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NRC PDR

50-327A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 8 1978

Docket Nos: 50-327
50-328

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

Dear Mr. Hughes:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - SEQUOYAH

In recent weeks, our continuing review of the Sequoyah Nuclear Plant has identified a number of positions and requests for additional information. Copies of all of the attached items have previously been made available to members of your staff, and in the areas of containment systems and electrical and instrumentation and control systems, have been discussed at some length.

As we noted in our letter of November 15, 1978, your earliest possible response to these requests for additional information is required to preclude further delays in completion of our review.

Sincerely,

Steven A. Varga

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

Enclosure:
Request for Additional
Information

cc: See next page

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Tennessee Valley Authority - -

ccs:
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Request for Additional Informatio..
Containment Systems Branch
Sequoyah Nuclear Plant, Units 1 & 2
Docket Nos. 50-327/328

1. We have reviewed the information provided in response to our request number 6.55 regarding the steam generator and pressurizer enclosure s.bcompartment analyses and find that we will need the information outlined below to complete our review:

A. Steam Generator Enclosure

For the assumed break at the second elbow in the steam line:

1) Provide figures showing the differential pressure as a function of time for the following:

a) NODE 51 to NODE 60

b) NODE 52 to NODE 54

c) NODE 53 to NODE 55

d) NODE 56 to NODE 58

e) NODE 57 to NODE 59.

2) Provide figures showing the moments and forces acting on the steam generator vessel as a function of time.

3) Provide figures and tables as needed to identify the coordinate system and the steam generator surface areas and projection angles used in the development of the above force and moment time histories.

4) Describe the manner in which the above loads have been used in the analysis of the steam generator support structures.

5) For the above asymmetric pressure loads, identify the margin afforded by the steam generator support design.

B. Pressurizer Enclosure

For the assumed break of the pressurizer spray line at the inlet nozzle:

1) Provide figures showing the moments and forces acting on the pressurizer vessel as a function of time;

2) Provide figures and tables, as needed, to identify the coordinate system and pressurizer vessel surface areas and projection angles used in the development of the above force and moment time histories;

3) Describe the manner in which the above loads have been used in the analysis of the pressurizer supports.

4) For the above asymmetric pressure loads, identify the margin afforded by the pressurizer vessel support design.

2. We have reviewed your response to staff comment 9.38 and will require the following additional information to complete our review of the intended use of the containment purge system during normal plant operating periods.

1. Provide figures showing the location of the debris screens, and design details of the screens and mounting equipment.

2. Specify the differential pressure(s) for which the debris screens will be designed.

3. Provide assurance that the debris screens will be designed to the same criteria as the purge system containment isolation valves and penetrations; i.e., seismic Category I loads and Quality Group B, or equivalent, design requirements.

ENCLOSURE

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REQUEST FOR ADDITIONAL INFORMATION

3. Provide for our review a listing of all Class 1E safety-related equipment supplied by Westinghouse through their scope of supply. This shall include all equipment located both inside and outside of containment. It shall include, but not be limited to, sensors, signal-handling equipment, logic, switchgear, busses, inverters, instruments, motor operators, motor control centers, pumps, relay racks and instrument racks. In particular, the equipment located inside containment that is required to function during and subsequent to a steam line break event inside containment shall be identified. Where more than one item of a given type has been supplied, it is only necessary to list the type and number supplied, and to present the required information for one representative item of that type. The information supplied for each type shall include: (a) Description of item including type or model number, (b) usage in the plant, (c) identification of the topical reports in which the qualification was described for both the seismic and environmental qualification and (d) the IEEE and other standards on which the qualification was based.

4. The present system of position indication provided for the cold leg accumulator isolation valves loses its redundancy because it loses its power source when power to the motor control centers for the valves is removed to prevent spurious valve motion. This does not meet the requirements of BTP 4 and 18 (ICSB) and is

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therefore, unacceptable. Please provide for our review the description of a design that will be implemented on these valves that meets the requirements of these positions.

5. To assist in completing our review of the environmental qualification of the BOP Class 1E equipment, the staff requires that copies of test reports that show that the equipment meets the requirements of the industry standards and criteria for typical, representative items of BOP Class 1E electrical and instrumentation and control equipment be furnished for our review. These reports or supplementary information should identify the standards and criteria that are applicable. Such representative equipment should for example include: (a) 6.9 kV diesel generator breaker, (b) RWST level sensor, (c) 480 volt motor control center, (d) 125 volt battery charger, (e) control room vent radiation monitor and (f) containment isolation valve motor operator.

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6. On the first site visit, the staff identified a shield wire support tower that was located in close proximity to the two unit start busses. The collapse of this tower could occur in such a manner as to damage both of these busses. We require that this tower either be relocated away from these busses or be braced so that it cannot fall on the busses or that an analysis be made that shows that the tower cannot fall into the unit start busses and damage them to the extent that the flow of offsite power to the onsite system is jeopardized. Provide this information for our review.

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TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNIT NOS. 1 AND 2
PRESERVICE AND INSERVICE INSPECTION PROGRAM
COMPLIANCE WITH 10 CFR PART 50, APPENDICES G AND H
REQUEST FOR ADDITIONAL INFORMATION

MATERIALS ENGINEERING BRANCH
MATERIALS INTEGRITY SECTION

PRESERVICE AND INSERVICE INSPECTION PROGRAM

Request for Relief from Certain Testing Requirements

a. General Comments and Guidance

The information submitted by TVA for Sequoyah Nuclear Plant Unit Nos. 1 and 2 in your letter dated April 19, 1978 and amended July 5, 1978 does not clearly identify those items that do not fully comply with Section XI of the ASME Boiler and Pressure Vessel Code. Pursuant to 10 CFR Part 50, paragraph 50.55a(g)(5), the applicant shall indicate to the NRC those specific code requirements that are impractical for the facility and submit information to support these determinations. The detailed information necessary to support the applicant's relief requests shall be sufficient to demonstrate that the ASME Code requirements are impractical within the limitations of design, geometry, and materials of construction of the components and to determine if the use of alternatives will provide an acceptable level of quality and safety, as required by 50.55a, Codes and Standards.

For each relief request submitted, the following information should be included:

1. Identify the component(s) and/or the examination requirement for which relief is requested.

2. Number of items associated with the requested relief.

3. ASME Code class.

4. Identify the specific ASME Code requirement that has been determined to be impractical.

5. Information to support the determination that the requirement is impractical; i.e., state and explain the basis for requesting relief.

6. For items for which relief is requested indicate the portion of the Section XI examination requirement that can be performed. Indicate alternative examinations that can be used in lieu of Section XI requirements or to supplement partially performed Secti) XI examinations.

7. Describe and justify any changes expected in the overall level of plant safety by performing the proposed alternate examinations in lieu of the ASME Section XI examination. If it is not possible to perform examinations, discuss the impact on the overall level of plant quality and safety.

For inservice inspection only, provide the following additional information:

1. State when the relief request would apply during the inspection period or interval; i.e., is the request to defer an examination.

2. State when the proposed alternate examinations will be implemented and performed.

3. State the time period for which the requested relief is needed.

Technical justification or data must be submitted to support the relief request. Opinions without substantiation that a change will not affect the quality level are unsatisfactory. If the relief is requested for inaccessibility, a detailed description or drawing which depicts the inaccessibility must accompany the request. A relief request is not required for tests prescribed in Section XI that do not apply to your facility. A statement of N/A (not applicable) or none will suffice.

b. <u>Specific Areas Identified in the Review which may Require Relief</u>

Based on our review of TVA's preservice and inservice inspection program for Sequoyah Nuclear Plant Unit Nos. 1 and 2, we have identified the following items that may require relief:

1. The NRC's position is that the inspection of like type of nozzles for the reactor vessel, Item B1.4, Category B-D be evenly distributed over the inspection interval to meet the requirements of ASME Section XI. Since your program indicates this requirement will not be met, relief is required for this item.

2. Your program indicates that the control rod drive housings will not be examined per the requirements of Item B1.18, Category B-O; therefore, a request for relief is required. To support your relief request, discuss the maximum crack opening (largest size line break) in which the reactor can be shut down and cooled by the reactor coolant makeup system only. Compare the normal cool-down capacity with the maximum opening created if the CRDM housing weld failed.

3. Clarify the extent of volumetric examination that can be achieved for the cast stainless steel RCS piping. If the complete volume is not examined for the Category B-J welds, relief from the requirement is needed. Describe the volume that will be examined.

4. State if there is any bolting two inches in diameter or less (Category B-G-2) associated with the reactor vessel. If there is, include it in the program.

5. In Table A of the ISI program for Class A components, 2½ inch diameter welds listed under "Safety Injection system, circumferential and socket welds" are scheduled for PT examination. Clarify if these welds are socket welds. If the welds are not socket welds, relief is required from the volumetric examination requirements of Item B4.5, Category B-J.

6. In Table A of the ISI program for Class A components, 11 inch diameter stainless steel and six inch diameter Inconel piping welds are scheduled for PT examination. State why these welds are not being examined in conformance with the requirements of Item B4.5, Category B-J. Request relief for these items is necessary if the above stated requirements cannot be met.

7. State if the reactor coolant pump casings are welded. If the pump casings are welded, include the examination of the welds in the preservice/ISI program.

8. Since the inspection of valve bodies per the requirements of Category B-M-2 will not be fulfilled by your program, relief from the requirement is needed. For preservice inspection, indicate if the visual inspections of the valve bodies have been performed by the manufacturer and if the test records are available.

9. Section XI requires that an independent third party be utilized as the inspection agency. In your program you state the TVA will perform the independent third party inspection function itself since it is an agency of the Federal government. TVA should

provide the technical and/or legal justification that allows relief or exemption from having the third party inspection performed by an independent agency, other than TVA.

10. Provide details regarding the system testing that will be performed. Will the hydrostatic and leak testing meet the requirements of ASME Section XI?

11. In your ISI program, you state the program will essentially comply with ASME Section XI, 1974 Edition through Summer 1975 Addenda. However, for the examination of piping you proposed to use Appendix III of Section XI from a later code edition. Since the approved code, 1974 edition including Summer 1975 Addenda requires ultrasonic testing to Appendix I or Article 5 of ASME Section V, relief from the requirements of the code is needed.

12. Table B of the licensee's ISI submission dated April 19, 1978 includes ASME Class 2 piping, Categories C-F and C-G. The total sample of the piping is listed but not the sample to be tested or the test schedule for this interval. This information must be supplied to demonstrate compliance with code requirements.

13. Clarify the program statement that integrally welded supports and support components, Classes A, B and C will be examined as prescribed by ASME Section XI but the examinations are not included in Tables A and B.

COMPLIANCE WITH 10 CFR PART 50, APPENDICES G AND H

a. State if the ferritic materials in the pressurizer and steam generators that are part of the reactor coolant pressure boundary meet the exact requirements of 10 CFR Part 50, Appendix G. Provide the fracture toughness data available for these materials. State the ASME Code Edition used for these components.

b. State if all material for bolting and other fasteners greater than one inch diameter in the reactor coolant pressure boundary meet the exact fracture toughness requirements of 10 CFR Part 50, paragraph IV.A.4. Provide the fracture toughness data available for all bolting greater than one inch diameter used for the reactor coolant pressure boundary. Indicate the ASME Code Edition used for these components. The mechanical properties which have been submitted for the closure head bolting for Unit Nos. 1 and 2 need not be provided again.

c. State if the requirements of 10 CFR Part 50, Appendix G, paragraphs IIIB.3., 4, and 5 were met for the fracture toughness tests which were performed. Clarify if the testing was performed by an organization with a quality assurance program in conformance with 10 CFR Part 50, Appendix B.

d. Although Westinghouse Topical Report WCAP-7924 has been accepted by the NRC, the method for determining RT_{NDT} shift was not accepted. Provide information demonstrating that the pressure-temperature limits for reactor vessel heat-up and cooldown will be constructed using the prediction for temperature shift contained in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

STEAM GENERATOR INSERVICE INSPECTION

Confirm that the preservice and inservice inspection of steam generator tubing will be conducted in accordance with Regulatory Guide 1.83 Revision 1. If any of these examination requirements cannot be met, a complete technical justification to support your conclusions must be provided.