August 25, 2006

Mr. Christopher M. Crane
President and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: REACTOR WATER CLEAN-UP HIGH ENERGY LINE BREAK DETECTION AND ISOLATION (TAC NO. MC6046)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 259 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek), in response to your application dated February 2, 2005, as supplemented by letters dated April 19, April 21, and June 13, 2006.

The amendment revises the Oyster Creek Technical Specifications to incorporate the isolation trip setting and the instrumentation surveillance requirements of the reactor water clean-up system high energy line break detection and isolation equipment.

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/

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. 259 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. 259
License No. DPR-16

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

   A. The application for amendment by AmerGen Energy Company, LLC, (the licensee) dated February 2, 2005, as supplemented by letters dated April 19, April 21, and June 13, 2006, 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. 259, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brooke D. Poole, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Technical Specifications

Date of Issuance: August 25, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 259

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Facility Operating License with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.1-8	3.1-8
3.1-15	3.1-15
4.1-8	4.1-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 259

TO FACILITY OPERATING LICENSE NO. DPR-16

AMERGEN ENERGY COMPANY, LCC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By application dated February 2, 2005, as supplemented by letters dated April 19, April 21, and June 13, 2006, AmerGen Energy Company, LLC, (AmerGen or the licensee) requested changes to the Facility Operating License for the Oyster Creek Nuclear Generating Station (Oyster Creek). The proposed amendment would revise the Oyster Creek Technical Specifications (TSs) to incorporate the isolation trip setting and the instrumentation surveillance requirements of the reactor water clean-up (RWCU) system high-energy line break (HELB) detection and isolation equipment.

At Oyster Creek, an automatic closure of the RWCU system isolation valves is initiated by a low-low reactor water level signal. General Electric Service Information Letter 604 indicated that if there is sufficient feedwater flow to maintain the reactor water level above the low-low water level setpoint, the required isolation signal setpoint might not be reached. This could occur if the reactor is operating at reduced power levels, or if the line break is smaller than a full guillotine break of the RWCU piping. In order to resolve this concern, in October 1998, the licensee added safety-grade pipe break detection/isolation equipment to monitor RWCU pump room temperature and initiate an RWCU system isolation when the RWCU pump room ambient temperature exceeded a preset limit below the process safety limit of 180 EF. This plant modification and its inclusion in the Oyster Creek Updated Final Safety Analysis Report (UFSAR) was done pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59. In 1997, General Public Utilities Nuclear (GPUN) (the licensee for Oyster Creek at that time) announced its intention to only operate the plant until the fall of 2000. Based on this decision to retire the plant early, GPUN determined that a license amendment to include the setpoint in the TSs was not required.

AmerGen subsequently purchased Oyster Creek and, on February 2, 2005, submitted the instant license amendment request (LAR) to include the instrument setpoint in the Oyster Creek TSs.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC or the Commission) staff used the following regulations and guidance documents in its evaluation of the LAR:

10 CFR Part 50.36, “Technical Specifications”, Section (c)(1)(ii)(A), which contains the requirements for limiting safety system settings for variables on which a safety limit has been placed; and paragraph (c)(2), which contains the requirements for limiting conditions for operation.

10 CFR Part 50, Appendix A, General Design Criterion 54, which specifies, in part, that piping systems penetrating primary containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating the systems.

Regulatory Guide (RG) 1.5, “Assumptions used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors [BWRs],” March 1971.

RG 1.105, “Instrumentation Setpoints for Safety-Related Systems,” Revision 2, February 1986.

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 3.6.1, October 1980.

NUREG-0800, “SRP for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 15.6.2, July 1981.

3.0 TECHNICAL EVALUATION

3.1 Description of Proposed Change

The RWCU HELB detection and isolation system was installed at Oyster Creek in September 1998, and consists of two trip systems, both of which are required to actuate to close the RWCU isolation valves. Each trip system contains two temperature switches, only one of which is required to trip its respective system. The proposed LAR contains the following TS changes:

TS Table 3.1.1, “Protective Instrumentation Requirements,” would be modified to add function P, “RWCU HELB Isolation, (1) RWCU Pump Room High Temperature.” The trip setting would be listed as #180 EF, with required operability in all Modes, with the exception of shutdown and refueling modes if the reactor coolant system temperature is less than 212 EF and the reactor vessel head is removed or vented or during reactor pressure vessel testing. The required minimum number of OPERABLE trip systems would be 2, with a minimum of 2 instrument channels per OPERABLE trip system. Caveat (oo) would be added, which details actions and completion times for single and instrument failures in one or both trip systems. Also, the required actions would be to close Isolation Valves V-16-1, V-16-2, V-16-14, and V-16-61.

> TS Table 4.1.1, “Minimum Check Calibration and Test Frequency for Protective Instrumentation,” would be modified to add Instrument Channel 33, “RWCU HELB High Temperature.” The test frequency would be specified as once every 3 months, with calibration each refueling outage.

The TS Bases would be modified to reflect these changes.

The licensee stated that the proposed changes to TS Tables 3.1.1 and 4.1.1 are consistent with the guidance provided in NUREG-1433, “Standard Technical Specifications - General Electric Plants BWR/4,” Revision 3, dated June 2004. The licensee also stated that, since the modification was installed in 1998, it has satisfactorily performed the proposed surveillance tests at frequencies specified in the proposed LAR.

3.2 RWCU System HELB Analysis

In its April 21, 2006, supplement, the licensee described the analysis utilized in postulating an RWCU system HELB. The analysis evaluates the thermal hydraulic response of the reactor building clean-up system and compartments to an RWCU line break. The results of the evaluation are used to provide input for establishing an appropriate setpoint for the temperature monitoring instrumentation that will isolate the cleanup system. The analysis evaluates the capability to detect, as well as the time to detect, a small break using an RWCU area temperature of 180 EF as the process safety limit.

The licensee calculated the time at which the break would be detected using the Generation of Thermal Hydraulic Information for Containments (GOTHIC) 5.0e computer code and generally conservative assumptions. GOTHIC is a general purpose thermal hydraulics computer program developed for modeling nuclear power plant containments and other confinement buildings. It models the mass, momentum, and energy transfer of multi-component, two-phase, fluids. The NRC staff has accepted previous analyses of this type using GOTHIC.

Some of the conservative assumptions included in the licensee’s analysis included modeling the line break as a smaller diameter (i.e., 0.75"), which would make detection more difficult. The break flow was directed towards a wall in the heat exchanger room, which minimizes the transport of steam to the pump room, where the detection would occur. Also, the licensee modeled the heat transfer to the walls, floor, and ceilings of the rooms. The licensee stated that the MAX option for heat transfer in GOTHIC was used. The MAX option uses the maximum condensation heat transfer and any convection or radiation heat transfer that is also specified. Modeling the maximum heat transfer away from the detection equipment is a conservative assumption and, therefore, acceptable to the NRC staff.

The licensee stated that the condensation was modeled with the maximum value of the Uchida and Gido/Koestel correlations. The NRC staff has previously found that the Gido/Koestel correlation is not acceptable for licensing calculations since a height dependence in the correlation was found to bias the predictions in a non-conservative direction[1]. This previous finding, however, was for a situation in which peak pressure was the parameter of interest and

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[1]Letter to Thomas Coutu, Site Vice President, Kewaunee Nuclear Power Plant, from Anthony C. McMurtray, NRC, September 29, 2003

condensation tends to lower the peak pressure. As it pertains to this case, where temperature is the parameter of interest, more condensation would push the result in a conservative direction. Therefore, the NRC staff finds this approach to be acceptable.

The licensee stated that as part of the evaluation of the 180 EF limit for RWCU system isolation, the impact on equipment qualification (EQ) was considered. The licensee's HELB analysis model was used to assess the time to reach the 180 EF analysis temperature value for a range of break sizes. The licensee's analysis results show that the smallest break (13 lbm/s coolant mass release) is detected in 60 seconds, while the largest break (850 lbm/s coolant mass release) is detected in 2 seconds. An integrated approach was used to assess which break would input the greatest amount of energy into the reactor building. The licensee's evaluation concluded that using the largest break and assuming that the detection time is 10 seconds (8 seconds longer than the calculated detection time) would provide bounding EQ profiles. The EQ profiles for pressure and temperature were developed in the licensee's calculation, "RB EQ Profiles Cleanup Line Modified Break Detection." The radiation levels are assessed in the licensee's calculation, "RWCU Area HELB TID." The licensee stated that the radiological analysis shows that the exposure associated with the postulated line break is below the capability of the qualified equipment within the RWCU areas. The NRC staff finds the EQ assumptions described by the licensee to be conservative and acceptable.

3.2 Radiological Consequences

In evaluating the radiological consequences of an RWCU system HELB using area temperature to detect the break, the licensee stated in its April 21, 2006, letter that the 1" diameter pipe break size is consistent with the current accident analysis documented in the Oyster Creek Facility Description and Safety Analysis Report and with Section 3.6.1, Appendix B, of the SRP. For this evaluation, the licensee stated that a conservative isolation time of 2 minutes was assumed.

In its evaluation of the offsite dose associated with the RWCU system HELB, the licensee applied the requirements of SRP Section 15.6.2 to determine the coolant mass release used in the dose calculation. The licensee stated that this mass release can be compared with the UFSAR mass release for the main steam line break (MSLB) analysis.

The licensee's dose analysis utilized the methodology identified in RG 1.5 and the reactor coolant activity was based upon the current Oyster Creek TS limit. The licensee stated that the calculation results indicated that the total coolant steam released is less than that assumed for the MSLB dose calculation in Chapter 15 of the Oyster Creek UFSAR. Therefore, the dose consequences of an RWCU system HELB are bounded by the MSLB accident analysis.

As discussed above, the NRC staff concurs that the dose consequences from a RWCU system HELB are bounded by the MSLB accident analysis and are, therefore, acceptable.

3.3 Calculation of Instrument Setpoint

The NRC staff has previously identified concerns relating to the use of the setpoints as limits that are used in the TSs to satisfy the requirements of 10 CFR 50.36. The NRC staff's concerns are documented in the Agencywide Documents Access and Management System (ADAMS) under accession numbers ML052500004, ML050870008, and ML051660447. During its review, the NRC staff requested additional information from the licensee to address these concerns.

By letter dated April 19, 2006, the licensee stated that, based on plant calculation C-1302-215-E610-060, the trip setting for the RWCU pump room high temperature instrumentation was selected to be #180 EF. This value was selected so that the system will detect a steam line break as small as 1" in diameter within one minute so that the room temperature will not exceed the environmental qualification value of 213 EF. It should be noted that the NRC staff evaluated this value as equivalent to the Allowable Value (AV) as defined in ISA-S67.04, Part I.

From the Trip Setting of #180 EF, the licensee in the originally-submitted plant calculation C-1302-215-E320-063, derived a total loop uncertainty (TLU) of 8 EF, an actual trip setpoint of 160 EF, an as-left band of ±4 EF, and an acceptable as-found band of ±12 EF. During its review, the NRC staff identified that the licensee used the as-left band of ±4 EF twice, first, in the TLU calculation and second, in the as-found band calculation. In response to a telephone conference call on May 31, 2006, and as documented by letter dated June 13, 2006, the licensee committed to reduce the as-found band to ±8 EF and to revise plant calculation C-1302-215-E610-060 accordingly. The resulting acceptable as-found value would then be 160 EF of ±8 EF. The NRC staff finds that this commitment will be adequately controlled by AmerGen’s Commitment Control Program and that the proposed actual trip setpoint is acceptable.

By letter dated April 19, 2006, the licensee also confirmed that, at Oyster Creek:

> Any TS or TS-supported instrument As-Found setpoint found outside of the As-Left limit must be reset to within the As-Left calibration tolerance before returning the instrument to service...An Issue Report (IR) is generated in accordance with [plant procedures] for instruments that do not satisfy the As-Found criteria established in the instrument calibration surveillance procedures. Operations [are] procedurally required to review IRs associated with plant equipment operability. The normal routing process of the IR will notify the associated System Manager of the instrument out of tolerance (OOT) [condition]. Exelon procedure ER-AA-2002 for System Health reporting requires that repeat instrument OOTs be identified as a “chronic problem,” and that action plans be developed to address the problem.

Based on this process, a TS instrument not performing within the acceptable as-found band, established by the setpoint calculation, would be identified, and an IR would be initiated. If the issue were a repeat, the chronic problem action plan would then identify corrective actions (e.g., instrument replacement, setpoint re-calculation, etc.). If the as-found value of the instrument were to exceed the trip setting in the TS, the instrument would be declared inoperable, and actions would be taken to restore operability.

In its letter dated April 19, 2006, the licensee confirmed that the Oyster Creek safety limits are those that are related to reactor pressure, reactor level, and neutron flux. The RWCU high area temperature trip setpoint function is to isolate the reactor pressure vessel due to certain HELBs as discussed previously. The NRC staff concludes that, because this function is not related to safety limits associated with the reactor core or the reactor coolant pressure boundary, this instrumentation does not require the TS notes mentioned in the September 7, 2005, letter from the NRC to the Nuclear Energy Institute (available under ADAMS accession number ML052500004).

The NRC staff reviewed the proposed Modes for which operability of the system would be required, minimum operable channels, tests, and calibration intervals. The NRC staff finds that the proposed requirements and intervals, which are consistent with NUREG-1433, Revision 3, are appropriate for the system.

Combined with the evaluations and the plant practices discussed above, the NRC staff concludes that the proposed addition to the TS conforms to the requirements of 10 CFR 50.36 and is, therefore, acceptable.

It should be noted that this review was limited to the single setpoint proposed and does not constitute approval of AmerGen's setpoint methodology as used for any other setpoints.

3.4 TS Bases

The NRC staff reviewed the provided Bases pages and finds that they describe and support the proposed amendment. Therefore, the NRC staff does not object to their inclusion in the Oyster Creek TSs.

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

This 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 effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 12744). 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: S. Mazumdar, R. Young

Date: August 25, 2006

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