December 19, 1996

Mr. David J. Firth
Program Director
Generic License Renewal Program
Framatome Technologies, Inc.
P.O. Box 10935
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION IN THE CASE OF BAW-2251, "DEMONSTRATION OF AGING EFFECTS FOR THE REACTOR VESSEL (RAI NOS. 8 THROUGH 18)

Dear Mr. Firth:

The attached request for additional information (RAI) pertaining to the NRC staff's review of BAW-2251, "Demonstration of Aging Effects for the Reactor Vessel." This request contains RAIs number 8 through 17. RAIs 1 through 7 were forwarded in a letter to Mr. Don Cronenberger dated August 28, 1996. Your prompt response to this RAI will ensure that these aspects of the staff's review can be completed.

If you have any questions concerning the attached RAI, please contact me at 415-1106.

Sincerely,

Original signed by

John P. Moulton, Project Manager
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 683

Attachment: RAIs

cc: See attached list

DFO3 1/1

PROJ 683

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION IN THE CASE OF BAW-2251, "DEMONSTRATION OF AGING EFFECTS FOR THE REACTOR VESSEL (RAI NOS. 8 THROUGH 17)

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BAW-2251, Request for Additional Information Nos. 8 through 17

8. Section 4.5.1 of the report addresses the management of fatigue. The report indicates that each of the participating utilities monitor occurrences of design transients and are thereby managing the potential for cracking resulting from fatigue. Describe how this monitoring is being performed. Include a discussion of the instrumentation, if any, used to compare design transients with actual plant transients. Also provide documents which are the basis for your statement: "It has been demonstrated that the existing fatigue usage factors...remain valid for the period of extended operation.."

9. The report references the staff recommendations for environmental assisted fatigue provided in SECY 95-245. The report interprets the staff recommendation to be an assessment of environmentally assisted fatigue is required only if the fatigue usage factor [without considering environmental effects] exceeds 1.0 during the extended period of operation. This interpretation is incorrect. The recommendation in SECY 95-245 is that those components, evaluated as part of the staff's Fatigue Action Plan sample, that had high usage factors [calculated with fatigue curves that included environmental effects] be evaluated further for any extended period of operation. Provide an additional discussion regarding environmentally assisted fatigue based on the above clarification of the staff recommendation.

10. Identify the capsules and "host" reactors in the Integrated B&WOG Surveillance program that will be used to monitor the effect of neutron radiation on the beltline reactor vessel materials for the participating plants in the B&WOG Generic License Renewal Program over the license renewal term. Identify the materials in the capsules and the neutron fluence to be received by the capsules. Compare the neutron fluence of the capsules to the values specified in ASTM E-185-82 and explain the basis for the conclusion that the proposed integrated surveillance program complies with the requirements of Appendix H to 10 CFR Part 50 for the license renewal term.

11. Provide B&WOG Integrated Surveillance program test results from welds fabricated using the same heats of weld wire as were used to fabricate the beltlines of the reactor pressure vessels at the participating plants in order to demonstrate that the embrittlement estimates in Appendix A, "Pressurized Thermal Shock," are conservative. Compare the chemistry factors from surveillance data that are calculated using the methodology in section 2.1 of Regulatory Guide (RG) 1.99, Rev. 2 to the values used in Appendix A. Provide a determination of whether or not the surveillance data meet the credibility criteria in RG 1.99, Rev. 2. Identify the chemical composition (percentage amounts of copper and nickel) of the surveillance weld and the beltline weld materials. If the copper and nickel content of the surveillance weld and beltline welds are different, adjust the surveillance data in accordance with the procedures in section 2.1 of RG 1.99, Rev. 2.

ATTACHMENT

12. With respect to Appendix A, for "Pressurized Thermal Shock," in Topical Report BAW-2251, for all reactor pressure vessel (RPV) materials listed in Tables A-1 — A-5, provide the basis and references for the revised neutron fluence values at the inside RPV surfaces.

13. In BAW-2251, the B&WOG proposed to manage age related cracking in partial penetration welded Alloy 600 vessel head penetration nozzles by performing ASME defined VT-2 visual examinations of 25% of the nozzles (> 2 NPS) in the head. The B&WOG stated that the VT-2 examinations should be capable of detecting cracking the partial penetration nozzles. The B&WOG has also proposed to use leakage detection and surveillance of boric acid corrosion as acceptable methods for managing cracking of Alloy 600 components. However, the NRC issued a draft Generic Letter (GL) entitled "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetration" in which the staff concludes that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary to provide sufficient assurance of the structural integrity of these penetrations. The B&WOG should consider the information provided in the draft generic letter and will be required to provide an appropriate aging management program for managing PWSCC of Alloy 600 vessel head pernetrations, including <2" NPS penetrations.

14. IGSCC of closure stud assembly made of 4340 steel.

According to the Summary of Technical Information and Agreements from NUMARC Industry Reports Addressing License Renewal (O. CHopra, D. Ma, and W. Shack, Argonne National Laboratory, November 1994), NUMARC/NRC agreements were that ASME XI examinations performed in accordance with Regulatory Guide (RG) 1.65, "Materials and Inspection for Reactor Vessel Closure Studs," were acceptable management methods for managing the IGSCC of closure stud assemblies. The B&WOG does not state in its report that ASME inspections will be done using the guidance in RG 1.65. Revise your topical report to commit to performing the inspections using the guidance in RG 1.65 or provide justification for not performing them.

15. Page 4-5 of the report indicates that surveillance capsule testing will be completed by year 2008. Discuss what capsules, if any, will remain in the reactors after 2008 and their withdrawal schedule.

16. Page 4-10 of the report indicates loss of material on the mating surfaces between the closure flanges is managed by ASME Section XI Examination Category B-N-1. It then discusses that B-N-1 requires a full visual examination of all accessible areas of the interior of the reactor vessel. Clarify whether B-N-1, that is, examination of all accessible areas, or just a small portion of B-N-1, that is, examination of only the flange areas, is an inspection necessary for renewal.

17. Page 4-10 of the report indicates that the fatigue usage factor for the Oconee Nuclear Station Units 1, 2, and 3 reactor vessel studs is projected to exceed 1.0 by the end of the renewal term and will be evaluated separately for renewal on a plant specific basis. However, because B&W plants are generally similar in design and operation, discuss why the studs at the other B&WOG GLRP plants are not similarly affected.

Project No. 683

Babcock & Wilcox Owners Group Generic License Renewal Program

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