October 18, 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: USE OF OPTION II SOLUTION FOR REACTOR INSTABILITY PROBLEM (TAC NO. MB4960)

Dear Mr. Skolds:

The Commission has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station in response to your application dated April 26, 2002, as supplemented on July 11 and September 12, 2002.

The amendment revised Sections 2.3, “Limiting Safety System Settings,” 3.1, “Protective Instrumentation,” and 3.10, “Core Limits,” of the Technical Specifications, and approved the use of flow control reference cards to support implementation of the Boiling Water Reactor Owners Group Option II solution for the long-term reactor stability problem.

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. 235 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. 235
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 April 16, 2002, as supplemented on July 11 and September 12, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 235, 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 18, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 235

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
2.3-1	2.3-1
2.3-2	2.3-2
2.3-4	2.3-4
2.3-5	2.3-5
2.3-6	2.3-6
3.1-1	3.1-1
3.1-7	3.1-7
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235

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 April 26, 2002, AmerGen Energy Company, LLC, (AmerGen or the licensee) submitted an application to amend the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TSs). By letters dated July 11 and September 12, 2002, AmerGen supplemented the application. The July 11 and September 12, 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 support implementation of the Boiling Water Reactor (BWR) Owners Group (BWROG) Option II solution for the long-term reactor stability problem. The licensee planned to use a new type of fuel after the fall 2002 refueling outage (RFO). To support implementation of the Option II solution along with new fuel and to allow plant operation in the previously NRC-approved extended load line limit analysis (ELLLA) region (see Enclosure 4 of the September 12, 2002, submittal), the licensee proposed to revise the setpoints of the average power range monitor (APRM) flow-biased neutron flux scram function and the control rod block function, and implement these changes using digital APRM Flow Control Reference (FCTR) Cards. In addition, associated Bases sections will be revised to reflect the proposed TS changes, and a few administrative changes will be made. The proposed changes to the TSs support Fuel Cycle 19 operation to incorporate a more conservative stability analysis solution appropriate for the use of General Electric (GE) 11 fuel design, and to improve operational flexibility.

The NRC staff's evaluation follows, as supported by a contractor's technical evaluation report (TER), attached.

2.0 REGULATORY EVALUATION

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) provides that nuclear plant TSs will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34 (which addresses, among other things, contents of the Final Safety Analysis Report (FSAR)). The existing TSs requirements, as well as the licensee's proposed amendment, are based on such analyses and evaluations.

Appendix A of 10 CFR Part 50, General Design Criteria (GDC) 10 requires that the reactor core and associated coolant, control, and protective systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated occurrences. GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to ensure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably detected and suppressed. GDC 13 requires instrumented variables to be monitored over anticipated ranges for normal and accident conditions. GDC 20 requires the protection system to be designed to automatically initiate to assure that fuel design limits are not exceeded.

Also, 10 CFR 50.36 requires that safety-related instrument setpoints be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. Regulatory Guide (RG) 1.105 Revision 2 endorses the Instrument Society of America (ISA) Standard 67.04, 1982, as an acceptable guide for setpoint calculation methodology. In addition, RG 1.105, Rev. 2, requires that the setpoint calculation methodology shall be based on 95/95 confidence level requirement. In the July 11, 2002, response to the NRC staff's request for additional information (RAI), the licensee confirmed that its in-house setpoint calculation methodology is based on ISA Standard 67.04, 1982, and meets the 95/95 confidence level requirement of RG 1.105, Rev. 2. Therefore, the licensee's setpoint calculation methodology is acceptable.

The licensee requested the amendment in accordance with the provisions of 10 CFR 50.90.

3.0 TECHNICAL EVALUATION

The possibility of power oscillations due to thermal-hydraulic (T-H) conditions in BWRs and the consequences of such events have been of concern for many years. To address these concerns, the BWROG initiated a project to investigate actions that should be taken to resolve the BWR stability issue. In 1988, in a letter to BWR owners, General Electric Corporation (GE) recommended interim corrective actions (ICAs). The NRC staff issued Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," approving the proposed BWROG/GE ICAs to detect and suppress BWR power oscillations.

The BWROG later issued the following topical reports: (1) NEDO-32645-A, "BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996; (2) NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995; (3) NEDO-31960-A, Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995; and (4) NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Application," August 1996. The NRC staff reviewed these topical reports and found four of the six long-term solutions recommended by these topical reports to be acceptable for mitigating T-H instability events. The acceptable solutions are Options I-A and I-D for regional exclusions, Option II for BWR/2 designs, and Option III and III-A to detect and suppress functions. The OCNGS reactor is a BWR/2 design and, for this design, the topical reports recommend the Option II solution as the most appropriate solution for alleviating the T-H instability concern.

The current long-term stability solution employed at OCNGS is based on BWROG stability analyses and methodologies reported in GE NEDO-31708, “Fuel Thermal Margin During Core Thermal Hydraulic Oscillations in a Boiling Water Reactor,” June 1989, and relies on the existing flow-referenced APRM neutron flux scram setpoint to provide fuel protection. The current system, which was approved by the NRC staff via Amendment No. 176 in December 1994, assumes an APRM neutron flux scram setpoint of 78% at 30% core flow. The current fuel design at OCNGS is based on an 8x8 fuel rod array, and the NEDO-31708 analysis for the current system was also based on fuel designs with an 8x8 fuel rod array (e.g., using GE8 and GE9 fuel types). In its application, the licensee stated that beginning with Cycle 19, OCNGS will use GE11 fuel instead of GE8 or GE9. The GE11 fuel uses a 9x9 fuel rod array.

GE’s evaluation of the proposed Option II solution with the new type of fuel demonstrates that the existing quadrant-based APRM trip system at OCNGS will (1) initiate a reactor scram for a postulated reactor instability, and (2) will not encroach upon the minimum critical power ratio (MCPR) safety limit, provided setpoints for the APRM flow-biased neutron flux scram and the rod block functions are revised. The setpoints for these functions will be revised to limit the oscillation magnitude at reactor trip, thereby limiting the associated critical power ratio (CPR) change, and to assure design-compliance with the MCPR safety limit in conjunction with MCPR operating limits. Similar changes (with Option II) were reviewed and approved by the NRC for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), via Amendment No. 168.

The licensee stated that implementation of the proposed changes for Cycle 19 startup after RFO 19 will (1) incorporate a more conservative stability analysis solution appropriate for the use of GE11 fuel in Cycle 19, and (2) improve operational flexibility, and reduce the number and/or magnitude of reactor maneuvers, thereby improving the capacity factor and reducing fuel and challenges to operators and safety system equipment. Additionally, the availability of lower core flow rates will enhance spectral shift operation and, therefore, will reduce fuel costs.

The licensee also proposed a digital system upgrade to a portion of the reactor protection system (RPS). To implement the scram setting changes, the licensee proposed to replace 8 analog APRM Flow Bias Trips units with digital APRM FCTRs.

3.1 <u>Proposed TSs Changes</u>

The licensee proposed to revise TSs pages 2.3-1, 2.3-2, 2.3-4, 2.3-5, 2.3-6, 3.1-1, 3.1-7, 3.10-2, 3.10-3, and 3.10-4. These changes will incorporate the BWROG-defined stability Option II solution methodology to assure that coupled neutronic and T-H instabilities are adequately detected and suppressed. Specifically, the changes would:

(1) Increase current flow-biased APRM scram setting by 4.3% and control rod block setting by 7% for transient protection with $W \geq 0.0\times10^{6}$ lb/hr;

(2) Add APRM flow-biased scram setting $S \leq (0.98\times10^{-6})W+41.4$ and control rod block setting $S \leq (0.98\times10^{-6})W+34.1$ with $W \leq 27.5\times10^{6}$ lb/hr;

(3) Increase maximum scram setpoint by 4.3% to 120% and to increase maximum rod block setpoint by 7% to 115% for core flow equal to 61×10^{6} lb/hr and greater;

(4) Revise the associated Bases sections.

3.2 Evaluation of Proposed Changes

The proposed Option II solution is basically a quadrant-based APRM trip system with a recirculation flow-biased trip. In the application, the licensee stated that the current quadrant-based APRM system at OCNGS is unique to BWR/2 designs in that local power range monitor (LPRM) instrument assignments to the APRMs are arranged in separate quadrants of the reactor. The current system at OCNGS relies on the existing flow-referenced APRM neutron flux scram setpoint to provide fuel protection, and was approved by the NRC in 1994. GE evaluated the plant-specific application of the Option II solution to the OCNGS design in NEDC-33065P, Rev. 0, "Application of Stability Long-Term Solution Option II to Oyster Creek," April 2002, for Cycle 19 with the new GE11 fuel. The results of the GE evaluation demonstrated that the current quadrant-based APRM trip systems will initiate a reactor scram for a postulated reactor instability and will not violate the MCPR safety limit; therefore, at OCNGS, the Option II solution can be achieved with the current LPRM, APRM, and recirculation flow signals and algorithms, by revising the setpoints and LSSSs for the TS APRM flow-biased neutron flux scram and for the rod block functions. Therefore, the licensee proposed to revise setpoints for these functional units. To ensure that the setpoint derived in NEDC-33065P will always prevent exceeding the MCPR safety limits, the licensee proposed to review these limits at each fuel reload. The licensee stated that the MCPR operating limits at OCNGS will be maintained in the core operating limits report (COLR). In addition, NEDC-33065P documented GE's calculation of the revised restricted-region boundary to be implemented at OCNGS for the Option II solution.

In BWRs, reactor power level can be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block signal to prevent rod withdrawal beyond a given point at a constant recirculation flow rate. The trip setpoint for the rod block function is also varied automatically with the recirculation flow rate (per the setpoint-versus-flow relationship), thereby preventing an increase in the reactor power level to excessive values due to control rod withdrawal. For the specified setpoint-versus-flow relationship, available margin to the safety limit increases as the recirculation flow decreases. The current APRM system at OCNGS consists of eight identical channels. Each APRM channel gets input from eight LPRMs, which enables it to compute an accurate average thermal neutron flux in a core quadrant. APRM trip signal outputs are utilized in the reactor protection system (RPS) for the reactor scram signal and in the reactor manual control system to generate control rod block signals. The averaging and trip circuits of APRMs receive the recirculation flow signal to vary the trip setpoints (for both the scram and the rod block functions) in accordance with a specific relationship between the quadrant average power and the total recirculation flow.

The NRC staff has reviewed NEDC-33065P, Rev. 0, and has found it acceptable because: (1) the initial MCPR values used are consistent with the approved detect-and-suppress methodology; (2) the implementation of a methodology to determine the magnitude of the peak fuel bundle power oscillation follows the approved Nine Mile Point methodology (reference GENE-A13-00360-02, "Application of Stability Long Term Solution Option II to Nine Mile Point Nuclear Station Unit 1," General Electric Company, August 1995 and ORNL/NRC/LTR-96/16, "Review of the Application of Stability Long Term Solution II to Nine Mile Point Unit 1," Jose March-Leuba, ORNL Letter Report, May 1996) which deviates slightly from other approved detect-and-suppress methodologies (i.e., Solution III and I-D) ; (3) the final MCPR value is determined using a generic correlation known as the DIVOM curve, which is subject to a 10 CFR Part 21 report for other options, but is still valid for OCNGS application; (4) an administratively controlled exclusion region is required to minimize the probability of startup instabilities and to satisfy the 100%-rod-line initial-condition assumption in the analysis; and (5) the reload confirmation procedures defined in Section 7 of NEDC-33065P, Rev. 0, are consistent with other reviewed and approved long-term solutions (Reference 7). The detailed evaluation is given in the attachment by the consultant from Oak Ridge National Laboratory.

3.2.1 Changes to Section 2.3, “Limiting Safety System Settings”

GE’s plant-specific analysis (NEDC-33065P, Rev. 0) for application of Option II at OCNGS concluded that for use of the new GE11 fuel in Cycle 19, the setpoint and LSSS of the APRM flow-biased neutron flux scram function must be revised to provide automatic protection while ensuring that anticipated T-H instabilities will not compromise the established fuel limits. Also, the setpoint and LSSS of the APRM flow-biased rod-block functions must be changed to be consistent with the revised scram settings with the new type of fuel. Therefore, the licensee is proposing to revise the setpoints and LSSSs values for these functional units. The licensee stated that the proposed setpoint changes will increase the APRM neutron flux scram and rod block setpoints, allowing full-power operation at a core flow as low as 85% of the rated flow. The proposed changes will result in more restrictive trip settings for the APRM flow-biased scram function in the low-flow regions of the power/flow operating map, the region where reactor instabilities are most probable. The new settings will provide the reactor scram signal sooner (i.e., at a lower power range) than the current settings, are therefore conservative, and will meet GDC 10 and GDC 12 for fuel design protection. The margins between the control rod block and neutron flux scram setpoints are based on the recommendations contained in the GE analysis.

The licensee stated that, for the setpoint calculations, previously applied conservatism in the actual setpoints are still maintained to account for instrument drift, uncertainties, and typical APRM process noise. The proposed setpoint revisions do not change the “maximum licensed power line.” To ensure that the maximum licensed power line is never encroached, operation within the upper boundary of the extended operating region will be controlled by administrative core-monitoring requirements for Cycle 19 and future cycles. The revised LSSSs will improve plant operations by allowing full utilization of the previously licensed OCNGS core power-flow operating domain, without compromising the plant safety.

The licensee stated that, except for the control rod drop accident (CRDA), credit for the APRM neutron flux scram function or for the rod block function has not been taken for any design-basis accidents (DBAs) which are included in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). The licensee evaluated the impact of the proposed setpoint and LSSS changes on the following DBAs and events: (1) the CRDA, (2) the rod withdrawal event (RWE), (3) the loss of feedwater heating (LOFH) event, (4) the main steam line isolation valve closure (MSLIVC) event, and (5) the startup-of-an-inactive-loop-at-an-incorrect temperature event.

The licensee’s evaluation determined that for the CRDA event and the MSLIVC event, the proposed 120% increase in the scram setting will not have any adverse impact. This is because the original generic evaluations for these events (the results of which are included in Chapter 15 of the UFSAR) were based on an assumed setpoint which was 120% of its nominal value. The licensee stated that for the MSLIVC event the proposed setpoint does not consider an additional 1% conservatism (assumed in the original analysis) to address impact of other transients although it is consistent with the original analysis. Excluding this 1% difference will introduce a slight actuation delay (i.e., actuation will be at 121% instead of 120%), which will result in a small increase in the calculated peak reactor pressure. But, because the resulting peak pressure will still be within the available pressure margin assumed in the original MSLIVC event analysis, the proposed change will not have any adverse impact for the MSLIVC event and therefore is acceptable. For the other remaining events, the evaluation concluded that the impact of the proposed setpoint and LSSS changes were bounded by the existing reload analyses and would not adversely impact the existing core thermal operating limits as described in the COLR. The combined results of these evaluations demonstrated that because no fuel thermal limits or other licensing basis acceptance criteria are adversely affected, the proposed changes are acceptable. Furthermore, the changes do not physically modify or change the functions, system interfaces, or components. Therefore, the probability of any accident previously evaluated will not change. Operation in the ELLLA region was approved by the NRC staff via Amendment No. 111, dated October 27, 1986. But the current LSSSs values for the APRM scram and rod block functions do not permit full utilization of this previously licensed power-flow operating domain because the effective flow window has to have a minimum core flow, which is as high as approximately 93% of rated flow at full reactor power. The revised LSSSs will improve plant operations by allowing full utilization of the licensed core power-flow operating domain. With the revised LSSS settings, full-power operation would be achieved with core flow as low as the minimum licensed value (i.e., 85% instead of 93% of rated), and the resulting increased flow window will improve operational flexibility and reduce the number and/or magnitude of reactor maneuvers, thereby improving the capacity factor and reducing challenges to fuel, equipment, and operators. In addition, the ability to operate at reduced core flow rates can improve fuel utilization, and the availability of lower core flow rates will enhance flow spectral shift operation, thereby reducing fuel costs.

The Option II solution is documented in NEDO-32465-A which has been previously reviewed and approved by the NRC. The proposed incorporation of the Option II solution to detect and suppress T-H instability in the OCNGS reactor core is consistent with the NRC-approved NEDO-32465-A. The NRC staff had previously approved a similar change for NMP1, via Amendment No. 168 on September 21, 1999. Accordingly, the proposed incorporation of the Option II solution and the proposed changes to setpoint and LSSS of the flow-biased APRM neutron flux scram and the rod block functions are acceptable. The NRC staff also reviewed the proposed changes to the TSs Bases and found these changes consistent with the TSs changes.

The licensee stated that to implement the proposed setpoint changes, it planned to replace eight analog APRM flow bias trip units with new digital APRM FCTR cards. These cards can be designed to implement a multi-sloped curve of scram and rod block setpoints based on input feedwater flow. This constitutes an analog to digital change to a portion of the RPS. The licensee proposed to install these digital FCTR cards under the provisions of 10 CFR 50.59. The NRC staff believes that these cards should not be installed under 10 CFR 50.59 since there is no consensus method for software reliability, specifically its failure probability, and since failure modes may exist that can create results different from those previously evaluated in the UFSAR. Therefore, a licensing amendment would normally be required for this type of change. However, as a result of supplemental information provided by the licensee, and an on-site review, the NRC staff received sufficient information concerning the use of these cards to determine that the use of the GE Option II FCTR cards is acceptable to implement the scram setpoints and changes specified in the license amendment. The reasons for acceptance are set forth below.

The NRC staff accepted the use of identical digital FCTR cards for BWR coupled neutronic/thermal-hydraulic instabilities in a February 12, 1998, safety evaluation (for NEDC-32339P, Supplement 2, “Reactor Stability Long-Tern Solution: Enhanced Option I-A Solution Design”), and also plant-specifically for NMP1 (Amendment No. 168). In an RAI dated June 17, 2002, the NRC staff questioned the FCTR cards’ plant-specific environmental qualification, their operating history at OCNGS, the differences if any between the proposed cards and those previously approved by NRC staff and the response to the plant-specific comments discussed in the February 1998 safety evaluation.

The licensee stated that the new digital FCTR cards are designed to meet appropriate environmental qualification parameters which envelop the plant-specific environmental conditions for temperature, humidity, pressure, seismicity, and electro-magnetic compatibility. The licensee responded to the plant-specific item of the February 1998 safety evaluation regarding design-basis environmental conditions and stated that they are enveloped by the required equipment qualification values. Having reviewed on-site documentation and the licensee’s September 12, 2002, response, the NRC staff finds the environmental qualification to be acceptable.

The licensee stated that at the end of Cycle 18 operation, personnel will perform a shakedown and burn-in process for the new digital FCTR cards to identify any plant-specific revisions that may be required to prevent future spurious trips and alarms. The licensee stated in its September 12, 2002, response that the results of this process will be reported to NRC when complete.

The licensee stated that the proposed FCTR cards are identical to the ones previously approved by the NRC for NMP1. However, there are some plant-specific differences which relate to output power range and the type of electronically programmable read-only-memory (EPROM) used in these cards. Since the EPROM contains the scram setpoints that correspond to recirculation feed flow, the NRC staff believed that a method should be in place that would ensure the correct scram setpoints have been programmed in the FCTR EPROM. The licensee stated that the EPROMs would be programmed by the vendor, GE, and delivered to the site. Since the cards will implement a multi-sloped curve, the licensee planned to test an adequate number of test points to ensure proper functionality in the different regions. The EPROM contains binary information that corresponds to the scram and rod block setpoints, hence the NRC staff was interested in determining the method that verifies that the binary information on the EPROM is the same as delivered by GE and that will be in use in OCNGS. In the on-site review, the NRC staff examined the method to be used to ensure this. The licensee stated it will compare the checksum from the cyclic redundancy check (CRC) performed at GE with the resultant checksum that is output from the in-the-plant installed FCTR card to confirm that the binary information is the same as delivered. As the card operates, it will continually and periodically perform a CRC self-test. The NRC staff believes that this check and the checking of points to verify proper feedwater versus setpoint slope check is an acceptable means to verify the proper EPROM information. Accordingly, the NRC staff finds the FCTR GE Option II cards model 148C7112G002 to be acceptable for use to implement the existing TS and proposed changes.

Section 5, Items 3 and 4 of the February 1998 safety evaluation related to administrative controls for manually bypassing channels/protective functions and plant-specific human factors review for changes to the operator control panels. The licensee stated that administrative controls remain unchanged from those previously in place for the APRM trip units. The changes impacting human factors are minimal and the licensee stated that human factors review is not required. In the on-site review, the NRC staff discussed these changes with the licensee and found the plant-specific actions to be acceptable.

3.2.2 Changes to Section 3.1, “Protective Instrumentation”

The licensee proposed to delete Specification 3.1.B.3 and its associated Bases 3.1, which was added by Amendment No. 176, dated December 29, 1994. The NRC staff reviewed the proposed changes and found them acceptable because the licensee has accepted GE’s recommendation for OCNGS transition to the Generic BWROG Long Term Stability Solution Option II for use of GE11 fuel in operating Cycle 19. The GE recommendation addresses LPRM operability in a different manner and does not require these operability constraints.

3.2.3 Changes to Section 3.10.C, “Minimum Critical Power Ratio (MCPR)”

The licensee proposed to remove the MCPR operating limits to the COLR, revising its associated Bases 3.10 to reflect the TS revision and adding a statement for the OCNGS MCPR flow multiplier, k_f , on page 3.10-4 to limit MCPR when operating at low flow conditions.

The NRC staff reviewed these proposed changes and found them acceptable because MCPR operating limits will be reviewed each reload and will be maintained in the COLR to assure the setpoint derived in NEDC-33065P will prevent the MCPR safety limits from being exceeded.

3.2.4 Summary of Evaluation

Based on the above evaluation, the NRC staff agrees that implementation of the Option II solution will enhance safety by adequately detecting and suppressing the coupled neutronic T-H instabilities, thereby alleviating the reactor long-term stability concern. The Option II solution is documented in NEDO-32465-A, which was previously reviewed and approved by the NRC staff. The licensee’s setpoint calculation methodology is acceptable because the proposed changes are consistent with NEDO-32465-A.

Based on results of the NRC staff’s review of NEDC-33065P, Rev. 0, and the UFSAR event analyses, the NRC staff found the proposed changes are acceptable because they are based on the analyses provided in NEDC-33065P, Rev. 0, and the results of four UFSAR events are within applicable limits.

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. 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 36926). 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.

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Principal Contributors: T. Huang
C. Graham

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Author:	**José March-Leuba**
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Prepared by the
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831
Managed by
UT-Battelle, LLC
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Attachment

SUMMARY

This report documents our review of AmerGen's Technical Specification Change Request No. 308 for Oyster Creek[1] and the supporting documentation provided in NEDC-33065P, [2] a licensing topical report that describes a plant-specific analysis that documents the applicability of Long Term Solution Option II to Oyster Creek, a boiling water reactor (BWR) of type BWR-2.

Our review is based on data presented in the submitted documentation, a meeting at the Oyster Creek plant on August 29, 2002, and during a number of previous meetings with the Boiling Water Reactor Owners' Group (BWROG) and General Electric (GE). Based on our technical evaluation of these data, we find that Long Term Solution Option II is applicable to Oyster Creek because of its low power density, the unfiltered flow-biased thermal-power scram, and the quadrant symmetry of its average power range monitoring (APRM) system.

The implementation of Option II in Oyster Creek will require modifications to the technical specifications to reflect the more restrictive flow-biased scram setpoints that are required to avoid safety limit violations. The implementation will also require the administrative enforcement of an exclusion region.

Based on this review, we conclude that, following the implementation proposed in Technical Specification Change Request No. 308[1] and NEDC-33065P,[2] General Design Criteria (GDC) 12 will be satisfied by Oyster Creek even in the unlikely event that unstable power oscillations were to develop.

4. BACKGROUND

A long term solution to the stability problem is required to prevent the violation of specified acceptable fuel design limits (SAFDL) in the event of out-of-phase instabilities or core-wide instabilities with large local power peaking. Under these events, the reactor protection system (specifically the high APRM scram, or the flow-biased thermal-power scram) may not provide sufficient margin to prevent SAFDL violations under all postulated operating conditions in all reactors.

The reactor protection system in BWR-2's (e.g., in Oyster Creek) is based on an APRM system with quadrant symmetry. All other BWRs have an APRM system that averages neutron flux measurements from all over the core. Because of the quadrant symmetry, the APRM signal in BWR-2's does not "average out" the oscillations in out-of-phase instabilities, and automatic protection for this type of instabilities is possible. Long Term Solution Option II takes advantage of this special configuration and shows by analysis that the existing reactor protection system in BWR-2's provides protection against all expected instability modes.

Only two BWR-2's are in operation in the U. S.: Oyster Creek and Nine Mile Point-1. Oyster Creek submitted a topical report[3] in 1991 showing by analysis that their plant satisfied the requirements of Long Term Solution Option II. This report was reviewed and accepted[4] in 1992 following the evaluation and acceptance of other Long Term Solutions.[5-8] Nine Mile Point-1, the other Solution II plant submitted a request[9,10] to fully implement Option II in 1995. The request was reviewed and accepted[11] in 1996 following the evaluation and acceptance of other *Detect and Suppress* based Long Term Solutions[11-13] and the review of the flow control trip reference (FCTR) card, which was originally designed for Enhanced Solution IA.[14]

5. EVALUATION

Topical report NEDC-33065P[2] documents a detailed evaluation of the applicability of Solution II to Oyster Creek. In addition, Section 7 of this report documents the reload confirmation evaluations that must be performed every cycle to confirm that the generic results in the topical report are still applicable. This report follows the methodology already used and approved for Nine Mile Point.[10]

Because Solution II is in essence a *Detect and Suppress* option, the evaluation in NEDC-33065P[2] follows a procedure similar to the one already submitted[12] and reviewed[13] for other *Detect and Suppress* options, such as Solution III and I-D. As with other application of the *Detect and Suppress* methodology, the Option II implementation in Oyster Creek has followed three major steps:

1. Step 1 is to define the minimum critical power ratio (MCPR) that exists prior to the onset of the oscillation. The topical report assumes two initial MCPR (IMCPR) conditions: (a) operation at operating limits with 45% flow at the 100% rod line, and (b) operation at nominal conditions with a conservative MCPR followed by an all pump coast down to natural circulation.

   The IMCPR values used in topical report NEDC-33065P[2] are consistent with the approved Detect *and Suppress* methodology and are technically acceptable. The selection of a conservative MCPR to avoid cycle-specific dependence is technically acceptable because the conservative value used is significantly larger than the present MCPR safety limit.

   The choice of "point 1" (45% flow and 100% rod line) for delta-CPR evaluation is technically acceptable, because that flow is the highest intercept of the exclusion region and the Oyster Creek operating map. Thus, instabilities are not expected in Oyster Creek for flows higher than 45%. Note that this choice of flow threshold affects the actual flow-biased scram setpoint, which has a discontinuity at 45% flow to cover the maximum extended road line operation.

2. Step 2 is to determine the magnitude of the peak fuel bundle power oscillation. The Oyster Creek implementation follows the approved Nine Mile Point methodology,[9-11] which deviates slightly from other approved *Detect and Suppress* methodologies (i.e., Solution III and I-D) in the following items:

   a. The flow-biased trip setpoint has been adjusted in NEDC-33065P[2] in order to satisfy the MCPR safety limit criteria. The existing flow-bias trip setpoint is not adequate and must be lowered to satisfy these criteria. While this is a technically acceptable deviation, it poses some possible future restrictions on reload confirmations; and the possibility exists that the flow-biased setpoints may have to be modified in the future.

   b. The Oyster Creek implementation conservatively uses the most limiting oscillation contour to define the ratio between APRM and hot bundle oscillations. The implementation also uses a 1.10 penalty on the peak hot bundle oscillation to account in a deterministic manner for the overshoot caused by the oscillation growth rate and the scram time delay. The Use of conservatively limiting numbers avoids the need for the Monte Carlo-type calculations that are performed by other *Detect and Suppress* methodologies. This is a technically acceptable deviation.

c. The average power for the initial condition is assumed to be at the 100% rod line. The choice of a high average power reduces the oscillation amplitude required to reach the scram setpoint. If the oscillations were to occur at a lower operating power, the oscillation amplitude when the APRM scram setpoint is reached would be significantly larger and MCPR safety limits may be violated. This is the most questionable assumption in the topical report and must be weighted along with the other conservative assumptions and the proposed administrative restriction (an administratively-controlled exclusion region) to judge its technical acceptability. Because an administratively controlled exclusion region is enforced in Oyster Creek, the most likely instability scenario would be a flow-reduction event, which is likely to occur from the 100% rod line. The exclusion region would minimize the likelihood of startup instabilities. Thus, we conclude that the 100% rod line assumption for initial conditions is a technically acceptable assumption for these calculations.

3. The final step 3 is to determine the final MCPR by using a "generic" correlation that defines the loss in CPR for a given peak fuel bundle power oscillation. This generic correlation is known as the DIVOM curve, and its generic applicability has been question recently, resulting in Part 21 event, which is still on-going at this time.

   Oyster Creek has used the generic DIVOM curve for this application as the best available information at the time. Oyster Creek has made a verbal commitment to review the applicability of their current evaluation once the Part 21 DIVOM issue is resolved.

   We conclude that the use of the best-available information at this time (i.e., the generic DIVOM curve) is an acceptable technical approach for Oyster Creek because it will results in more conservative scram setpoints for Cycle 19 that the ones currently in place at Oyster Creek for cycle 18. We recommend that NRC follow up with Oyster Creek and review their evaluation once a final DIVOM approach is reached.

The application in topical report NEDC-33065P [2] of the three steps described above indicates that Oyster Creek satisfies the requirements of a Long Term Solution Option II if the flow biased scram setpoint is reduced to less than 54.6% of rated power at natural circulation conditions (22% rated flow) and 68.4% of rated power at 45% flow. In the Technical Specification Change Request No. 308 submittal,[1] Oyster Creek proposes to implement this change by replacing the flow reference trip control cards by the new cards developed originally for Option I-A,[14] because they allow significant flexibility to specify discontinuous scram set points as function of flow. This approach is similar to the one used in Nine Mile Point 1, and it is technically acceptable.

Section 7 of topical report NEDC-33065P [2] documents the reload confirmation procedures that will be required for future Oyster Creek core loadings. These procedures are consistent with those for other reviewed long-term solutions and are technically acceptable.

4. CONCLUSIONS AND TECHNICAL RECOMMENDATIONS

The main conclusions from this review are:

1. Long Term Solution Option II is applicable to Oyster Creek because of its flow biased, unfiltered scram system, and the quadrant symmetry of its average power range monitoring (APRM) system.

2. The proposed Oyster Creek Option II implementation satisfies the main criteria of a Long Term Solution by providing a viable detect and suppress function that will guarantee, in the case of instability, a very small likelihood of core damage without the need of operator intervention. This implementation is defined in Technical Specification Change Request No. 308[1] and NEDC-33065P,[2] and it includes a modification of the flow-biased scram hardware and a Technical Specifications modification to lower the setpoint to a value consistent with the calculation assumptions.

3. An administratively controlled exclusion region is required to minimize the probability of startup instabilities and to satisfy the 100%-rod-line initial-condition assumption in the analyses.

4. The reload confirmation procedures defined in Section 7 of and NEDC-33065P,[2] are consistent with other reviewed and approved long term solutions,[12] and they are technically acceptable for this Oyster Creek implementation.

Based on this review, we conclude that, following the implementation proposed in Technical Specification Change Request No. 308[1] and NEDC-33065P,[2] General Design Criteria (GDC) 12 will be satisfied by Oyster Creek even in the unlikely event that unstable power oscillations were to develop.

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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