UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 22, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: CONTROL ROD DRIVE SYSTEM ROD NOTCH TESTING FREQUENCY (TAC NO. MD8932)

Dear Mr. Pardee:

The Nuclear Regulatory Commission has issued the enclosed Amendment No.275 to Renewed Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek), in response to your application dated June 9, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081620236), as supplemented by letters dated March 30, 2009, and September 4, 2009 (ADAMS Accession Nos. ML090890777 and ML092470521, respectively). The amendment revises Surveillance Requirement 4.2.D to decrease the frequency of performing control rod drive rod notch testing from weekly to once per 31 days.

A copy of our Safety Evaluation is enclosed and a Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

for

G. Edward Miller, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures:
1. Amendment No. 275 to Renewed DPR-16
2. Safety Evaluation

cc: Distribution via ListServ

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No.275
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC, dated June 9, 2008, as supplemented by letters dated March 30, 2009, and September 4, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-16 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 275 , are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements:

   This license amendment is effective as of the date of issuance, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: October 22, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 275

RENEWED FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 3	Page 3

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4.2-1	4.2-1

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, or special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate such byproduct, source, or special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Exelon Generation Company is authorized to operate the facility at steady-state power levels not in excess of 1930 megawatts (thermal) (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 275 are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) <u>Fire Protection</u>

Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated March 3, 1978, and supplements thereto, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. SDM shall be verified:
   1. Prior to each CORE ALTERATION, and
   2. Once within 4 hours following the first criticality following any CORE ALTERATION.

B. The control rod drive housing support system shall be inspected after reassembly.

C. The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

   1. For all control rods prior to THERMAL POWER exceeding 40% power with reactor coolant pressure greater than 800 psig, following core alterations or after a reactor shutdown that is greater than 120 days.

   2. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "a" or "b" as follows:

      a.1 Specifically affected individual control rods shall be scram time tested with the reactor depressurized and the scram insertion time from the fully withdrawn position to 90% insertion shall not exceed 2.2 seconds, and

      a.2 Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure prior to exceeding 40% power.

      b. Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure.

   3. 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.

D. Each withdrawn control rod shall be exercised at least once per 31 days. This test shall be performed within 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.275 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-16

EXELON GENERATION COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated June 9, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081620236), as supplemented by letters dated March 30, 2009, and September 4, 2009 (ADAMS Accession Nos. ML090890777 and ML092470521, respectively), Exelon Generation Company, LLC (Exelon or the licensee)[1] submitted a license amendment request for the Oyster Creek Nuclear Generating Station (OCNGS). The proposed amendment would revise the Technical Specification (TS) surveillance requirements (SRs) associated with control rod testing. Specifically, SR 4.2.D currently requires, in part, that:

> Each partially or fully withdrawn control rod shall be exercised at least once per week.

The proposed amendment would revise the above requirement to read as follows:

> Each withdrawn control rod shall be exercised at least once per 31 days.

The purpose of SR 4.2.D is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. This ensures that the control rod is not stuck and is free to insert on a scram signal.

The supplements dated March 30, 2009, and September 4, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 12, 2008 (73 FR 46928).

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[1] The license application was submitted by AmerGen Energy Company, LLC (AmerGen) and Exelon. Effective January 8, 2009, the license for OCNGS was transferred from AmerGen to Exelon. By letter dated January 9, 2009 (ADAMS Accession No. ML090120538), Exelon adopted and endorsed AmerGen docketed submittals that requested specific licensing actions for OCNGS.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated occurrences," states that "[t]he protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

Section 3.1 of the OCNGS Updated Final Safety Analysis Report (UFSAR) discusses conformance of the OCNGS design with the GDC. With respect to GDC 26, UFSAR Section 3.1.22 states, in part, that reactor shutdown by the control rod drive (CRD) system is sufficiently rapid to prevent fuel damage limits being exceeded during either normal operation or any operational transients. With respect to GDC 29, UFSAR Section 3.1.25 states, in part, that the high probability of correct protection system and reactivity control system response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance.

Section 50.36(c)(3) of 10 CFR Part 50 requires that TSs include SRs "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

3.0 TECHNICAL EVALUATION

3.1 Background

Control rods are components of the CRD system, which is the primary reactivity control system for the reactor. The CRD system, in conjunction with the Reactor Protection System, provides the means for reliable control of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, specified fuel design limits are not exceeded. In addition, the control rods provide the capability to maintain the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase if there were to be a malfunction in the CRD system.

The CRD system consists of a CRD mechanism (CRDM) by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical-hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube, welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism (which consists of the locking collet, collet piston, collet return spring and unlocking cam), provides the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

3.2 TSTF-475, Revision 1

As discussed in the licensee's application dated June 9, 2008, the proposed change to OCNGS SR 4.2.D is based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) change traveler, TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action." A notice of availability for this TS improvement was published in the *Federal Register* on November 13, 2007 (72 FR 63935). The notice included a model safety evaluation (SE) that may be referenced by licensees in plant-specific applications to adopt the TSTF-475 changes.

TSTF-475 contains the following changes to the STS: (1) revise the TS control rod notch surveillance frequency; (2) clarify the TS requirements for inserting control rods with one or more inoperable SRMs; and (3) clarify the applicability of the 1.25 surveillance test interval extension provision in SR 3.0.2. Due to differences between the STS and the OCNGS TSs, the licensee stated in its application dated June 9, 2008, that only the first change contained in TSTF-475 (i.e., revise the TS control rod notch surveillance frequency) is applicable for OCNGS. The licensee also stated in its application that the justifications included in the NRC staff's model SE for TSTF-475 are applicable to OCNGS and justify the proposed amendment.

As discussed in the model SE, the NRC staff approved the TSTF-475, Revision 1, proposal to revise the TS control rod notch surveillance frequency in the STS from 7 days to monthly based on the following considerations: (1) slow crack growth rate of the CRT; (2) improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) General Electric (GE) chemistry recommendations; and (5) no known CRD failures being detected during notch testing. The NRC staff concluded that this change would reduce the number of control rod manipulations, thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the high reliability of the CRD system. Specifically, the NRC staff's model SE for TSTF-475, Revision 1 stated that:

According to the BWROG [Boiling Water Reactor Owners Group], at the time of the first CRT crack discovery in 1975 each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

Subsequently, many BWRs [boiling-water reactors] have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

In response to the NRC staff RAIs [requests for additional information] and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and GE Nuclear Energy report, CRD Notching Surveillance Testing for Limerick Generating Station (CRDNST). The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the Technical Specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks have been observed in the final improved CRT design.

To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on [the] BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the US are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the IGSCC [intergranular stress-corrosion cracking] crack growth rates were evaluated, at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are also tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

3.3 OCNGS Proposed Change

TSTF-475, Revision 1, is only applicable in decreasing the notch test surveillance frequency for fully withdrawn control rods, as many licensees have already decreased the surveillance frequency for partially withdrawn control rods (i.e., prior to TSTF-475). The proposed change for OCNGS would apply to both partially withdrawn and fully withdrawn control rods. In its supplement dated September 4, 2009, Exelon stated that it had reviewed the model SE and determined that the evaluation and analysis provided for the fully withdrawn control rods is applicable to the partially withdrawn control rods. The NRC staff has also reviewed the model SE and determined that the justifications for reducing control rod notch surveillance frequency are also applicable to partially withdrawn control rods.

Based on the justifications provided in the model SE for TSTF-475, the NRC staff concludes that the proposed change to OCNGS SR 4.2.D will reduce the number of control rod manipulations, thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the high reliability of the CRD system. Therefore, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (73 FR 46928). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Edward Miller
R. Ennis

Date: October 22, 2009

October 22, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: CONTROL ROD DRIVE SYSTEM ROD NOTCH TESTING FREQUENCY (TAC NO. MD8932)

Dear Mr. Pardee:

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Sincerely,
/ra/
G. Edward Miller, Project Manager
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Docket No. 50-219

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