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Docket Nos. 50-390
and 50-391

JAN 5 1972

Tennessee Valley Authority
Attn: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

The enclosed request for information regarding the Watts Bar application supplements our earlier request to you dated November 23, 1971.

Our current schedule for Watts Bar is based on the assumption that this additional information will be available for our review by February 11, 1972. If you cannot meet this date, please inform us within seven days after receipt of this letter so that we may revise our schedule.

Sincerely,

Original Signed By
R. C. DeYoung

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

Enclosure:
Request for Additional Information

cc:
Mr. Robert H. Marquis
629 New Sprankle Building
Knoxville, Tennessee 37919

OFFICE ▶	DRL:PWR-2	DRL:PWR-2	DRL:AD/PWRs			
SURNAME ▶	PSCheck:bn	CGLong	RCDeYoung			
DATE ▶	1/4/72	1/5/72	1/5/72			

Docket

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

January 5, 1972

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and 50-391

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REQUEST FOR ADDITIONAL INFORMATION

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNITS 1 & 2

DOCKET NOS. 50-390 AND 50-391

1.0 GENERAL

1.1 Update Appendix D to the PSAR to reflect the Commission's General Design Criteria as published in the Federal Register on July 7, 1971.

4.0 REACTOR COOLANT SYSTEM

4.1 To evaluate the adequacy of the proposed heatup and cooldown limits for this plant, provide the following information:

4.1.1 For all pressure-retaining ferritic components of the reactor coolant pressure boundary whose lowest pressurization temperature* will be below 250°F, provide the material toughness properties (Charpy V-notch impact test curves and drop weight test (DWT) NDT temperature, or others) that will be reported or specified for plates, forgings, piping, and weld material. Specifically, for each component provide the following data for materials (plates, pipes, forgings, castings, welds) to be used in the construction of the component or your estimates based on the available data: (a) the highest of the NDT temperatures obtained from DWT, (b) the highest of the temperatures corresponding to the 50-ft-lb value of the C_v fracture energy, and (c) the lowest of the upper shelf C_v energy values for the weak direction (WR direction in plates) of the material.

4.1.2 Identify the location and the type of the material (plate, forging, weld, etc.) in each component for which the data listed above will be obtained. Where these fracture toughness parameters occur in more than one plate, forging, or weld provide the information requested in 4.1.1 (a), (b), and (c) for each of them.

4.1.3 Provide the information requested in 4.1.1 and 4.1.2 for reactor beltline materials, including welds, and in addition specify (a) the highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the weak direction of the material (WR direction in plates) and (b) the minimum upper shelf energy value which will be acceptable for continued reactor operation toward the end-of-service life of the vessel.

*Lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25% of the normal operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr., under normal operation, system hydrostatic tests, or transient conditions.

4.1.4 Furnish the proposed heatup and cooldown curves which will be used to control the pressures and temperatures to which the ferritic material of the reactor coolant pressure boundary will be exposed during operation of the plant until the scheduled removal of the first material capsule.

4.2 Describe the plans which will be followed to avoid partial or local severe sensitization of austenitic stainless steel during heat treatments and welding operations for core structural load bearing members and component parts of the reactor coolant pressure boundary. Describe welding methods, heat input, and the quality controls that will be employed in welding austenitic stainless steel components.

4.3 If nitrogen will be added to stainless steel types 304 or 316 to enhance its strength (as permitted by ASME Code Case 1423 and USAS Case 71), provide justification that such material may not be susceptible to stress corrosion cracking under severely sensitized conditions.

4.4 Describe the reactor vessel material surveillance program to indicate the degree of compliance with the "Reactor Vessel Material Surveillance Program Requirements" 50.55a, Appendix H published in the Federal Register on July 15, 1971, especially with regard to the requirements on retention of representative test stock (archive material) and chemical composition.

4.5 State the number of Charpy V-notch specimens oriented with respect to the weak direction (WR orientation in plates) of plates, forgings, and weld materials that will be included in the reactor vessel material surveillance program.

4.6 If the process of electroslag welding will be used in the fabrication of components within the reactor coolant boundary, describe the process variables and the quality control procedures to be applied to achieve the desired material properties in the base metals, heat affected zones, and welds.

4.7 With respect to reactor coolant pump flywheels, describe the extent to which you conform or fail to conform to the regulatory position presented in Safety Guide 14.

4.8 Provide the following information regarding the reactor coolant leakage detection system:

4.8.1 Describe the methods to be used to provide positive indications in the control room of leakage of coolant from the reactor coolant system to the containment.

4.8.2 Discuss the adequacy of the leak detection subsystem which depends on reactor coolant activity for detection of changes in leakage during the initial period of plant operation when the coolant activity may be low.

4.8.3 With reference to the proposed maximum allowable leakage rate from unidentified sources in the reactor coolant pressure boundary, furnish the following information:

(a) The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

(b) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

(c) The mathematical model and data used in such analyses.

4.8.4 Specify the proposed maximum allowable total leakage rate for the reactor coolant pressure boundary, and the basis for the proposed limit. Furnish the ratio of the proposed limit to:

(a) The normal capacity of the reactor coolant makeup system.

(b) The normal capacity of the containment water removal system.

4.8.5 Provide the sensitivity (in gpm) and the response time of each leak detection system. For the containment air activity monitors, provide the sensitivity and the response time as a function of the percentage of failed fuel rods or of the corrosion product activity in the reactor coolant, as applicable.

4.8.6 Estimate the anticipated normal total leakage rates and major leakage sources on the basis of operational experience from other plants of similar design.

4.8.7 Describe the adequacy of the proposed leakage detection systems to differentiate between identified and unidentified leaks from components within the primary reactor containment and indicate which of these systems provide a means for locating the general area of a leak.

4.8.8 Describe the proposed tests to demonstrate sensitivities and operability of the leakage detection systems.

4.9 Section XI of the ASME Boiler and Pressure Vessel Code recognizes the problems of examining radioactive areas where access by personnel will be impractical, and provisions are incorporated in the rules for the examination of such areas by remote means. In some cases the equipment to be used to perform such examination is under development. Provide the following information with respect to your proposed inspection program:

4.9.1 Describe the equipment that will be used, or is under development for use, in performing the reactor vessel and nozzle inservice inspections.

4.9.2 Describe the system to be used to record and compare the data from the baseline inspection with the data that will be obtained from subsequent inservice inspections.

4.9.3 Describe the procedures to be followed to coordinate the development of the remote inservice inspection equipment with the access provisions for inservice inspection afforded by the plant design.

4.10 Describe the proposed inservice inspection program for fluid systems other than those composing the reactor coolant pressure boundary, including items to be inspected, accessibility requirements, and the frequency and types of inspection. The fluid systems to be considered are applicable engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems.

5.0 CONTAINMENT

5.18 Provide the following information with respect to the containment leakage testing program:

5.18.1 Indicate the degree of compliance with the AEC proposed "Reactor Containment Leakage Testing for Water Cooled Power Reactors," §59.54(o), Appendix J, published in the Federal Register on August 27, 1971.

5.18.2 Describe the design features of the containment airlocks that will permit testing of the airlocks at the calculated peak containment pressure. Describe the test method that will be used to verify leak tighness of airlock doors, door penetrations, and door gaskets.

5.18.3 Describe the instrument system including type and location of pressure, temperature, and dew point sensors in the containment during the integrated leak rate test when the ice condensers are filled. Discuss the possible measurement errors and show the effect of the most adverse combination of measurement errors on leakage rates measured with such a low containment test pressure.

5.19 Describe the seal between the auxiliary building and the control building. Provide sketches of the details and state the criteria governing the design of the seal with regard to differential movements (both static and dynamic), pressure capability of the seal, and the effects of the environment on the seal.

5.20 Provide the details of the manner by which the equipment hatch and the personnel lock barrels are attached to the shield building. State the design criteria which ensure that the attachment will withstand seismic and other forces.

7.0 PROTECTION AND EMERGENCY POWER SYSTEM

7.1 Provide the information requested in the Commission's Information Guide 2, dated October 27, 1971, and include:

7.1.1 In your response to Question 1.c of Information Guide 2, describe the degree of conformance of your protection system to the provisions of IEE Standard 338-1971 "IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems." Describe and justify any exceptions to these Criteria.

7.1.2 In your response to Request 3 of Information Guide 2, describe the degree of conformance of your seismic testing program to IEEE Standard 344-1971 "IEEE Guide for Seismic Qualifications of Class I Electric Equipment for Nuclear Power Generating Stations." Describe and justify any exceptions to this standard.

7.1.3 In your response to Request 7 of Information Guide 2, describe the degree of conformance of your environmental testing program for continuous-duty motors to IEEE Standard 334-1971 "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations." Describe and justify any exceptions to this Standard.

7.1.4 In your response to Request 12 of Information Guide 2, refer to IEEE Standard 308-1971 in lieu of IEEE as stated in the Guide. (Note: Where a conflict exists between the "eight hour" provision of Section 5.2.3.4 of IEEE Standard 308-1971 and General Design Criterion 17, the applicable provisions of Criterion 17 govern.)

7.2 We understand that placing one reactor trip logic channel in the test mode and concurrently closing the trip bypass breaker in the opposite logic channel will defeat the reactor trip system. What interlocks or other automatic devices will be included in the design to preclude the postulated condition?

7.3 Confirm that the instrumentation associated with the Emergency Gas Treatment System is designed to conform to IEEE Standard 279-1971.

7.4 Will you conform to the provisions of IEEE Standard 336-1971 "IEEE Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations" during the construction phase of the plant? Describe and justify any exceptions to this standard.

7.5 Does the design of the fuel oil transfer system for the diesel generators satisfy the single failure criterion?

7.6 Provide the results of your grid stability analyses relating to the sudden loss of (a) one of the nuclear units and (b) the largest unit on the grid. Assume (a) and (b) are not concurrent.

12.0 CONDUCT OF OPERATIONS

12.1 With respect to corporate technical support, identify by resume the Engineer-in-Charge as defined in ANSI N18.1. Identify the key staff specialists supporting the Engineer-in-Charge.

12.2 Qualification requirements for the Shift Engineer and the Health Physicist are at variance with ANSI N18.1 requirements. You are required to conform with this standard.

12.3 Provide qualification requirements for other plant positions (e.g., plant chemist, reactor engineer, instrumentation and control supervisor, operators, technicians and repairmen).

12.4 Indicate on the proposed Plant Organization Chart (Figure 12.2-4 of the PSAR) which positions will be licensed as Senior Reactor Operator and which as Reactor Operator.

12.5 Insofar as practicable, provide resumes outlining the qualifications of the initial appointees to plant managerial and supervisory positions. Resumes should include formal education, training and experience of prospective incumbents.

12.6 Section A.4.6 of Appendix A to the PSAR indicates the plan to use the Safety Review Board (Independent Review and Audit Group as defined in Proposed Standard ANS-3.2) during the design and construction of the facility as part of the overall quality assurance program. Provide complete information concerning this group that will indicate conformance with the requirements of Proposed Standard ANS-3.2 "Standard for Administrative Controls for Nuclear Power Plants," Draft No. 6 of 9/8/71. Provide information concerning the interface between the Plant Operations Review Committee (Onsite Review Organization as defined in Proposed Standard ANS-3.2) and the Safety Review Board.

12.7 Section 12.4.5 of the PSAR, Site Emergency Plans Manual, is not considered to be responsive to the requirements of 10 CFR Part 50, Appendix E (II) in that Section 12.4.5 does not include sufficient information to enable us to adequately assess the preliminary plans being made for coping with emergencies. Provide, as a minimum, a description of those aspects of emergency planning that are required in the PSAR under the provisions of 10 CFR Part 50, Appendix E (II).

12.8.0 Provide the following information in the area of Industrial Security *:

12.8.1 An overview of the organization, administration and conduct of the Industrial Security Program.

12.8.2 A description of personnel selection policies (including employee performance and evaluation procedures, and the industrial security training program) to assure that reliable and emotionally stable personnel are selected, trained, and assigned to the plant staff.

12.8.3 A description of plant design and arrangement providing for and enhancing industrial security protection features that reduce the vulnerability of the plant to deliberate acts which may adversely affect the public health and safety. Provide cartographic or other illustrative material directly relating to plant security aspects that will support the adequacy of the advance planning for the security arrangements of the facility.

12.9 Will the nuclear power training program conform to the requirements of ANSI N18.1? Provide information concerning proposed subject matter content of the formal nuclear training program. How will the training program effectiveness be evaluated? Provide a description of general employee training to be provided to all persons regularly employed in the nuclear power plant.

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*Note: It may be desirable to withhold from disclosure information supplied in response to these requests by submitting this information as provided for in Section 2.790 of the Commission's Regulations.