October 10, 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 - ISSUANCE OF AMENDMENT RE: REFUELING INTERLOCKS (TAC NO. MB2893)

Dear Mr. Skolds:

The Commission has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station in response to your application dated September 11, 2001, as supplemented on June 27 and September 19, 2002.

The amendment revised the Technical Specifications, Section 3.9, "Refueling," and its corresponding bases to permit the continuation of core alterations during refueling operations with the refueling interlocks inoperable by providing alternate actions which will preserve the intended design function of the inoperable interlocks.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 234 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 234
License No. DPR-16

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

   A. The application for amendment by AmerGen Energy Company, LLC, et al., (the licensee), dated September 11, 2001, as supplemented on June 27 and September 19, 2002, 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 Facility Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234, are hereby incorporated in the license. AmerGen Energy Company, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 10, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.9-1	3.9-1
3.9-3	3.9-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234

TO FACILITY OPERATING LICENSE NO. DPR-16

AMERGEN ENERGY COMPANY, LCC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated September 11, 2001, AmerGen Energy Company, LLC, (AmerGen or the licensee) submitted an application to amend the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TSs). By letter dated June 27 and September 19, 2002, AmerGen supplemented the application. The June 27 and September 19, 2002, letters provided clarifying information within the scope of the original application and did not change the Nuclear Regulatory Commission (NRC) staff's initial proposed no significant hazards consideration determination.

The proposed amendment would revise the TSs Section 3.9, "Refueling," and its corresponding bases to permit the continuation of core alterations during refueling operations with the refueling interlocks inoperable by providing alternate actions which will preserve the intended design function of the inoperable interlocks.

The refueling interlocks (the all-rod-in (ARI), one-rod-out, refueling platform position, refueling platform main hoist, and service platform hoist fuel loaded) are designed to physically minimize the possibility of reactivity-initiated events by restricting combinations of fuel movements and control rod withdrawals. With the reactor mode switch in the "refuel" position, the refueling equipment interlocks receive and process signals from various sources to block control rod movement or operation of the fuel-loading equipment. The all-rods-in interlock receives and processes full-in position indications from all the control rods and gives an ARI permissive signal for the operation of the refueling platform, main hoist grapple, and service platform. The refueling platform position interlock de-energizes if there is no ARI permissive and the platform is near or over the core. The refueling platform main hoist interlock also de-energizes if there is no ARI permissive signal and the interlock detects loading indicative of a fuel assembly. The service platform hoist also de-energizes if there is no ARI permissive signal and the interlock detects loading indicative of a fuel assembly. Therefore, the refueling equipment interlock logic combines the ARI permissive signal, the position indication of the refueling platform, and the loading of the main hoist grapple and the service platform to prevent fuel movement over the core if all control rods are not inserted and blocks all control rods withdrawals if fuel movement is in progress.

The reactor core is designed with sufficient shutdown margin to ensure that the core will remain subcritical with the highest worth control rod withdrawn to its full-out position. With one control rod withdrawn, the one-rod-out interlock prevents the selection or withdrawal of a second control rod. The refueling limiting conditions for operation (LCOs) in Section 3.9 enforce the functions of these refueling interlocks, since these interlocks are design-basis assumptions intended to preclude fuel loading and control rod withdrawal errors.

The OCNGS refueling LCO 3.9 specifies the reactivity management and controls for refueling operations involving (1) fuel movements (3.9.A, B, C, and D), (2) single control rod or control rod drive (CRD) removal for maintenance (LCO 3.9.E), and (3) multiple control rods or CRDs removal for maintenance (LCO 3.9.F). Specification 3.9.G requires that fuel handling or control rod removal activities cease if the applicable requirements cannot be met. AmerGen also proposed to add an alternative requirement to Specification 3.9.C.

2.0 <u>REGULATORY EVALUATION</u>

General Design Criterion (GDC) 26, “Reactivity control system redundancy and capability,” of Title 10 of the *Code of Federal Regulations,* Part 50 (10 CFR Part 50), Appendix A, states that one of two reactivity control systems must be capable of holding the reactor subcritical under cold conditions. The control rods, when fully inserted, serve as the system capable of maintaining the reactor subcritical in cold conditions during all fuel movement activities and accidents. Instead of analyzing the possible reactivity-initiated events or their radiological consequence, General Electric designed the refueling interlocks to prevent inadvertent reactivity-initiated events. Section 15.4 of the OCNGS Updated Final Safety Analysis Report (UFSAR) assumes that the refueling interlocks are functioning and will prevent reactivity-initiated events.

NUREG-1433, “Standard Technical Specifications General Electric Plants, BWR/4, (Revision 2),” was developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Reactors, dated July 22, 1993, which was subsequently codified by changes to 10 CFR 50.36. The NUREG specifies the LCO for operation for the refueling interlocks along with its associated basis. Licensees adopting portions of the Improved Standard Technical Specifications (ISTS) to existing TSs should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

3.0 <u>TECHNICAL EVALUATION</u>

The refueling interlocks, when operable, impose barriers to preclude an inadvertent criticality during refueling operations. Inadvertent criticality is precluded by preventing: (1) the operation of loaded refueling equipment (fuel grapple hoist, frame-mounted auxiliary hoist, trolley-mounted auxiliary hoist, and the refueling bridge) over the core when any control rod is withdrawn, or (2) withdrawal of any control rod when fuel-loaded equipment is operating over the core. In addition, when the reactor mode switch is in refuel position, only one rod can be withdrawn, and selection of a second rod would initiate a rod block.

Currently, TSs Section 3.9, “Refueling,” states that during core alterations the reactor mode switch is locked in the refuel position and control rods or rod drive mechanisms cannot be removed unless all the refueling interlocks are operable as required in Section 3.9.C. Section

3.9.C stipulates that interlocks for the grapple hoist, frame-mounted auxiliary hoist, trolley-mounted auxiliary hoist or the service platform hoist must be operable to assure that criticality does not occur during refueling.

3.1 Proposed Changes

The licensee proposed to add the following compensatory measures to Section 3.9.C:

> Fuel handling operations with the head off the reactor vessel can be performed with the refueling interlocks inoperable provided all the following specifications are satisfied:
>
> 1. All control rods are verified to be fully inserted.
>
> 2. Control rod withdrawal has been disabled.

The licensee proposed to add the following explanation to Basis 3.9:

> The refueling interlocks may be inoperable provided that all 137 control rods are verified to be fully inserted and control rod withdrawal has been disabled prior to commencing or recommencing fuel handling operations with the head off the reactor vessel. This will ensure that all control rods remain fully inserted during fuel handling operations with the head off the reactor vessel. Therefore, Specification 3.9.A is met and the core will remain subcritical during fuel handling operations.

3.2 Core Criticality Concerns

Specification 3.9.A requires that "fuel shall not be loaded into a reactor core cell unless the control rod in that core cell is fully inserted." With the refueling interlocks inoperable, the licensee states that the administrative controls (verifying all control rods are inserted and electrically or hydraulically disabling control rod withdrawals) will provide an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn.

Core physics calculations indicate that the creation of two loaded adjacent uncontrolled fuel cells may result in prompt critical conditions. The condition of two loaded uncontrolled fuel cell (LUFC) can be created by an inadvertent control rod withdrawal adjacent to a loaded uncontrolled fuel cell, and inadvertent loading of fuel into defueled uncontrolled fuel cells can also result in LUFCs. The proposed wording to Specification 3.9.C will provide:

1. "All control rods are verified to be fully inserted." This requirement ensures all 137 control rods are fully inserted prior to loading fuel, thus preventing the possibility of inadvertently loading fuel into defueled uncontrolled fuel cells.

2. "Control rod withdrawal has been disabled." This requirement prevents conducting two activities that affect reactivity at the same time, and it also minimizes the probability of inadvertently withdrawing control rods from loaded fuel cells (i.e., creating LUFC). Disabling control rod withdrawals after all control rods are verified to be inserted ensures that LUFCs do not occur.

The NRC staff agrees with the licensee that these two required conditions to be added to Specification 3.9.C will ensure safety protection equivalent to operable interlocks. However, the proposed change will replace an automatic ARI permissive feature with manual verification that all control rods are inserted, which is subject to human error. In the June 27, 2002, supplement the licensee described the activities involved in implementing the alternative compensatory actions. Before bypassing the ARI permissive circuitry, the operators will verify that the control rod position indication for each individual rod (Panel 4F) show that all control rods are fully inserted. Upon confirmation that all rods are at their full-in position, control rod withdrawal will be physically disabled. Specifically, the licensee will use the proposed alternative option if (1) the ARI permissive fails, or (2) it becomes necessary to work on the position indication probe (PIP), which may falsely indicate a withdrawn control rod.

The NRC staff asked the licensee to explain how it will positively verify that all control rods are inserted if the ARI permissive signal is lost or a PIP that provides the control rod position indication becomes inoperable. In the September 19, 2002, supplement the licensee stated:

> The all-rods-in (ARI) signal is produced by the Reactor Manual Control System (RMCS) using the full-in position switch from each control rod position indication probe (PIP). If any one of the full-in switches were to fail to actuate, the ARI signal would be lost. Each full-in switch also actuates a green back-lighting for the associated control rod on the full core control rod position display located on control room panel 4F. Each PIP contains a switch for position “00,” adjacent to the full-in switch, that provides an alternate indication that the control rod is fully inserted. The control rod position is displayed on the full core control rod position display located on control room panel 4F, and is also input to the Rod Worth Minimizer and passed on to the Plant Computer System and Core Monitoring Computer. The green back-lighting and the “00" position indication provide redundant indications that a control rod is fully inserted.

The NRC staff accepts that the “full-in” and the “00” reed switches provide adequate redundancy, unless a PIP failure leads to loss of all control rod position signals. In this circumstance, the licensee can confirm that the control rod (with the inoperable PIP) is inserted visually or by using video camera.

The NRC staff evaluated the integrated refueling activities that would be allowed if the proposed amendment is implemented. For multiple control rod or control rod drive (CRD) removal, Specification 3.9.F requires all of the refueling interlocks to be operable. Since the proposed change to Specification 3.9.C requires all control rods to be inserted, multiple control rod removal will not be permitted when refueling interlocks are inoperable. For single control rod or CRD removal, Specification 3.9.E requires the reactor mode switch to be locked in the refuel position (activating the refueling interlocks), but does not explicitly require the interlocks to be operable. However, since all 137 control rods must be inserted per Specification 3.9.C, the licensee likewise cannot perform single control rod withdrawal or removal operation concurrent with fuel loading. Thus, the NRC finds that while the proposed change result in operating flexibility, it continues to ensure safe reactivity management during refueling.

3.3 Operability of Interlocks during Refueling Operations

The NRC staff was concerned that the compensatory measures would obviate the need for restoring the interlocks to an operable status and the need to perform the specified surveillance. In the June 27, 2002, supplement, the licensee stated that the proposed change would still require the performance of the initial Section 4.9.A surveillance prior to in-vessel fuel movement. Also, the refueling interlocks would be operable during fuel moves except for an unexpected equipment failure or during maintenance that would otherwise result in false indications of rod withdrawal. In such cases, the rods will be verified as fully inserted and rod withdrawal prevented. To clarify the intent of the proposed amendment, the licensee proposed that the following paragraph be added to the TS bases:

> It is not the intent of the alternative option in Specification 3.9.C to eliminate the first performance of Technical Specification Surveillance 4.9.A prior to in-vessel fuel movement. It is expected that the refueling interlocks would be operable during fuel moves except for equipment failures or during maintenance that would otherwise result in false indications of rod withdrawal during which all rods will be verified as fully inserted and rod withdrawal prevented.

In the September 19, 2002, letter the licensee provided further clarification with respect to the operability of the refueling interlocks and performance of the required surveillances. The licensee stated that when the ARI permissive is inoperable or disabled in accordance with the proposed change, the refueling interlock will not be used during refueling and Section 4.9.A would not be applicable until the ARI permissive is restored. Consequently, the restrictive nature of the compensatory measures would not allow the option of withdrawing individual control rods in defueled cells as provided by the existing Section 3.9.F. As a result, refueling interlocks must be first restored to operable status when Section 3.9.F applies. Thus, Surveillance 4.9.A is applicable in this condition. Therefore, the NRC staff finds the bypassing of the ARI permissive and compensatory measures acceptable because the refueling interlocks are required to be operable in order to meet Section 3.9.F and may not remain inoperable during the entire refueling outage.

3.4 Summary of Technical Evaluation

Based on the preceding discussions, the NRC staff finds that bypassing of the ARI permissive and the compensatory measures satisfy the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) and NUREG-1433 for refueling interlocks. As a result, inadvertent criticality is prevented by the licensee when the compensatory measures are implemented prior to bypassing the ARI permissive. In addition, the licensee has committed to revise the Reactor Refueling Procedure No. 205.0 and the Rod Withdrawal/Insertion During Refueling Procedure No. 205.5 to incorporate the specified compensatory actions. Further, the corresponding Section 3.9 Bases will be revised to reflect the TS changes.

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 use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (67 FR 10008). 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 needs 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: Z. Abdullahi
G. Hatchett

Date: October 10, 2002

Oyster Creek Nuclear Generating Station

cc:
Chief Operating Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Senior Vice President - Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Mid-Atlantic Operations Support
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Senior Vice President -
Mid Atlantic Regional Operating Group
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Kevin P. Gallen, Esquire
Morgan, Lewis, & Bockius LLP
1800 M Street, NW
Washington, DC 20036-5869

Kent Tosch, Chief
New Jersey Department of Environmental Protection
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

Vice President -
Licensing and Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Site Vice President
Oyster Creek Nuclear Generating Station
AmerGen Energy Company, LLC
PO Box 388
Forked River, NJ 08731

H. J. Miller
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 445
Forked River, NJ 08731

Director - Licensing
Exelon Generation Company, LLC
Correspondence Control Desk
P.O. Box 160
Kennett Square, PA 19348

Oyster Creek Generating Station Plant Manager
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

Regulatory Assurance Manager
Oyster Creek Nuclear Generating Station
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

Vice President, General Counsel and Secretary
Exelon Generation Company, LLC
300 Exelon Way
Kennett Square, PA 19348

J. Rogge, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Manager Licensing - Oyster Creek and Three Mile Island
Exelon Generation Company, LLC
Nuclear Group Headquarters
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

Correspondence Control Desk
Exelon Generation Company, LLC
200 Exelon Way, KSA 1-N-1
Kennett Square, PA 19348