedline break analysis methods and results reported in WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture", with those reported in present FSAR.

222.4 Describe in detail the methods used to analyze the consequences of the steam generator tube rupture accident including the following information:

a.) Details of the computer programs used to calculate the primary and secondary system pressures. Include the nodalization diagram and major assumptions used in the calculations.

b.) Describe the model for calculating the primary to secondary tube leakage. Provide plots for primary system pressure and pressurizer level.

c.) Range of failed tubes that can be considered in this calculations.

222.5 Describe in detail the calculational model used to evaluate the "Complete Loss of Forced Reactor Coolant Flow" transient. In this description include:

a.) All the computer programs used and their interaction with each other.

b.) Nodalization diagrams for computer programs.

c.) Discuss the major assumptions used in the analysis.