September 14, 2000

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: SAFETY EVALUATION FOR EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2, REQUEST FOR RELIEF NO.: RR-APPENDIX VII-1 (TAC NOS. MA9277 AND MA9278)

Dear Mr. Sumner:

By letter dated July 31, 2000, which supercedes your letter dated June 15, 2000, you submitted Relief Requests RR-32, RR-33, and RR-APP. VIII-1. We have completed our review of RR-APP. VIII-1 and our safety evaluation is enclosed. The staff concludes that the proposed alternative to use 0.75 inch RMS in lieu of the requirements in Subparagraph 3.2(b) to Appendix Vlll of Section XI of the American Society of Mechanical Engineers Code will provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative RR-APP. Vlll-1 is authorized for the third 10-year interval. Relief requests RR-32 and RR-33 will be addressed in future correspondence.

Sincerely,

/RA/

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

REQUEST FOR RELIEF NO.: RR-APPENDIX VIII-1

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

The inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where alternatives have been authorized by the Commission pursuant to 10 CFR 50.55a(a)(3). 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements may be used providing the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the third 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission Approval.

By letter dated July 31, 2000, Southern Nuclear Operating Company, Inc. (the licensee) requested relief from certain ultrasonic testing (UT) requirements pertaining to the examination of Class 1 reactor pressure vessel (RPV) nozzle-to-vessel welds (RR-32), annual UT training requirements of Section XI of the Code (RR-33), and RPV length sizing qualification

requirement (RR-APP. Vlll-1) for the third 10-year ISI interval at Edwin I. Hatch Nuclear Plant. RR-32 and RR-33 will be reviewed in a separate safety evaluation.

2.0 RR-APP. Vlll-1, RPV LENGTH SIZING TOLERANCE

The system/component for which relief is requested is the RPV Welds subject to inservice examination requirements according to Supplement 4 to Appendix Vlll of Section XI of the Code.

2.1 Requirements for which Relief is Requested

10 CFR 50.55a(b)(2)(xv)(C)(1) requires a depth sizing acceptance criterion of 0.15 inch root mean square (RMS) be used in lieu of the requirements of Subparagraph 3.2(b) to Supplement 4 to Appendix Vlll of Section XI of the 1995 Edition with 1996 Addenda of the Code.

2.2 Licensee's Basis for Relief

On January 12, 2000, discussions were held between NRC staff, representatives from the Electric Power Research Institute Nondestructive Examination Center, and representatives from the Performance Demonstration Initiative (PDI). The discussion included the difference between Supplement 4 to Appendix Vlll of Section XI of the Code, 10 CFR 50.55a(b)(2)(xv)(C)(1), and the implementation of Supplement 4 to Appendix Vlll by the PDI Program. Supplement 4, Subparagraph 3.2(b) provides a length sizing qualification criterion that requires flaw length estimated by ultrasonic be the true length -¼ inch +1 inch. The rule changed Subparagraph 3.2(b) to a depth sizing requirement of 0.15 inch RMS, and the PDI Program has been using a length sizing tolerance of 0.75 inch RMS for paragraph 3.2(b). The NRC staff acknowledged that 10 CFR 50.55a(b)(2)(xv)(C)(1) is in error and should actually be a length sizing tolerance of 0.75 inch RMS, the same tolerance that is being used in the PDI Program.

2.3 Licensee's Proposed Alternative to Code

The licensee proposed that a length sizing qualification criterion of 0.75 inch RMS be used in lieu of the sizing requirements in Subparagraph 3.2(b) to Appendix Vlll of Section XI of the 1995 Edition with 1996 Addenda of the Code.

2.4 Evaluation

The nuclear utilities in the United States created the PDI to implement performance demonstration requirements contained in Appendix Vlll of Section XI of the Code. The PDI developed a performance demonstration program for qualifying UT techniques. In 1995, the NRC staff performed an assessment of the PDI program and identified PDI's use of a length sizing tolerance of 0.75 inch RMS for the reactor pressure vessel performance demonstrations. The criterion was introduced to reduce testmanship (passing the test based on manipulation of

results rather than skill). The staff noted in the assessment report dated March 6, 1996, that the length sizing tolerance was not according to Appendix Vlll but did not take exception to the PDI's implementation of the 0.75 inch RMS length sizing tolerances. The staff requested that the length sizing difference between the PDI and the Code be resolved.

The solution for resolving the differences between the PDI program and Code was for the PDI to participate in the development of a code case. The code case was presented to ASME for discussion and consensus building. NRC representatives participated in this process. ASME approved the code case and published it as Code Case N-622, "Ultrasonic Examination of RPV and Piping and Bolts and Studs, Section XI, Division 1."

Operating in parallel with these actions, the staff incorporated most of Code Case N-622 criteria in 10 CFR 50.55a(b)(2)(xv). On January 12, 2000, the PDI identified the omission of the length sizing tolerance in 10 CFR 50.55a(b)(2)(xv)(C)(1). The staff agreed that the omission of the length sizing tolerance 0.75 inch RMS in the rule was an oversight and that the inclusion of depth sizing tolerance in Subparagraph 3.2(b), Supplement 4 to Appendix Vlll of Section XI of the Code was an error. Based on the above discussion, the staff believes that the proposed alternative to use a length sizing tolerance of 0.75 inch RMS in lieu of the requirements in Subparagraph 3.2(b) will provide an acceptable level of quality and safety.

2.5 Conclusion

The staff concludes that the proposed alternative to use 0.75 inch RMS in lieu of the requirements in Subparagraph 3.2(b) to Appendix Vlll of Section XI of the Code will provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative RR-APP. Vlll-1 is authorized for the third 10-year interval.

Principal Contributor: D. Naujock

Date: September 14, 2000

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