DISTRIBUTION:
Docket Nos. 50-390/391
LB #4 r/f
DEisenhut
EAdensam
TKenyon
MDuncan
SHanauer
RTedesco
RVollmer
TMurley
RHartfield, MPA
OELD
OIE (3)

bcc: TERA
NRC/PDR
L/PDR
NSIC
TIC
ACRS (16)

Docket Nos: 50-390
and 50-391

JAN 22 1982

RECEIVED
JAN 25 1982
US NUCLEAR REGULATORY COMMISSION
DOCUMENT MANAGEMENT BR.
TIDC

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A 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. These requests provide clarification to the concerns discussed in the draft SER sent to you on December 30, 1981.

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

Attachment	Q Nos.	Subject
1	121.20-121.23	Preservice Inspection Program
2	31.150	Instrument and Control Systems
3	001.2	Waste Heat Park
4	212.122-212.132	Reactor Systems

Most of these questions were transmitted to your staff informally in December 1981. Please respond to these requests for information no later than February 12, 1982. If you have any questions concerning these matters, please contact the project manager, T. J. Kenyon, at (301) 492-7266.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

/s/

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

8202100144 820122
PDR ADOCK 05000390
Q PDR

Enclosures:
As stated

cc:
See next page

OFFICE	See next page	DL:LB #4	LA:DL:LB #4	DL:LB #4			
SURNAME		TKenyon/hmc	MDuncan	EAdensam			
DATE		1/ /82	1/22/82	1/22/82			

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E11B33
Knoxville, Tennessee 37902

Mr. W. Luce
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. David Lambert
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. J. F. Cox
Tennessee Valley Authority
400 Commerce Avenue, W10B85C
Knoxville, Tennessee 37902

Resident Inspector/Watts Bar NPS
c/o U.S. Nuclear Regulatory
Commission
Rt. 2 - Box 300
Spring City, Tennessee 37831

Mr. David Ormsby
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

ATTACHMENT 1

REQUEST FOR ADDITIONAL INFORMATION

121.0 Review of the Preservice Inspection Program

121.20 Your PSI Program does not specifically identify the examinations of the Emergency Core Cooling or Containment Heat Removal Systems. Paragraph 50.55a(b)(2)(iv)(A) of 10 CFR Part 50 requires that extent of examination for Residual Heat Removal, Emergency Core Cooling, and Containment Heat Removal Systems be determined by requirements of paragraph IWC-1220, Table IWC-2520, Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 of Section XI of the ASME Code. Paragraph IWC-2411 requires the equivalent of 100% examination of one of the multiple streams of a system which perform the same or redundant function. The control of water chemistry to minimize stress corrosion described in Paragraph IWC-1220(c) of the 1974 Edition and Addenda through the Summer 1975 is not an acceptable basis for exempting components from examination because practical evaluation, review and acceptance standards cannot be defined. To satisfy the inspection requirement of General Design Criteria 36, 39, 42, and 45, the inservice inspection program must include periodic volumetric and/or surface examination of a representative sample of welds in the Residual Heat Removal, Emergency Core Cooling and Containment Heat Removal Systems including components exempted from examination based on the "chemistry control" provisions of paragraph IWC-1220(c).

Discuss the preservice examination of Residual Heat Removal, Emergency Core Cooling, and Containment Heat Removal Systems.

121.21 General Design Criterion 32 requires that the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.

We have recently identified a problem concerning the effectiveness of ultrasonic examination techniques to examine the primary piping sytem. Certain ultrasonic techniques may not be adequate to consistently detect and reliably characterize service-induced flaws during the inservice inspection of thick-wall cast stainless steel components to acceptance standards of Paragraph IWB-3500 of Section XI.

Discuss the technical basis for determination that your preservice ultrasonic examination is capable of detecting and characterizing crack-like indications in the reactor coolant boundary piping.

When using Appendix III of Section XI for inservice examination of either ferritic or austenitic piping welds the following should be incorporated:

A. Any crack-like indication, 20 percent of DAC or greater, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determine the shape, identity, and location of the reflector.

B. The Owner should evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

121.22 To evaluate your compliance with 10 CFR Part 50.55a(g)(2), we will require that all Class 1 and 2 pressure retaining welds that cannot be examined as required by Section XI of the ASME Code be identified with a supporting technical justification.

A. Where relief is requested for pressure retaining welds in the reactor vessel and steam generator shell welds, identify the specific welds that did not receive a 100% preservice ultrasonic examination and estimate the extent of the examination that was performed.

B. Where relief is requested for piping system welds (Examination Category B-J, C-F, and C-G), provide a list of the specific welds that did not receive a complete Section XI preservice examination including a drawing or isometric identification number, system, weld number, and physical configuration, e.g., pipe to nozzle weld, etc. Estimate the extent of the preservice examination that was performed. When the volumetric examination was performed from one side of the weld, discuss whether the entire weld volume and heat affected zone (HAZ) and base metal on the far side of the weld were examined. State the primary reason that a specific examination is impractical, e.g., support or component restricts access, fitting prevents adequate ultrasonic coupling on one side, component to component weld prevents ultrasonic examination, etc.. Indicate any alternative or supplemental examinations performed and methods(s) of fabrication examination.

121.23 To complete our evaluation of the preservice examination of the Units 1 and 2 reactor vessel, we will require the following information:

a. Electronic Gating of the Vessel Inner Surface. Discuss the extent of electronic gating that was used and estimate the weld volume that was not examined. Provide a description of the automated examination including calibration parameters.

b. Vessel Welds that were not Examined. Identify the specific welds that did not receive a 100% preservice ultrasonic examination and estimate the extent of examination on a sketch of the vessel welds. Indicate the welds that received an examination from only one side of the weld. State the primary reason that the specific examination was impractical, e.g., support or component restricts access or the automated scanning tool was not capable of reaching area. Discuss the type and extent of fabrication examination that was performed on the welds where the preservice examination was impractical.

c. Unit 2 Vessel Calibration Blocks. Provide a drawing or sketch of one of the calibration blocks used for the shell welds. Indicate whether the calibration block cladding process and surface finish are identical to the vessel being examined, i.e., the vessel was clad before the nozzle drop-outs were removed. Discuss the technical bases for-concluding that the acoustical properties of the calibration blocks are representative of the vessels being examined.

d. Underclad Cracking Examination. Provide a description of the augmented examination to determine whether underclad cracking was present. Identify the location and extent of crack indications and describe any metallurgical confirmation of the ultrasonic examination results.

REQUEST FOR ADDITIONAL INFORMATION

31.150 In response to Question 31.148, TVA included the response given for the Sequoyah Plant and noted that it is applicable to Watts Bar. We request that you elaborate further to clarify the specifics of Items (1), (2), and (3) with regard to the results for "RCS Inventory and Pressure Control," "Steam Generator Inventory and Pressure Control," and "ECCS Response" as noted on Page 31.148-3 of your response.

Further, we request that you address potential failures of power range neutron detector which could cause rod withdrawal. This item was identified as a potential concern by Westinghouse in previous analysis related to IE Information Notice 79-22.

ATTACHMENT 3

REQUEST FOR ADDITIONAL

INFORMATION

001.2 In a meeting with the TVA staff, the NRC was informed of the applicant's intent to build a waste heat ultilization park at the Watts Bar site. Since that meeting, the NRC staff received the July 31,1981 submittal to NEPA entitled, "Final Environmental Impact Statement - Proposed Watts Bar Waste Heat Park."

The facility is not described in the Watts Bar FSAR and no information regarding the safety impact of such a facility has been presented. Therefore, please describe the proposed waste heat park, including any modifications required to be made on existing systems. In addition, present appropriate analyses of the potential impact of this facility on the Watts Bar Nuclear Plant.

ATTACHMENT 4

REQUEST FOR ADDITIONAL INFORMATION

212.122 (212.106) (5.2.2) The response to Q212.106, dated 9/25/81 does not provide a comparison of Watts Bar parameters with those in WCAP-7769 tables 2-1 and 2-2 to justify applicability of the WCAP to Watts Bar. Of particular interest is "Rates of Safety Valve Flow to Peak Surge Rate." Either provide a comparison of all parameters in tables 2-1 and 2-2, or (if this comparison cannot be made or fails to justify applicability) provide an analysis of the design basis event (turbine trip with loss of main feedwater) not taking credit for the first reactor protection system safety grade trip (high pressurizer pressure).

212.123 (212.107) (5.4.7) The response to Q212.107 dated 9/25/81 references Sequoyah natural circulation startup tests to address the RSB 5-1 natural circulation test requirement. These tests were not performed under conditions satisfying all natural circulation test requirements. Reference and justify other tests to meet this requirement.

212.124 The analyses of a locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3 and GDC 17, we require that this event be analyzed assuming turbine trip and coincident loss of offsite power to the undamaged pumps.

Appropriate delay times may be assumed for loss of offsite power if suitably justified.

Steam generator tube leakage should be assumed at the rates specified in the Technical Specifications.

The event should also be analyzed assuming the worst single failure of a safety system active component. Maximum technical specification primary system activity and steam generator tube leakage should be assumed. The analyses should demonstrate that offsite doses are less than the 10 CFR 100 guidelines values.

212.125
(212.111)
(6.3)
The NPSH analyses provided on 10/28/81 do not sum properly. Correct the apparent arithmetic error and provide a corrected NPSH.

212.126
(212.111)
(6.3)
NPSH, RWST sizing, and Watts Bar sump tests are based on the post-LOCA containment flooding level assumption. During the Sequoyah review, it was discovered that crane wall penetrations were not properly sealed to allow flooding to the assumed elevation and modifications were necessary to provide a proper seal. Verify that Watts Bar penetrations are properly sealed.

212.127
(212.111)
(6.3)
NPSH calculations for Watts Bar assume an LPI flow rate of 4500 gpm per LPI pump. Sequoyah NPSH analyses assumed 5500 gpm, and in the review of Salem, system modifications were necessary to meet NPSH analysis assumptions. Justify that the as-built Watts Bar LPI pump flows will not exceed 4500 gpm (this value must be verified in preoperationaltests).

212.128
(212.115)
The response (8/26/81) to our request for LOCA analyses did not provide or cite sensitivity studies to justify selection of the worst case.

212.129
(212.116)
(6.3)
The applicant should discuss housekeeping procedures to ensure that the containment sump will be maintained in an as-licensed state of cleanliness and free of loose debris.

212.130 (212.116) (6.3) The applicant should identify how emergency procedures address the possibility of ECCS degradation because of debris clogging the ECCS sump. He should also discuss Watts Bar design flexibilities which, though not credited in Chapter 15, might be available to restore long-term core cooling in the event of sump clogging.

212.131 (212.119) The response to Q212.119 (boron dilution event) provided a shutdown alarm setting time at 30 minutes after plant shutdown. This schedule is inadequate since it does not provide a schedule for resetting the alarm count setting to stay within an acceptable margin over background. Provide a resetting schedule for the alarm.

212.132 Your response to question 212.118 (VCT level instrumentation malfunction) is not sufficient to specifically justify the Watts Bar design for the scenerio posulated in question 212.118. Justify the Watts Bar design against the question 212.118 scenerio by providing a chronological timetable indicating initiation of the event, indications and alarms (credit given for operator corrective action only upon receipt of a control room alarm) with the times at which they occur, the time available to take operator action (after alarm) before the charging pumps are damaged, and a subsequent scenerio analysis if an acceptable operator action time to avert pump damage cannot be justified. In the analysis, justify the number of charging pumps assumed to be operating. Discuss the auxiliary charging system and whether it could be used as a backup to assure safety for this scenerio.