Distribution:
Docket
AEC PDR
Local PDR
DRL Reading
DR Reading
~~XXXXX~~ PWR-2
C. K. Beck
E. J. Bloch
M. M. Mann
S. Hanauer
F. Schroeder
R. S. Boyd
R. C. DeYoung
D. Skovholt
H. Denton
T. R. Wilson
E. G. Case, DRS
R. R. Maccary
R. W. Klecker
DRL/DRS Branch Chiefs
F. W. Karas (2)
J. Scinto
ACRS (16)
N. M. Newmark
P. S. Check

Docket Nos. 50-390
and 50-391

JAN 07 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 requests to you dated November 23, 1971 and January 5, 1972, and completes our first-round review of the PSAR.

Our current schedule for Watts Bar is based on the assumption this additional information will be available to us by March 31, 1972. If you cannot meet this date, please inform us within seven days after receipt of this letter so we may revise our schedule.

Sincerely,

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 ▶	PWR-2	PWR-2	DRL:AD/PWRS			
SURNAME ▶	PSCheck:bn	CGLong	RCDeYoung			
DATE ▶	1/6/72	1/6/72	1/7/72			

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

Docket Nos. 50-390
and 50-391

January 7, 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 requests to you dated November 23, 1971 and January 5, 1972, and completes our first-round review of the PSAR.

Our current schedule for Watts Bar is based on the assumption this additional information will be available to us by March 31, 1972. If you cannot meet this date, please inform us within seven days after receipt of this letter so we may revise our schedule.

Sincerely,

Daniel R Muller for

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

REQUEST FOR ADDITIONAL INFORMATION

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNITS 1 & 2

DOCKET NOS. 50-390 AND 50-391

2.0 SITE AND ENVIRONMENT

2.19 Provide information on the slopes of the intake canal southeast of the intake pumping station, as shown in Fig. 2.2-4 of the PSAR. Indicate whether slopes are in old alluvial terrace materials or in recent alluvium, and indicate the nature of the dynamic analysis that will be carried out to evaluate the dynamic stability and/or liquefaction potential for these slopes when subjected to earthquake excitation.

4.0 REACTOR COOLANT SYSTEM

4.11 The System Quality Group Classifications identified in Section B.3 of the PSAR differ in some areas from the revised system classification scheme developed by the American Nuclear Society (ANS-20) for pressurized water reactors.[1/] The ANS system classification scheme has been reviewed by the regulatory staff and, as applied to the McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 50-370, provides a generally acceptable quality level for each pressure-containing component of (a) those applicable fluid systems relied upon to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary or to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (b) other associated safety related systems.

4.11.1 Unless you intend to apply all the ANS system quality group classifications to your Watts Bar Nuclear Plant, identify the differences and include a discussion which specifies the measures that will be applied to provide equivalency in quality level, as well as the quality assurance programs that will be implemented for such measures.

4.11.2 Delineate on the Piping and Instrumentation Diagrams submitted in the PSAR the system quality group classification boundaries of each system specified in Table B.3-6. The classifications should be noted at all valve locations in each fluid system where the respective classification changes in terms of the AEC Group Classification letters, for example, from A to B, B to C, C to D as well as other combinations or in terms of your corresponding classification notations.

4.12 Revise Table 4-12 of the PSAR which specifies the proposed code requirements for vessels, piping, pumps, and valves within the reactor coolant pressure boundary to indicate your compliance with the rules of 10 CFR Part 50, Section 50.55a, "Codes and Standards." In the event there are cases wherein your design does not now conform with the rules of Section 50.55a, indicate your plan to either provide information, or to conform to other criteria that provide an equivalent degree of protection.

4.13 Provide a list of ASME and ANSI code case interpretations which will be applied to each Class I (seismic) component within the reactor coolant pressure boundary.

1/ Draft document "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" N-18.2 issued November, 1970.

4.14 The list of transients to be used in the design and fatigue analysis of all components within the reactor coolant pressure boundary as specified in Section 4 of the PSAR appears to be incomplete. Specify all design transients and their number of cycles, such as control system or other system malfunction, component malfunctions, transients resulting from any single operator error, seismic events etc., which are contained in the ASME Code required "Design Specifications" for the components of the reactor coolant pressure boundary.

4.15 With regard to the proposed design criteria for component supports Table B.3-5 provides stress and strain limits for the combination of loads due to the design basis earthquake, pipe rupture, and normal operation, which allow plastic deformation of supports for Class I (seismic) components. Indicate the method by which your design approach includes the inelastic strain compatibility in the supports and supported components. The proposed criteria are acceptable only if the method used can be shown to produce results comparable to the results of a combined dynamic system analysis for all systems where the proposed stress and strain limits apply.

4.16 Paragraph I-701.5.4 of the ANSI B31.7 Nuclear Power Piping Code and Paragraph NB-3622.3 of the ASME Section III Nuclear Power Plant Components Code require that piping shall be supported to minimize vibration and that the designer is responsible by observation under startup or initial operating conditions to assure that vibration is within acceptable levels. Submit a description of the vibration operational test program which will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the design of the system. Include those transients introduced in systems other than the reactor coolant pressure boundary that will result in significant vibration response of reactor coolant pressure boundary systems and components.

4.17 For the combination of normal plus pipe rupture plus Design Basis Earthquake loadings some of the proposed primary stress limits applicable to vessels and piping within the reactor coolant pressure boundary exceed the component code stress limits considered appropriate for the faulted operating condition category (i.e., limit analysis and 3 S_m limits for vessels and piping respectively in accordance with the ASME Section III Nuclear Power Plant Component Code). Document your intention to comply with the applicable code limits for "faulted conditions" or present justification for exceeding those limits. If you intend to use the "simplified" analysis of I-705 of the ANSI B31.7 Code or NB 3650 of the ASME Section III Code for design of piping within the reactor coolant pressure boundary, confirm that the stress limits employed for the faulted operating condition categories will not exceed the code limit cited above.

4.18 To facilitate our review of the bases for the pressure relieving capacity of the reactor coolant pressure boundary, submit the "Report on Overpressure Protection" which has been prepared in accordance with the requirements of the ASME Section III Code. If the report is not available, indicate the approximate date for submission. In the event you do not intend to submit the report until either the Operating License review or late in the construction schedule for your plant, provide the bases and analytical approach (e.g., preliminary analyses) being utilized to establish the overpressure relieving capacity required for the reactor coolant pressure boundary.

4.19 The primary stress limit criteria for reactor vessel internal structures (Page 3.1-12 in the PSAR) may not be sufficiently conservative when compared to the criteria of the January 1971 draft of the ASME Code for Core Support Structures. State whether it is your intent to employ core support structure stress limits comparable to those of the draft ASME Code or provide the basis and justification for proposed core support structure stress limit criteria which will result in stress intensity limits higher than those permitted by the draft ASME Code.

4.20 Identify the prototype reactor (i.e., the initial reactor of the same design, size, and configuration) from which test data is applicable in evaluating the design adequacy of the Watts Bar reactors core support structures to sustain flow induced vibration effects. Provide a detailed comparison of the applicable design parameters for the Watts Bar and prototype units verifying that no significant design or fabrication differences exist between the subject reactors which could materially affect the vibrational response characteristics of the reactor internals. Describe the internals vibration assurance program which will be employed for the Watts Bar units including the bases for the test operating conditions and measurements which will be made.

4.21 For the purpose of determining stress limits, pumps and valves within the reactor coolant pressure boundary are classified as either active[1/] or inactive[2/]. Active pumps and valves are required not only to serve a pressure-retaining function, but also to operate reliably in order to perform a design safety function such as safe shutdown of the reactor or mitigation of the consequences of an hypothesized pipe break in the system. Therefore, to assure that active pumps and valves will function as

---

1/ Active pumps and valves are those whose operability is relied upon to perform a safety function (including a reactor shutdown function) during the transients or events considered in the respective operating condition categories.

2/ Inactive pumps and valves are those whose operability (e.g., valve opening, or closure, pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition categories.

designed in the event of a pipe rupture (faulted condition) in the reactor coolant pressure boundary we consider stress limits associated with elastic action,i.e., stresses at or near yield stress, as appropriate in lieu of the code stress limits for the "faulted condition". Provide a list of active pumps and valves as defined above and state whether it is your intention to comply with the limits indicated for active pumps and valves. Justify any exceptions noted in your response.

B.0 DESIGN CRITERIA AND CLASSIFICATIONS

B.1 The proposed seismic design spectra, Figures B.2-1 and B.2-2, show a peak amplification factor of approximately 2.9 for 2% damping with the spectral line for 2% damping effectively returning to the specified ground accelaration value at 0.07 second period. Response spectra derived from natural seismic records show amplification factors of 2.5 to 4.5 for 2% damping (in the period range 0.5 to 0.15 seconds) with the spectral line for 2% damping returning to ground acceleration at approximately 0.33 second period. Provide a more appropriate seismic design basis by developing design spectra for the Watts Bar site which include a more appropriate amplification factor and which reflect the effects of distance between the seismic disturbances and the site on the predominant response periods.

B.2 Discribe the design control measures which will be instituted to assure that adequate seismic input, including any necessary feed-back from structural and system dynamic analyses is specified to vendors of purchased Class I (seismic) components and equipment. Identify the responsible design groups or organizations who will assure the adequacy and validity of the analyses and tests employed by vendors of Class I (seismic) components and equipment. Provide a description of the review procedures to be utilized by each group or organization.

B.3 Describe the testing procedures and analysis used to design Class I (seismic) components, equipment, instrumentation and electrical systems to withstand the specified seismic loading conditions. Include the methods and procedures to be used to consider the frequency spectra and amplitudes calculated to exist at the equipment supports and the criteria to be used to account for possible amplification of seismic floor imput by support frames and instrument racks. Where tests or analyses will not include evaluation of the equipment in the operating mode, describe the basis for assuring that Class I (seismic) plant features will function when subjected to seismic and accident loadings. If vertical and horizontal excitations will be considered separately, describe the criteria employed to assure adequacy of equipment, instrumentation, etc. when subjected to directly combined horizontal and vertical seismic loads.

B.4 With respect to seismic instrumentation, submit a statement of your intent to implement a program such as described in AEC Safety Guide 12, Instrumentation for Earthquakes (March 10, 1971). Submit the basis and justification for elements of the proposed program which differ substantially from Safety Guide 12.

B.5 With respect to Class I (seismic) piping buried or otherwise located outside of the containment structure, describe the seismic design criteria that will be employed to assure that allowable piping and structural stresses will not be exceeded due to differential movement at support points, at containment penetrations and at entry points into other structures.

B.6 With regard to the development of system and equipment seismic design criteria by the time history method:

B.6.1 Provide plots that show a comparison of the smoothed site response spectra and the spectra derived from the earthquake records for all damping values to be used in the time history system analyses. Identify the system period intervals at which the response spectra acceleration values will be calculated and demonstrate that the period interval used is sufficient to produce accurate spectra that do not deviate significantly below the smooth design response spectra.

B.6.2 Provide a description of the measures that were taken to consider the effects on the floor response spectra of expected variations in assumptions made for structural properties, dampings, and soil structure interactions (e.g., peak width and period coordinates).

B.6.3 Justify use of the proposed time history averaging technique by demonstrating that the resulting design spectra do not deviate significantly below the specified smooth design response spectra.

B.7 Because various assumptions are made regarding structure material properties and soil structure interaction, calculated periods of vibration are not exact. Describe the measures that will be taken to assure that the calculated response of Class I (seismic) structures by the normal mode response spectrum method will conservatively reflect the expected variations in the periods of vibration of the structures.

B.8 The use of constant vertical load factors as vertical response loads for the seismic design of all Class I (seismic) structures, systems, components and equipment, in lieu of a multi-mass dynamic analysis and subsystem analysis may not be suitably conservative. Provide a more appropriate seismic design basis by considering:

B.8.1 The possible combined horizontal and vertical amplified response loading for the seismic design of the building and floors.

B.8.2 The possible combined horizontal and vertical amplified response loading for the seismic design of equipment and components, including the effect of the seismic response of the building and floors.

B.8.3 The possible combined horizontal and vertical amplified response loading for the seismic design of piping instrumentation, including the effect of the seismic response of the building, floors, supports, equipment, components, etc.

B.9 With regard to the seismic design procedures for soil supported structures submitted in Appendix B:

B.9.1 Provide a list of all soil supported Class I (seismic) structures.

B.9.2 Identify the depth of soil over bedrock for each structure listed.

B.9.3 Provide justification for modifying the specified site seismic input when analyzing soil-founded structures. This justification should verify that the validity of the soil amplification analyses is not adversely affected by: (1) the accuracy of in situ soil measurements, (2) the affect of slanted soil layers, (3) soil density variations, (4) the lack of existing bedrock records and (5) neglecting vertical responses.

B.10 Describe the procedures which will be used to account for the number of earthquake cycles during one seismic event, and specify the number of loading cycles for which Class I (seismic) systems and components will be designed for this event as determined from the expected duration of the seismic motions or the number of major motion peaks.

B.11 Provide the criteria employed to determine whether equipment is "rigid" or "not rigid" for purposes of seismic analysis, as described on page B.3-12 of the PSAR.

B.12 The use of static loads equivalent to peak of the floor spectrum curve for the seismic design of equipment (Page B.3-12 of the PSAR) may not always be sufficiently conservative. Justify the use of peak spectrum values by demonstrating that the criteria employed will assure that the contribution of all significant dynamic modes of response under seismic excitation will be included in the analyses to be performed.

B.13 Provide the design criteria and analytical procedures applicable to piping that take into account the relative displacements between piping support points, i.e., floors and equipment, at differenct building elevations.