SEP 22 1981

Docket Nos: 50-390/391

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500 A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Request for Additional Information Concerning the Watts Bar Nuclear Plant, Units 1 and 2

Attached are requests for additional information developed as a result of our review of the Final Safety Analysis Report for the Watts Bar Nuclear Plant Units 1 and 2. To expedite the review of your facility, many of these items were forwarded to your staff informally on September 2, 1981.

Below is a list of the subject areas included in this package:

Enclosure	Q Nos.	Subject
1	321.19 - 321.25	Effluent Treatment Systems Branch review questions
2	231.6 - 231.7	Thermal-Hydraulics Section review questions
3	421.2 - 421.3	Quality Assurance Branch review questions
4	212-105 - 212-113	Reactor Systems Branch review questions
5	121.19	Materials Engineering Branch review question

These items may appear in the SER as open items unless they can be answered satisfactorily by your staff prior to October 1, 1981.

Appl 3

OFFICE ▶							
SURNAME ▶							
DATE ▶							

If you have any questions concerning these matters, please contact the project manager, T. J. Kenyon.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As stated

cc: See next page

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WATTS BAR

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ENCLOSURE (1)

REQUEST FOR ADDITIONAL INFORMATION
BASED ON
THE SEQUOYAH-WATTS BAR FACILITY COMPARISON

321.19 (6.5.1) The Engineered Safety Feature (ESF) Filtration Systems at Sequoyah, including the emergency gas treatment system, the auxiliary building gas treatment system, the reactor building purge system, and the main control room emergency air system, have technical specification-surveillance requirements to meet the guidelines of Regulatory Guide 1.52 (Rev. 2), March 1978. The Watts Bar FSAR Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 are based on Regulatory Guide 1.52 (Rev. 0) and should be updated in order to complete the comparison.

321.20 (6.5.1) (9.5) In order to measure flow in the ESF and normal ventilation systems at Sequoyah, several additional flow monitors were required to be installed by the technical specifications. The comparison should address the addition of flow monitors to meet the guidelines of Regulatory Guides 1.52 (Rev. 2) and 1.140.

321.21 (3.2.2) (11.0) As in our review of the Sequoyah radwaste management systems, we will compare the seismic design and classification to the guidelines in Regulatory Guide 1.143. Does the TVA Class D Designation apply to the nine gas storage tanks in the gaseous waste process system? List the components of the liquid, gaseous, and solid radwaste systems that do not meet the positions in Regulatory Guide 1.143 and explain the difference in "limited" seismic design TVA Class G and K at the Watts Bar and Sequoyah plants.

321.22 (11.3) The hydrogen and oxygen analyzers on the gaseous waste processing system were restricted by technical specifications at the Sequoyah plant. Will the design at Watts Bar include the special automatic control features recommended in SRP 11.3 or will it be manually controlled and therefore limited by the technical specifications as the Sequoyah system?

321.23 (11.4) (11.5) The FSAR figures 11.4-1 to 11.4-6 refer to the solid radwaste system in Section 11.5. Clarify the figure numbers and provide the radiation monitoring instrumentation P&IDs.

321.24 (11.4) The Tables 11.4-3 and 11.4-4 of the Watts Bar FSAR do not agree with specifications 3.11.1.1 and 3.11.2.1 of the Sequoyah technical specifications. Were these tables to be updated?

321.25 (11.5) In order to complete our review of the solid radwaste systems at the Watts Bar plant, we need a description of the reduction and solidification (NRS) system since the design will be different from Sequoyah's system. Provide a schedulefor submitting this information.

ENCLOSURE (2)

REQUEST FOR ADDITIONAL INFORMATION

231.6 Standard format and content of Safety Analysis Reports, Regulatory Guide 1.70, states that in Chapter 4 of the SAR

> "...the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operation modes..."

Are the analyses presented in Section 4.4 representative of the initial core only or have future cycles been analyzed? Provide a discussion of how power distributions for future cycles are considered in the FSAR analyses. Is there any assurance that the Watts Bar Units can operate at the licensed power level without excessive DNB trips throughout future cycles? Will revisions to the design methodology be required in order to maintain sufficient thermal margin?

231.7 In response to Question 221.13 the applicant provided a description of the Loose Parts Monitoring System (LPMS) which would be installed in the Watts Bar Units. The applicant stated that the training program for the operators of the LPMS will be the same as that for Sequoyah. Before the staff can determine the acceptability of the Watts Bar LPMS, we will require that the applicant submit a detailed description of the operator training program.

Enclosure (3)

Request for Additional Information

421.2 Augment Chapter 17 of the Watts Bar FSAR with a list of items which will be subject to the pertinent requirements of the FSAR operational quality assurance program.

421.3 The list of items provided from the Watts Bar OQAM needs to be supplemented in Chapter 17 of the FSAR in accordance with the following:

a. The following items do not appear on the list. Add the appropriate items to the list in Chapter 17 of the FSAR and provide a commitment in Chapter 17 of the FSAR that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

1. Spent fuel pool and liner.
2. Fuel handling machine.
3. Spent fuel assembly handling tool.
4. Reactor vessel head lifting rig.
5. Radioactivity monitoring (fixed and portable).
6. Radioactivity sampling (air, surfaces, liquids).
7. Radioactive contamination measurement and analysis.
8. Personnel monitoring internal (e.g., whole body counter) and external (e.g., TLD system).
9. Instrument storage, calibration, and maintenance.
10. Decontamination (facilities, personnel, and equipment).
11. Respiratory protection, including testing.
12. Contamination control.
13. Measuring and test equipment used for CSSC.
14. Masonry walls in the control and auxiliary buildings.
15. PORV block valves and actuators.
16. EGTS oxygen monitors.
17. Accident-related meteorological data collection equipment.

18. Expendable and consumable items necessary for the functional performance of safety-related structures, systems, and components (i.e., weld rod, fuel oil, boric acid, snubber oil, etc.).

19. Intake channel slopes.

20. Missile protection slabs and backfill.

b. The following items from Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," (November 1980) do not appear on the list. Add the appropriate items to the list in Chapter 17 of the FSAR and provide a commitment in Chapter 17 of the FSAR that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

		NUREG-0737 Enclosure 2 Clarification Item
1.	Plant-safety-parameter display console.	I.D.2
2.	Reactor coolant system vents.	II.B.1
3.	Plant shielding.	II.B.2
4.	Post accident sampling capabilities.	II.B.3
5.	Accident monitoring instrumentation.	II.F.1
6.	Instrumentation for detection of inadeque core-cooling.	II.F.2
7.	Automatic PORV isolation.	II.K.3(1)
8.	Automatic trip of reactor coolant pumps.	II.K.3(5)
9.	Emergency plans (and related equipment).	III.A.1.1/III.A.2
10.	Equipment and other items associated with the emergency support facilities.	III.A.1.2
11.	Inplant I_2 radiation monitoring.	III.D.3.3

ENCLOSURE (4)

REQUEST FOR ADDITIONAL INFORMATION

212.105 (6.3) Questions 212.96 and 212.26 address valve #8813 in the miniflow header common to the HPI pumps. The applicant has provided a power removal inhibit function in the control room to assure against mispositioning of this valve by an active single failure, but has not responded to question 212.96 which requires provision to verify the flow path through the valve during normal operation (assurance against passive failures). We require that this issue be resolved prior to operation of Watts Bar.

212.106 (9.2.7) (212.37) (212.87) An analysis of NPSH for RHR pumps in the recirculation mode of ECCS operation, dated April, 1979, and the analysis summary presented in FSAR Section 9.2.7.1 (Amendment 35) are inconsistent in results and calculational assumptions. The 4/79 analysis employed techniques subsequently approved for Sequoyah; however, inputs for the Sequoyah analyses had to be corrected in order to make them acceptable and the resultant calculated NPSH margin reduced from 31.3 ft. to 2.8 ft. Explain the discrepancy between Watts Bar documentations (4/79 Analyses, and Amendment 35) and, in tabular form, list input assumptions (LPI flow, worst case, etc), criteria (NPSH required at given flow, test rate "as installed" LPI must meet, etc), and summary of calculated NPSH available (specify term by term).

212.107 (6.3) (9.2.7) (212.39) Table 6.3-3a lists times, volumes, etc, associated with ECCS switchover from injection to recirculation and provides a basis for RWST sizing. This table does not include instrument uncertainty (assume ± 3% of full instrument space unless documented justification of another value is provided and approved), nor does it consider single failures in switchover. Justify the RWST volume, setpoints, and alarms, addressing the above consideration. The justification should identify all functional requirements for the RWST (e.g., adequate injection volume to assure NPSH for recirculation, sufficient remaining volume to complete switchover, sufficient volume to accommodate possible containment spray requirements, a maximum volume greater than that for which excessive containment post-accident flooding levels might result, etc), and consider instrument uncertanity, unusable volume, and single failures.

212.108 (6.3) (212.36) In response to our past requests to perform preoperational tests to show conformance with Regulatory Guide 1.79 (sump testing) the applicant has referenced FSAR Table 14.2-1 (W3.3) and cited that scale model test would be performed. It is our understanding that scale model test program originally proposed for Watts Bar and Sequoyah was expanded to address plant specific concerns for Sequoyah, and that additional documentation had not been provided to justify the specific tests and the added considerations for Watts Bar. It is our additional understanding that the Watts Bar sump is not in the same containment location as are the Sequoyah sumps and that extended plant specific documentation provided to resolve concerns for Sequoyah may not apply to Watts Bar. We require that the applicant resolve this issue prior to power operation of the Watts Bar facility.

212.109 (212.100) (6.3.3) In question 212.100 we requested that the applicant analyze and discuss RHR pipe break events. We await his response to this question which must be resolved prior to power operation of Watts Bar.

212.110 (6.3.3) (200.4) (212.49) In previous questions 200.4 (August 5, 1976) and 212.49 (April 19, 1977) and in our status summary dated March 21, 1979, we have requested that the applicant provide LOCA analyses to show conformance with the requirements of 10CFR 50.46(b) and Appendix K. To date these analyses have not been provided. This issue must be resolved prior to plant operation.

212.111 With regard to the potential for debris impairing containment sump/ECCS
(6.3) performance, four considerations should be addressed;

1) containment design and potential sources of sump debris (e.g. insulation, etc.)
2) maintenance of containment "as licensed" cleanliness (free of loose debris),
3) instrumentation and alarms to alert the operator to a degradation in post-LOCA ECCS performance,
4) procedures to be taken in the event of post-LOCA ECCS performance degradation.

For item #1, a detailed survey of insulation and other potential debris sources is required; however, this item may be resolved by providing a general summary of the insulation used at Watts Bar, an assessment by the applicant of its potential to produce post-LOCA debris, and a commitment to provide a detailed survey prior to startup following the first refueling outage.

The following questions are provided as guidance into the nature and detail of the type of response expected to resolve the above concerns.

For items that have been previously resolved, you may respond by referring to the previous documentation. Items 1, 2, and 3 must be resolved prior to full power operation; item 4 must be resolved prior to startup following the 1st refueling.

1. In addition to insulation debris resulting from LOCA effects, debris can be generated within the containment from other sources, such as (1) degraded materials (paint chips), and (2) items which are taken into and left in the containment following maintenance and inspection activities.

Describe how the housekeeping program for Watts Bar will control and limit debris accumulation from these sources. The objective is to assure that debris capable of defeating the post-LOCA core cooling functions are identified and removed from the containment. The response should include references to specific procedures or other means to assure that "as licensed" cleanliness will be attained prior to initial operation and prior to each resumption of operation.

2. Address the degree of compliance of Watts Bar with the following recommendation which is also set forth as item C.14 of Regulatory Guide 1.82:

"Inservice inspection requirements for coolant pump components (trash racks, screens, and pump suction inlets) should include the following:

a. Coolant sump components should be inspected during every refueling period downtime, and

b. The inspection should be a visual examination of the components for evidence of structural distress or corrosion."

3. The resolution of the concerns noted below plus the provisions of adequate NPSH under non-debris conditions, and adequate housekeeping practices are expected to reduce the likelihood of problems during recirculation. However, in the event that LPI recirculation system problems such as pump cavitation or air entrainment to occur, the operator should have the capability to recognize and contend with the problems.

Both cavitation and air entrainment could be expected to cause pump vibration and oscillations in system flow rate and pressure. Show that the operator will be provided with sufficient instrumentation and appropriate indications to allow and enable detection of these problems. List the instrumentation available giving both the location of the sensor and the readout.

The incidence of cavitation, air entrainment or vortex formation could be reduced by reducing the system flow rate. The operator should have the capability to perform indicated actions (e.g., throttling or terminating flow, resort to alternate cooling system, etc). Show that the emergency operating instructions and the operator training consider the need to monitor the long-term performance of the recirculation system and consider the need for corrective actions to alleviate problems.

4. With regard to the sump tests on Watts Bar, the responses to the following concerns pertaining to potential sump screen blockage are required:

   a. Various types of insulation may be used in the containment. For each type provide the following information:

      (1) The manufacturer, brand name, volume and area covered.

      (2) A brief description of the material and an estimate of the tendency of this material either to form particles small enough to pass through the fine screen in the sump or to block the sump trash racks or sump screens.

      (3) Location of the material (metal mirrored, foam glass, foam rubber foam concrete, fiberglass, etc.) with respect to whether a mechanism exists for the material to be transported to the sump.

   b. Provide an estimate of the amount of debris that the sump inlet screens may be subjected to during a loss-of-coolant accident. Describe the origin of the debris and design features of the containment sump and equipment which would preclude the screens becoming blocked or the sump plugged by debris. Your discussion should include consideration of at least the following sources of possible debris: equipment insulation, sand plug materials, reactor cavity annulus sand tanks or sand bags for biological shielding, containment loose insulation, and debris which could be generated by failure of non-safety related equipment within the containment. Entry of sand plug materials into the containment sump and the possibility of sand covering the recirculation line inlets prior to the initiation of recirculation flow from the containment should be specifically addressed.

Please provide this information along with your conclusion regarding the percentage of the screens which would be expected to be blocked by particles of all sizes, including those greater than 250 mils.

c. With respect to the conclusion that debris with a specific gravity greater than unity will settle before reaching the sump cover, consider the potential for flow paths which may direct significant quantities of debris laden coolant into the lower containment in the vicinity of the sump and the availability or lack of sufficient horizontal surface areas or obstructions to promote settlings or holdup of debris prior to reaching the sump.

d. Does metal mirror insulation house other materials, fibrous or otherwise, which could become debris if the insulation were blown off as a result of a LOCA?

e. If the Watts Bar containment contains loose insulation, include examples of how the insulation will be precluded from reaching the sump.

f. Provide a schematic drawing of the post-LOCA water level in containment during the recirculation mode relative to the elevation of the ECCS sump floor. Include on this drawing the location of the containment water level sensor and the elevations corresponding to readings of zero and 100 percent of range on the control room indicator.

g. Provide several large scale drawings of the containment structures, systems and components at elevations.

h. Does the Watts Bar utilize sand or similar materials in the containmen during power operation for purposes such as reactor cavity annulus biological shielding (e.g., sand tanks or sand bags) or reactor cavity blow out sand plugs?

212.112 (6.3) Recently the concern has been raised that, for certain PWRs without appropriate design provision, after very small break LOCAs with initiation of ECCS, the RCS pressure could remain at or near the primary system safety valve setpoint and prevent sufficient charging pump flow to avoid damage to those pumps. Address this concern for Watts Bar and justify that such an event would not cause damage to the pumps and/or prevent adequate core cooling.

212.113 (5.1) (6.3) Recently, concern has been raised about control of the charging pump suction during normal operation for PWRs. This concern postulates that a plant normally takes suction from the volume control tank (VCT), and that when high level is sensed in the VCT, letdown flow (from the RCS) is diverted to a hold up tank (instead of the VCT); whereas, if low level is sensed in the VCT, charging pump suction is switched to the refueling water storage tank as the source. The concern is related to the level indication which is only control grade, and the consequences if this level instrumentation should fail giving a false "high" indication. This could lead to damage of the charging pumps (due to loss of suction inventory and failure to make the protective suction switch) and thereby compromise both the capability to shut down and ECCS capability to respond to the full range of accidents. Discuss the status of the Watts Bar design with respect to this issue.

ENCLOSURE (5)

REQUEST FOR ADDITIONAL INFORMATION

121.19 In FSAR Section 5.2.3.2. (1st paragraph, 1st sentence), it states that "all of the ferritic low alloy and carbon steels which are used in principal pressure retaining boundaries are provided with a corrosion resistant cladding on all surfaces that are exposed to the reactor coolant." A definition of "principal" is not provided, and it infers there are non-principal pressure retaining boundaries of carbon steel in contact with reactor coolant without corrosion resistant cladding.

Are there low alloy and/or carbon steels in the reactor coolant boundary exposed to reactor coolant which are not provided with a corrosion resistant cladding equivalent to those stated in Section 5.2.3.2? If there are such components, please identify, state their function, and address the engineering approach (e.g. inservice inspection) applied to assure satisfactory performance for the life of the plant.