December 5, 2001

Mr. Oliver D. Kingsley, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: REQUEST TO USE ALTERNATIVE TO AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (ASME CODE) SECTION XI INSERVICE INSPECTION PROGRAM RELATED TO DEPTH SIZING QUALIFICATION CRITERION FOR THE OYSTER CREEK NUCLEAR GENERATING STATION (TAC NO. MB0905)

Dear Mr. Kingsley:

By letter dated December 21, 2000, as supplemented on February 20, 2001, AmerGen Energy Company, LLC, (AmerGen), submitted proposed Alternative VIII-1, Depth Sizing Criteria, and proposed Alternative VIII-2, Annual Training, for the Inservice Inspection (ISI) Program at the Oyster Creek Nuclear Generating Station (Oyster Creek). This submittal was supplemented by letter dated February 20, 2001, revising the proposed Alternative VIII-1 Depth Sizing Criteria. By Alternative VIII-1, the licensee proposed to use a length sizing qualification criterion of 0.75 inch root mean square (RMS) in lieu of the criterion contained in Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2(b), of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Also by proposed Alternative VIII-1, the licensee proposed to use the depth sizing requirement of 0.15-inch RMS consistent with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(b)(2)(xv)(C)(1) in lieu of the requirements contained in Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2(c), of the ASME Code. The U. S. Nuclear Regulatory Commission (NRC) staff has reviewed the alternatives proposed in Alternative VIII-1. Alternative VIII-2 will be addressed by separate correspondence.

The alternative proposed for Subparagraph 3.2(b) of the ASME Code, Section XI, Appendix VIII, Supplement 4, is no longer needed. On March 26, 2001, the NRC published in the *Federal Register* (66 FR 16390) a rule change to 10 CFR 50.55a(b)(xv)(C)(1), which deals with flaw detection criteria. The rule change corrected an earlier administrative error in the regulation, and the alternative you sought is no longer required.

Based on the information provided, the NRC staff concludes that the alternatives proposed by Alternative VIII-1 to Subparagraph 3.2(c) of the ASME Code, Section XI, Appendix VIII, Supplement 4, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the ISI Program alternative proposed in Alternative VIII-1 for Subparagraph 3.2(c) for the third ISI 10-year interval at Oyster Creek. The

NRC staff's safety evaluation is enclosed. If you have any questions, please call Helen N. Pastis, the Senior Project Manager for Oyster Creek at (301) 415-1261.

Sincerely,

/RA/

L. Raghavan, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosure: Safety Evaluation

cc w/encl: See next page

NRC staff's safety evaluation is enclosed. If you have any questions, please call Helen N. Pastis, the Senior Project Manager for Oyster Creek at (301) 415-1261.

Sincerely,

/RA/

L. Raghavan, Acting Chief, Section 1
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

PROPOSED ALTERNATIVE VIII-1

AMERGEN ENERGY COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

The inservice inspection (ISI) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, Class 2, and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable editions and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if 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 first 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 twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI Code of Record for the Oyster Creek Nuclear Generating Station (Oyster Creek), for the third 10-year interval is the 1986 Edition of the ASME Code. 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.

The NRC staff has reviewed the information submitted by AmerGen Energy Company, LLC (AmerGen or licensee), in a letter dated December 21, 2000, as supplemented on February 20, 2001, requesting relief from certain ASME Code-required ultrasonic testing (UT) criteria. The licensee's proposed alternative to the ASME Code requirements is contained in proposed Alternative VIII-1 for the third 10-year ISI interval at Oyster Creek.

2.0 ALTERNATIVE VIII-1, UT LENGTH SIZING TOLERANCE FOR REACTOR PRESSURE VESSEL PERFORMANCE DEMONSTRATIONS

2.1 ASME Code Requirements for which Relief is Requested

The licensee requested relief from Appendix VIII, Supplement 4, Subparagraphs 3.2(b) and 3.2(c), to Section XI of the ASME Code, 1995 Edition with 1996 Addenda.

2.2 Licensee's Proposed Alternative to the ASME Code

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed using a length sizing qualification criterion of 0.75-inch root mean square (RMS) in lieu of Appendix VIII, Supplement 4, Subparagraph 3.2(b). The licensee also proposed to use the RMS value of 0.15-inch as specified in 10 CFR 50.55a(b)(2)(xv)(C)(1) which modifies the depth sizing criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c).

2.3 Evaluation

The alternative proposed for Subparagraph 3.2(b) of the ASME Code, Section XI, Appendix VIII, Supplement 4 is no longer needed. On March 26, 2001, the NRC published in the *Federal Register* (66 FR 16390) a rule change to 10 CFR 50.55a (b)(2)(xv(C)(1), which deals with flaw detection criteria. The rule change corrected an earlier administrative error in the regulation, and the alternative sought is no longer required.

Section 50.55a(g)6)(ii)(C) imposes implementation of Appendix VIII to the 1995 Edition with 1996 Addenda of Section XI of the ASME Code. The imposed implementation schedule for Supplement 4 to Appendix VIII was November 22, 2000. Supplement 4, Subparagraph 3.2(c), requires that the UT performance demonstration results be plotted on a two-dimensional plot with the measured depth plotted along the ordinate axis and the true depth plotted along the abscissa axis. For qualification, the plot must satisfy the following statistical parameters: (1) slope of the linear regression line is not less than 0.7; (2) the mean deviation of flaw depth is less than 0.25 inches; and (3) correlation coefficient is not less than 0.70.

As an alternative, the licensee proposed eliminating the use of Supplement 4, Subparagraph 3.2(c), which imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15-percent through-wall. The differences between actual versus true value produce a tight grouping of results which resemble a shotgun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus, making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the ASME Code is too lax with respect to evaluating flaw depths within the inner 15-percent of wall thickness. Therefore, the licensee proposed to use the more appropriate criterion of 0.15-inch RMS of

10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application because it is based on the linear regression from Subparagraph 3.2(c)(1).

In 1991, the U.S. nuclear utilities created the Performance Demonstration Initiative (PDI) project at the Electric Power Research Institute's (EPRI's) Nondestructive Examination (NDE) Center in Charlotte, North Carolina, to implement performance demonstration requirements contained in Appendix VIII of Section XI of the ASME Code. The EPRI NDE center provides technical support and administration for this project on behalf of the utilities. To this end, PDI has developed a performance demonstration program for qualifying UT equipment, procedures, and personnel.

PDI was aware of the inappropriateness of Subparagraph 3.2(c) early in the development of their program. They brought the issue before the appropriate ASME committee which formalized eliminating the use of Supplement 4, Subparagraph 3.2(c) in Code Case N-622. NRC staff representatives participated in the discussions and consensus process of the code case. Based on the above, the NRC staff has determined that the use of the Subparagraph 3.2(c) requirements in this context is inappropriate and that the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c) will provide an acceptable level of quality and safety.

3.0 CONCLUSION

Based on the discussion above, the NRC staff has concluded that the alternative proposed in Alternative VIII-1 for the third 10-year interval will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed alternative for the third 10-year interval.

Principal Contributor: H. N. Pastis

Date: December 5, 2001