July 11, 2002

Mr. John L. Skolds, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - RELIEF REQUEST R17, REVISION 1, EXAMINATION COVERAGE FOR REACTOR PRESSURE VESSEL AXIAL WELDS (TAC NO. MB2940)

Dear Mr. Skolds:

By letter dated September 14, 2001, as supplemented June 3, 2002, AmerGen Energy Company, LLC (AmerGen), submitted Relief Request R17, Revision 1. The submittals requested relief for Oyster Creek Nuclear Generating Station (OCNGS) from certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, requirements, regarding volumetric examination coverage requirements for reactor pressure vessel (RPV) welds.

The 1986 Edition of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.10, requires examination of all welds in the first inspection interval and one beltline region weld in successive inspection intervals. However, Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(6)(ii)(A)(2) concurrently requires all licensees to augment their RPV examinations by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel" (i.e., inspection of essentially 100% of the volume of the vessel axial welds). AmerGen requested Nuclear Regulatory Commission (NRC) authorization of its proposed alternative to the requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2).

The NRC staff reviewed the referenced submittals and has set forth details of its findings in the enclosed safety evaluation. Based on its review, the NRC staff concludes that the approximately 60% composite coverage proposed by AmerGen provides 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 inservice inspection interval at OCNGS.

If you have any questions regarding this relief, please call Mr. Peter Tam, NRC Project Manager, at (301) 415-1451.

Sincerely,

/RA by SRichards for RLaufer/

Richard J. Laufer, 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

If you have any questions regarding this relief, please call Mr. Peter Tam, NRC Project Manager, at (301) 415-1451.

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**S. Richards signed and concurred for R. Laufer.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM, RELIEF REQUEST R17, REVISION 1

OYSTER CREEK NUCLEAR GENERATING STATION

AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-219

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 is to be performed in accordance with Section XI of the ASME Code and applicable edition 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). In addition, 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the Nuclear Regulatory Commission (NRC), if the applicant 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) will 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) 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 Oyster Creek Nuclear Generating Station's (OCNGS) third 10-year ISI interval is the 1986 Edition of the ASME Code.

By letter dated September 13, 2000, the NRC staff approved relief request R17 for the OCNGS reactor pressure vessel (RPV) Code, Category B-A, volumetric examination coverage limitations. The limitation percentages were based on an internal vessel accessibility study of the RPV prior to performing the subject examinations. By letter dated September 14, 2001, as supplemented June 3, 2002, AmerGen Energy Company, LLC (the licensee), submitted Relief Request R17, Revision 1, for the same OCNGS volumetric examination coverage requirements for the same RPV welds, Code Category B-A, based on actual percentages calculated after completion of the volumetric examinations. These actual percentages are substantially less than the percentages projected from the accessibility study.

Enclosure

2.0 NRC STAFF EVALUATION

2.1 Code Requirements for which Relief is Requested

ASME Code Section XI, Table IWB-2500-1, Category B-A, Item B1.12 requires examination of all welds in the first inspection interval and one beltline region weld in the successive inspection intervals. However, 10 CFR 50.55a(g)(6)(ii)(A)(2) states that all licensees shall augment their reactor vessel examinations by implementing the examination requirements for RPV welds in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB of the 1989 Edition of ASME Code, Section XI, Division 1, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A)(2). For the purposes of this augmented examination, "essentially 100%," as used in Table IWB-2500-1, means more than 90% of the examination volume for each weld.

2.2 Licensee's Basis for Relief

The licensee determined that conformance with the volumetric coverage requirements is impractical. The OCNGS reactor, a BWR-2, was designed and built before ASME Code, Section XI was developed and access for inspections became a design requirement. The licensee indicated that there is little external access to the outside diameter of the RPV axial shell welds due to inadequate clearance between the bioshield wall and vessel insulation. This access limitation forced the licensee to perform an internal surface directed ultrasonic inspection of the axial shell welds using the General Electric (GE) GERIS-2000 inspection system.

The welds where limited examination volume was achieved and the reason for the limitation is listed in Table 1 below:

Table 1

Weld Identification	Examination Limitations	Code Volume Examined
NR02 2-563A*	None	100.0%
NR02 2-563B*	Steam Dryer Support Lug	92.8%
NR02 2-563C	Main Steam Nozzle Plug	80.4%
NR02 2-563D	Feedwater Sparger, Manipulator Scan Limits	62.3%
NR02 2-563E	Feedwater Sparger, Manipulator Scan Limits	59.7%
NR02 2-563F	Feedwater Sparger, Manipulator Scan Limits	59.7%
NR02 2-564A	Feedwater Sparger, Manipulator Scan Limits	0.0
NR02 2-564B*	None	100.0%
NR02 2-564C	Shroud Repair Tie Rod	39.7%
NR02 2-564D	Recirculation Outlet Nozzle	56.1%

NR02 2-564E	Shroud Repair Tie Rod	74.7%
NR02 2-564F	Feedwater Sparger Brackets Interference with Scanner	0.0

*Meets Code requirements for examination volume

The licensee stated that it had submitted Relief Request R17 based on the estimated coverage that would be obtained. This estimated coverage was based on GE's assessment of weld locations and clearances based on drawings. The actual coverages obtained after completion of the examinations are listed in Table 2 below:

Table 2

Weld Identification	Weld Configuration	Weld Length	Estimated Coverage	Actual Coverage
NR02 2-563A*	Longitudinal, Upper Shell @ 15° AZ	132.6"	100.0%	100.0%
NR02 2-563B*	Longitudinal, Upper Shell @ 135° AZ	132.6"	99.2%	92.8%
NR02 2-563C	Longitudinal, Upper Shell @ 255° AZ	132.6"	99.4%	80.4%
NR02 2-563D	Longitudinal, Int. Upper Shell @ 330° AZ	132.6"	65.3%	62.3%
NR02 2-563E	Longitudinal, Int. Upper Shell @ 90° AZ	132.6"	65.3%	59.7%
NR02 2-563F	Longitudinal, Int. Upper Shell @ 210° AZ	132.6"	62.6%	59.7%
NR02 2-564A	Longitudinal, Lower Int. Shell @ 219° AZ	133.6"	93.0%	0.0
NR02 2-564B*	Longitudinal, Lower Int. Shell @ 339° AZ	133.6"	93.0%	100.0%
NR02 2-564C	Longitudinal, Lower Int. Shell @ 99° AZ	133.6"	94.1%	39.7%
NR02 2-564D	Longitudinal, Lower Shell @ 258° AZ	83.4"	55.1%	56.1%
NR02 2-564E	Longitudinal, Lower Shell @ 18° AZ	131.6"	76.0%	74.7%
NR02 2-564F	Longitudinal, Lower Shell @ 138° AZ	131.6"	76.0%	0.0

*Meets Code requirements for examination volume

Welds 2-563D, 2-563E, 2-563F, and 2-564E, coverages were slightly less than the estimated coverages by 3 to 6%. Weld 2-563C was estimated to achieve the ASME Code coverage but was limited to 80.4% actual due to the main steam line plug which was unanticipated during the accessibility study. Weld 2-564A was not located at the azimuth appearing on the drawing, but was located behind a tie rod, which is six degrees counter clockwise from the position shown on the drawing. The same tie rod offset configuration existed for weld 2-564C which reduced examination coverage to 39.7%. Weld 2-564F is located directly below a feedwater nozzle where the nozzle-to-sparger configuration makes it impossible to position the transducer array flush with the weld.

The licensee stated that the NRC staff acceptance of the original relief request was based upon an estimated 82% total coverage with 100% coverage of the axial welds in the beltline region. The actual was 60% total coverage with 57% coverage of the axial welds in the beltline region. Since the actual coverage was significantly less than estimated, a failure probability was performed using the VIPER code developed by Structural Integrity Associates, Inc. (SIA). The evaluation was based on the methodology presented in Electric Power Research Institute (EPRI) Report, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” BWRVIP-05. Based on the evaluation, the inspections performed were effective in inspecting regions that account for essentially all of the vessel fracture failure risk. The examination coverage resulted in a small failure probability of 2.5×10^{-12}/reactor-year risk which the licensee argued is an acceptable level of quality and safety.

2.3 Evaluation

The 1986 Edition of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.10, requires examination of all welds in the first inspection interval and one beltline region weld in successive inspection intervals. However, 10 CFR 50.55a(g)(6)(ii)(A)(2) concurrently requires all licensees to augment their RPV examinations by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in Item B1.10 of Examination Category B-A, “Pressure Retaining Welds in Reactor Vessel,” in Table IWB-2500-1 of subsection IWB of the 1989 Edition of Section XI of the ASME Code, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A)(3) and (4). The licensee is requesting NRC authorization of its alternative to the requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2), which requires an augmented examination of essentially 100% of the volume of the vessel axial welds. Three welds in the lower intermediate shell course were approximately six degrees counter clockwise from the location specified on the drawings. Two welds could not be examined because of the shroud repair tie rod and feedwater sparger bracket interferences.

The licensee stated that it had performed a failure probability of the welds in the beltline region using the VIPER code developed by SIA. The evaluation by SIA was performed to assess the effect on the probability of fracture due to the 57% composite coverage performed on the vessel axial welds. The evaluation was based on the methodology presented in EPRI Report, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” BWRVIP-05. The licensee stated the inspections performed were effective in inspecting regions that account for essentially all of the vessel fracture failure risk based on a very small failure probability of 2.5×10^{-12}/reactor-year as a result of the 57% composite beltline examination coverage.

The failure frequency value is determined by the product of the conditional failure probability and a low temperature overpressurization transient probability. The licensee stated in its letter dated June 3, 2002, that OCNGS’ specific weld chemistry values were used in the evaluation and these values are significantly lower than the bounding values used in the BWRVIP-05 calculations. Secondly, the failure probabilities in the BWRVIP-05 were determined using bounding values for weld chemistry and RPV geometry. The BWRVIP-05 analysis used an RPV thickness of 5.25 inches and a diameter of 225.2 inches. This compares with a thickness of over 7 inches and a diameter of 213 inches for the OCNGS vessel. The licensee stated that these differences result in a significant reduction in the hoop stress compared to the BWRVIP-05 evaluation, which significantly reduces the failure probability for OCNGS. Though the staff did not review the probability risk assessment in detail, the greater thickness and lower diameter of the OCNGS vessel provide a reasonable basis for the calculated low failure probability of 2.5×10^{-12}/reactor-year. This value is sufficiently below the RPV failure frequency due to failure of the limiting axial welds in the boiling water reactor fleet at 5×10^{-6}/reactor-year, which is consistent with the guidelines of Regulatory Guide 1.154, and provides reasonable assurance of a low failure probability due to the limited examination coverage.

Finally, though the actual coverage of each weld was significantly lower than the projected coverage, the NRC staff believes that the actual coverage obtained would identify any pattern of degradation, particularly in the high fluence region of the vessel wall. The high fluence region is the area that experiences the highest neutron bombardment and would be the first region to display any embrittlement issues caused by irradiation-assisted cracking. The NRC staff concludes that the approximately 60% composite coverage provides an acceptable level of quality and safety.

3.0 CONCLUSION

Based on the discussion above, the NRC staff concludes that the alternative will provide an acceptable level of quality and safety, and is in compliance with 10 CFR 50.55a(a)(3)(i). Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the NRC staff authorizes the proposed alternative for the third 10-year inservice inspection interval for OCNGS, identified as Relief Request R17, Revision 1.

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Date: July 11, 2002

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