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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Oyster Creek Nuclear Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Oyster Creek Nuclear Generating Station – Response to Request for Additional Information Regarding Control Rod Drive Notch Testing Frequency (TAC NO. MD8932)

References:

(1) Letter from P. B. Cowan (AmerGen Energy Company, LLC) to U.S. Nuclear Regulatory Commission, "Application for Technical Specification Change Regarding Revision of Control Rod Notch Surveillance Test Frequency, Clarification of SRM Insert Control Rod Action, and Clarification of a Frequency Example Using the Consolidated Line Item Improvement Process," dated June 9, 2008

(2) Letter from P. B. Cowan (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request RE: Application for Technical Specification Change Regarding Revision of Control Rod Notch Surveillance Test Frequency, Clarification of SRM Insert Control Rod Action, and Clarification of a Frequency Example Using the Consolidated Line Item Improvement Process," dated March 30, 2009

(3) Letter from E. Miller (U.S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 - Request for Additional Information Regarding Control Rod Drive Control System Replacement License Amendment (TAC NO. MD8932)," dated August 28, 2009

By letter dated June 9, 2008 (Reference 1), AmerGen Energy Company, LLC (now Exelon Generation Company, LLC (Exelon)) requested a change to the Technical Specifications to implement TSTF/CLIIP-475. Reference 2 involved additional information requested by the NRC associated with the proposed change. Subsequently, the NRC determined that additional information is needed to complete its review (Reference 3). Exelon's response to the NRC question in Reference 3 is provided in Attachment 1 to this letter.

Exelon has determined that the information provided in this response does not change the proposed Technical Specifications marked-up pages (Reference 2) and does not impact the conclusions of the No Significant Hazards Consideration as stated in Reference 1.

There are no regulatory commitments contained in this letter.

A copy of this letter and its attachments are being provided to the designated State official and the chief executive of the township in which the facility is located.

Should you have any questions concerning this letter, please contact Frank J. Mascitelli at (610) 765-5512.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of September 2009.

Respectfully,

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments: 1) Oyster Creek Nuclear Generating Station – Response to Request for Additional Information Regarding Control Rod Drive Notch Testing Frequency (TAC NO. MD8932)

cc: S. J. Collins, Administrator, USNRC Region I
M. S. Ferdas, USNRC Senior Resident Inspector, Oyster Creek
C. Gratton, USNRC Project Manager, Exelon Fleet
G. E. Miller, USNRC Project Manager, Oyster Creek
Mayor of Lacey Township
P. Baldauf, Assistant Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection

<u>Attachment 1</u>

Oyster Creek Nuclear Generating Station – Response to Request for Additional Information Regarding Control Rod Drive Notch Testing Frequency

(TAC NO. MD8932)

The following is the question received in a letter from E. Miller (U.S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station-Request for Additional Information Regarding Control Rod Drive Rod Notch Testing Frequency (TAC NO. MD 8932)," dated August 28, 2009, followed by the corresponding Exelon response to the question.

Question 1:

The model safety evaluation referenced in the license amendment request evaluates a reduction of the frequency for performing control rod notch testing for fully withdrawn control rods while the proposed change for Oyster Creek would decrease the frequency for both fully and partially withdrawn control rods. Please provide justification for why the decreased frequency of performing control rod notch testing in accordance with SR 4.2.D for partially withdrawn control rods will adequately assure the necessary quality of the control rod system and its components. Additionally, please identify and discuss any differences in the frequency or method of performing scram time testing for fully and partially withdrawn control rods.

Response:

Exelon has reviewed the model safety evaluation and determined that the evaluation and analysis provided for the fully withdrawn control rod drives is applicable to the partially withdrawn Control Rod Drives (CRDs). The operating experience related to the changes in CRD performance provided additional justification to reduce the notch test frequency for the partially withdrawn control rods. There is no difference in the design, operation or maintenance between a partially and fully withdrawn CRD. The only minor difference between a partially and fully withdrawn CRD is that the index tube of a partially withdrawn CRD rests on the collet at a different position than it does for the fully withdrawn CRD. Pressure and flow required for CRD motion is independent of notch position. In regards to control rod drive design and operation, there is no significant difference between the Oyster Creek Nuclear Generating Station (OCNGS) reactor core design, structure and operation and the typical reactor core on which the model safety evaluation is based that would affect the analysis or assumptions. The maintenance practices and methodologies are consistent with GE Energy-Nuclear recommendations for both partially and fully withdrawn CRDs.

As noted in the model safety evaluation after a review of the accumulation of operating experience, the NRC concluded that no known CRD failures have been detected during the notch testing exercises. The NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system. In addition, the model safety evaluation identified no difference in the predicted crack growth rate of the Collet Retainer Tube (CRT) between partially and fully withdrawn control rod drives. As stated in the model safety evaluation, CRT crack growth is dependent on the functions of water oxygen level, conductivity, material sensitization and applied loads and not on the position of the CRD. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs for partially withdrawn control rods.

In regards to the scram time testing, there are no differences in the frequency or method of performing scram time testing for fully and partially withdrawn control rods. OCNGS Technical Specification 4.2.C.3 states that control rods shall be scram time tested "On a frequency of less than or equal to once per 180 days of cumulative power operation, for at least 20 control rods, on a rotating basis, with reactor coolant pressure greater than 800 psig." The OCNGS scram time testing procedure requires the CRD to be fully withdrawn prior to performing scram time testing. The primary assurance of scram system reliability is provided by scram time testing since it monitors the system scram operation, CRD structure and integrity, and the complete travel of the CRD, which CRD notch testing does not accomplish.

In addition, the NRC recently approved a similar license amendment request (Reference 1) for Nine Mile Point 1.

<u>Reference 1:</u>

ML090160353: Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment Re: Revision of Control Rod Notch Surveillance Test Frequency Using the Consolidated Line Item Improvement Process (TAC No. MD9539), dated February 11, 2009.