~~PDR~~ PDR

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

August 20, 1979

Mr. T. M. Anderson, Manager
Nuclear Safety Department
PWR Systems Division
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Anderson:

We are reviewing your Topical Report WCAP-9398, "Steam Generator Retubing and Refurbishment" submitted by your letter dated January 2, 1979. We find that we require additional information in order for us to continue our review.

Enclosure 1 details additional information which we need. It is anticipated that additional information requests will be forthcoming as our review continues.

Your timely responses to the enclosed requests will be appreciated. If you have any questions, the Project Manager for this review is Don Neighbors at 301-492-7037.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosure:
Request for additional information

7909250016

1023 161

WCAP-9398 STEAM GENERATOR RETUBING AND REFURBISHMENT
REQUEST FOR ADDITIONAL INFORMATION

Section 3.2

1. Provide a detailed description of the weld preparation and configuration, welding technique, post weld heat treatment, and NDE which will be used for the installation of the two 16" manway nozzle forgings in the lower steam generator shell.

2. Describe the technique that will be used to remove the tube-to-tubesheet welds.

Section 3.4

Provide a detailed description of the weld preparation and configuration, welding technique, post weld heat treatment, and NDE which will be used for the installation of the manway access ports in the secondary side of the steam generator.

Section 3.5

Provide a detailed description of the weld preparation and configuration, welding technique, post weld heat treatment, and NDE which will be used for the re-installation of the upper shell assembly and steamline.

Section 4.2.1.2

1. Is the tube expansion process mechanical or hydraulic?

2. Discuss the potential for springback following tube expansion and possible crevice formation as a result of different elastic properties of the tube and support plate materials.

Section 4.2.1.4

Describe the proposed heat treatments and complete experimental results supporting the conclusion that these heat treatments can result in a significant increase in resistance to stress corrosion cracking.

Section 4.2.1.5

What controls will be maintained to assure that tubes will not be expanded beyond the tubesheet and what would be the potential for corrosion of a tube expanded beyond the tubesheet?

Section 4.2.1.6

1. What Inconel alloy will be used for the internal blowdown pipe?

2. Provid a schematic of the blowdown pipe and design details of the blowdown line fastening method and materials.

Section 4.2.1.7

Document the baffle plate surface conditions, especially in the vicinity of the clearance holes and the edge of the center cut-out.

Section 4.2.1.8

Provide a detailed description of the experimental programs and results which supplied the corrosion characteristics of SA-240 Type 405 stainless steel. Include details of the testing environments including water chemistry and temperature.

Section 4.2.1.10

Illustrate the positioning and method of mounting the tube lane blocking devices. What materials are used for the blocking device and mounting components?

Section 4.2.1.12

Provide criteria which will be adhered to in making decisions on the several options noted in this section.

Section 4.2.1.13

Illustrate the configuration of the wrapper to shell lateral support blocks and contrast their number and design to the original lateral support blocks.

Section 5.0

Demonstrate that the methods and decontamination solutions used will not degrade or adversely affect the reactor coolant piping or components which are part of the primary system boundary. Further show that the decontamination solution will not have deleterious latent effects in subsequent plant operation.

Section 5.3.2.7.

Section 5.3.2.7 indicates that the tube stubs will be pulled out of the tubesheet from the secondary side. Discuss the removal process including the dislodging of the rolled portion of the tube from the tubesheet and pulling of this section through the tube hole. What are the effects on the tubesheet?

Section 6.1

1. Is WCAP-8370, Revision 8A in conformance with ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants" and has it received NRC review and approval?

2. With respect to the requirements of Regulatory Guide 1.43, what controls are used by Westinghouse to ensure the product is fine grain as specified?

3. Regulatory Guide 1.50 specifically states that Section IX, ASME Code welding procedure qualifications, are not adequate and that the additional requirement to qualify welding procedures at the minimum preheat temperature is necessary. Indicate your intent to comply with position c.1.b as stated in Regulatory Guide 1.50 or provide justification for an alternate position.

4. Is WCAP-8370 in conformance with ANSI N45.2.6-1973 as required by Regulatory Guide 1.64 and has it received NRC review and approval?

5. Regarding plugging criteria for degraded steam generator tubes, the tube plugging criteria must be determined on a plant specific basis and requires NRC approval. Proposed changes in tube plugging criteria will be evaluated by the NRC in accordance with Regualtory Guide 1.121.

6. Does WCAP-8370, Revision 8A conform with the requirements of ANSI N45.2.1 as required in Regulatory Guide 1.37 and has it been approved by the NRC?

7. Indicate the degree of compliance with the recommendations contained in Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel (May, 1973), and 1.71, Welder Qualification For Areas of Limited Accessibility (December, 1973).

Section 7.2.1.1

1. What are the required tube to tube hole tolerances necessary to ensure a successful tube roll which will maintain its integrity?

2. Clarify the discussion on required hydrotests following retubing. It is our interpretation that a Section XI hydrotest will be required following the modification.

3. Expand the description of the tubesheet analysis. Describe the development of the equivalent solid plate and its properties and describe the meaning of an interaction program.

4. Justify consideration of primary stresses only. Is thermal shock a significant concern?

5. Describe in detail the methodology for performing the fatigue evaluation of the tubesheet.

6. Discuss the applicability and use of Section III, Nonmandatory Appendix A, Article A-8000 of the ASME Code.

Section 7.2.1.2

1. The fracture mechanics analysis presented to determine the allowable flaw sizes is unacceptable. Treatment of the tube sheet as a solid homogenous section is unrealistic and the interaction of cracks and holes must be addressed to provide a realistic analysis.

Section 8.1

The scope of the transient/accident evaluation in the proposed topical report is limited by the following assumptions:

a) the licensing regulations and guidelines in effect at the time of the original license are assumed to apply, and

b) only changes in the safety analyses due to equipment changes are considered.

As a result of these assumptions, the evaluations in the proposed topical are limited to comparisons to the FSAR, to show that accident/transient analyses presented in the FSAR remain valid with the refurbished SG design.

Our review indicates that the current Technical Specifications, licensing regulations, and ECCS analyses are not necessarily limited to the regulations and guidelines in effect at the time of the original license. Our position is that the scope of the W generic transient/accident evaluation should be broadened such that plant-specific evaluations would apply licensing regulations, guidelines and Technical Specifications in effect for a plant at the time of steam generator refurbishment. For potential Technical Specification changes due to SG refurbishment, the evaluation of accidents should reflect up-to-date accident analysis models and requirements; and the evaluation of transients should determine the impact on the appropriate reference cycle(s), not necessarily the FSAR alone. Likewise, the evaluation of potential unreviewed safety questions due to SG refurbishment should be based on up-to-date regulations and use licensing guidelines in effect for a plant at the time of application.

2. The LOCA evaluation is based on a comparison of ECCS performance with the refurbished and original steam generators. However, in most cases the FSAR ECCS analysis using the original steam generators is based on a model which the staff no longer finds acceptable. Therefore, such FSAR analyses (or comparative evaluations) cannot be used to satisfy the requirements of 10 CFR 50.46. Also, the ECCS analysis current at the time of steam generator refurbishment would probably have been performed assuming a significant number of plugged steam generator tubes. If credit is to be taken for the unplugged configuration of the steam generator, a new LOCA analysis performed with the currently approved model is needed. For these reasons, we find that a LOCA analysis performed with the currently (at the time of application) approved model must be submitted on a plant specific basis prior to operation with the refurbished steam generators. The topical report should be supplemented to reflect this need for plant-specific ECCS analysis.

3. The topical report should be supplemented with a proposed methodology for performing the plant-specific review discussed in Section 8.1 which assures the applicability of the generic evaluation. This methodology should provide guidelines for the plant-specific review and should establish criteria for determining that the generic evaluation is applicable. We expect that this methodology would include the following:

   (a) a determination that each steam generator parameter is within the expected limits identified in the topical report (3.(c) above), and

   (b) a determination that the generic conclusion for each transient evaluation is valid considering plant unique design and analyses, and

   (c) a description of requirements for submitting the plant-specific evaluation for our review. For example, we would expect that a deviation from 4a or 4b would be identified and resolution discussed by submittal.

Section 8.2

Since the refurbished units will have fewer tubes than the original units, there will be a reduction in steam generator flow area, assuming tube diameter remains constant. Discuss how the increased resistance (from decreased flow area) will alter steam venting from the core during reflood and affect accident analysis?

Section 8.3

The topical report addresses the currently known differences in the refurbished steam generator design parameters but does not provide limits for these expected changes. Also, other steam generator parameters that potentially could affect the transient analysis are not addressed. For these reasons, the proposed topical should be supplemented with the following:

   (a) identify all steam generator parameters that could affect the transient analysis,

   (b) provide a quantitative estimate of the expected change in each parameter for the refurbished condition compared to the original condition (if a parameter is not expected to change, indicate such), and

   (c) provide a quantitative limit on the change for each parameter for which the transient evaluation as presented in the topical is expected to be valid.

Section 8.3.1

For startup of an inactive reactor coolant loop, confirm that all plants assume flow in the inactive loop accelerates to its nominal full flow value instantaneously, or discuss how the analysis is affected by the lower resistance to flow in the primary side. Also, discuss the startup of an inactive loop from a configuration with the loop stop valve initially closed.

Section 8.3.2

The evaluation of a reduction in feedwater enthalpy states that the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters is an extreme example of excess heat removal by the feedwater system. Excessive feedwater transients caused by accidental full opening of a feedwater control valve are not discussed. Confirm that the accidental opening of the feedwater heater bypass valve is the limiting FSAR analysis for reduction in feedwater enthalpy events for all plants or discuss other events that are limiting.

Section 8.3.6

For a steamline break, the report states that one mode for safety injection system actuation is pressurizer low pressure coincident with low pressurizer level. Provide an update for this mode of actuation, considering low pressurizer level signal removal after the TMI event. Provide any additional updating necessary as a result of other changes implemented or anticipated as a result of followup to the TMI event.