March 30, 2006

C. N. Swenson
Site Vice President
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731-0388

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE OYSTER CREEK NUCLEAR GENERATING STATION, LICENSE RENEWAL APPLICATION (TAC NO. MC7624)

Dear Mr. Swenson:

By letter dated July 22, 2005, AmerGen Energy Company, LLC (AmerGen or the applicant) submitted to the U.S. Nuclear Regulatory Commission (NRC or the staff) an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54), to renew the operating license for Oyster Creek Nuclear Generating Station. The NRC staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These questions were discussed with members of your staff during a conference call on March 16, 2006. A mutually agreeable date for a response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3191 or via e-mail at DJA1@nrc.gov.

Sincerely,

/RA/

Donnie J. Ashley, Project Manager
License Renewal Branch A
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosure:
As stated

cc w/encl: See next page

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Oyster Creek Nuclear Generating Station

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Warrenville, IL 60555

Ltr. to C.N. Swenson from Donnie Ashley dated: March 30, 2006

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE OYSTER CREEK NUCLEAR GENERATING STATION, LICENSE RENEWAL APPLICATION (TAC NO. MC7624)

Adams Accession No.: **ML060890660**

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OYSTER CREEK NUCLEAR GENERATING STATION
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)

RAI 4.2.2-1

Please provide the bounding values for the percentage decrease in the upper-shelf energy (USE) at 50 effective full-power years (EFPY) based on the equivalent margin analysis (EMA), as well as the predicted percentage decrease in USE at 50 EFPY, as determined from Regulatory Guide (RG) 1.99, Revision 2, for all Reactor Vessel (RV) beltline materials.

RAI 4.2.2-2

Table 4.2.2-1 in Section 4.0 of the LRA provides the Adjusted Reference Temperature (ART) values for the RV beltline materials. The chemistry data (%Cu and %Ni) and chemistry factor (CF) values for the Lower-to-Lower Intermediate Shell Circumferential Weld 3-564; Lower Shell Axial Welds 2-564A, B, and C; and Lower Intermediate Shell Axial Welds 2-564D, E, and F from Table 4.2.2-1 are less conservative than the corresponding chemistry data and CF values that were established in the staff's reactor vessel integrity database (RVID) for these welds.

Please supplement Section 4.0 of the LRA with the following information:

a. verification of whether the chemistry data contained in Table 4.2.2-1 of Section 4.0 are valid for the above welds.

b. justification for the use of these chemistry data for the above welds, including the source of the data, and a specific reference for the documentation/analysis demonstrating that these chemistry data represent the best available estimate of the weld chemistries.

RAI 4.2.7-1

Boiling Water Reactor (BWR) Vessels and Internals Project-26, "BWR Vessels and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines," indicates that BWR stainless steel components exposed to a fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC). The safety evaluation report (SER) for BWRVIP-26 considers IASCC of BWR reactor internals a time-limited aging analysis (TLAA) issue.

Section 4.2.7, Reactor Internals Components, of the LRA indicates that the core shroud, incore instrumentation dry tubes, and top guide have been exposed to a fluence exceeding 5×10^{20}/cm^2 (E > 1 MeV) and are, therefore, considered susceptible to IASCC, based on fluence calculations performed for these components. However, no TLAA associated with IASCC exists for the core shroud, incore instrumentation dry tubes, or top guide.

Please clarify why there is no TLAA for the core shroud, incore instrumentation dry tubes, and top guide given that these components have been exposed to a fluence exceeding 5×10^{20} n/cm^2 (E > 1 MeV) and are considered susceptible to IASCC.

RAI 4.7.4-1

Section 4.7.4, Reactor Vessel Weld Flaw Evaluations, of the LRA includes a discussion of several flaws that were detected in two axial RV welds during inservice inspections performed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME) Code, Section XI and documented in a 2000 inservice inspection report. The flaws were previously evaluated and found to be acceptable for the current licensing period in accordance with the ASME Code, Section XI, IWB-3600. Section 4.7.4 of the LRA indicates that these flaws were reevaluated for conditions of extended operation through 50 EFPY and found to be acceptable in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, IWB-3600.

Please submit the analysis demonstrating that these flaws are acceptable in accordance with the ASME Code, Section XI, IWB-3600 for conditions of extended operation through 50 EFPY.

RAI 4.7.5-1

Section 4.7.5, Control Rod Drive (CRD) Stub Tube Flaw Analysis, of the LRA includes a discussion of several cracks that were found in the CRD stub tubes during construction and a subsequent repair of the cracks. This repair includes the grinding out of the observed cracks followed by the application of a weld overlay. The LRA indicated that, following the repair, an analysis was performed to demonstrate that any crack that would have remained undetected following the repairs would not propagate through the weld overlay during the life of the plant. Furthermore, the LRA states that this analysis demonstrated that more than 1000 startup and shutdown cycles would be required in order for any such postulated crack to propagate through the overlay to the surface of the CRD stub tube. The LRA states that this information was provided to the Atomic Energy Commission (AEC) in Amendment 37 to the provisional operating license application.

The cumulative number of startup and shutdown cycles through the end of the period of extended operation is projected to be less than 275. Therefore, the LRA stipulates that the above evaluation remains valid for ensuring CRD stub tube integrity through the end of the period of extended operation.

Given the extent of operating experience since the time of the original analysis, there is a possibility that other CRD stub tube degradation mechanisms that were not known or considered at the time of the original analysis could potentially compromise the integrity of the CRD stub tube over the period of extended operation. Please discuss whether there are any known degradation mechanisms discovered since the time of implementation of the CRD stub tube repair that could potentially invalidate the original analysis discussed above. If any CRD stub tube degradation mechanism is known to exist that was not taken into consideration at the time of the original analysis, thereby, potentially invalidating that analysis, please submit a revised TLAA for the CRD stub tubes demonstrating that the integrity of these components will be maintained over the period of extended operation.