April 26, 2007

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: APPLICATION OF ALTERNATE SOURCE TERM METHODOLOGY (TAC NO. MC6519)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 262 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek), in response to your application dated March 28, 2005, as supplemented by letters dated November 2, 2005, January 24, February 2, March 16, March 23, and March 28, 2007.

The amendment revises the Oyster Creek Licensing Basis in the area of radiological dose analyses for design-basis accidents using the alternative source terms depicted in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Additionally, the amendment revises the Oyster Creek Technical Specifications consistent with the amended design-basis.

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

Enclosure:
1. Amendment No. 262 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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Package Accession Number: **ML071080024**
Amendment Accession Number: **ML071080019**
TS(s) Accession Number: **ML07071170389**

OFFICE	LPLI-2/PM	LPLI-2/LA	DCI/CGSB/BC(A)	DRA/AADB/BC	DIRS/ITSB/BC	OGC(NLO)	LPLI-2/BC
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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 262
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 March 28, 2005, as supplemented by letters dated November 2, 2005, January 24, February 2, March 16, March 23, and March 28, 2007, 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 Facility Operating 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. 262, 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 its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: April 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 262

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace page 3 of Facility Operating License No. DPR-16 with the attached revised page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Technical Specifications with the attached page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
3.2-3	3.2-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 262

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 March 28, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML050940234)), as supplemented by letters dated November 2, 2005, January 24, February 2, March 16, March 23, and March 28, 2007 (ADAMS Accession Numbers ML053120376, ML070310101, ML070360330, ML070850816, ML070920067, and ML070880393, respectively), 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 Licensing Basis in the area of radiological dose analyses for design-basis accidents (DBAs) using the alternative source terms (ASTs) depicted in Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” Additionally, the amendment would revise the Oyster Creek Technical Specifications (TSs) consistent with the amended design-basis. The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on May 10, 2005 (70 FR 24646).

2.0 REGULATORY EVALUATION

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as explained in Regulatory Position 4.4 of Regulatory Guide 1.183 (RG 1.183), “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC-19), “Control Room.” Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in the latest revision of the following documents in performing this review.

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”

- Standard Review Plan (SRP) Section 15.0.1, “Radiological Consequence Analysis Using Alternative Source Terms”

- RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors”

- RG 1.194, “Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments at Nuclear Power Plants”

- SRP Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”

- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

The NRC staff also considered relevant information in the Oyster Creek updated final safety analysis report (UFSAR), TSs, and the licensee’s November 17, 2005, letter which provided additional information on the licensee’s June 12, 2003, response to Generic Letter 2003-01, “Control Room Habitability.” Additionally, the NRC staff performed independent confirmatory calculations to evaluate the licensee’s revised design basis dose analyses. The computer code discussed in NUREG/CR-6604, “RADTRAD: A Simplified Model for <u>RAD</u>ionuclide <u>T</u>ransport and <u>R</u>emoval <u>A</u>nd <u>D</u>ose Estimation,” and its supplements, was used by the staff in performing independent dose calculations.

3.0 <u>TECHNICAL EVALUATION</u>

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by AmerGen in support of its proposed license amendment. Information regarding these analyses was provided in Section 4.0 and Attachments 1 and 3 of the submittal and in supplementary letters dated January 24, February 2, March 23 and March 28, 2007. The NRC staff also conducted telephone discussions and public meetings with AmerGen and their contractor during the course of the review during which the licensee was able to clarify the information submitted in its application and supplements. The NRC staff reviewed the assumptions, inputs, and methods used by AmerGen to assess the impacts of the proposed license amendment. The NRC staff performed independent calculations to confirm the conservatism of the licensee’s analyses. However, the findings of this SE input are based on the descriptions of the licensee’s analyses and other supporting information docketed by AmerGen.

3.1 Loss-of-Coolant Accident Radiological Consequence Analysis

AmerGen performed the radiological consequence analysis of the most limiting DBA to show compliance with 10 CFR 50.67. The licensee determined that the loss-of-coolant accident (LOCA) offsite and control room dose results are bounding for the remaining DBAs in the

Oyster Creek design and licensing basis. Regulatory Position 1.3, "Scope of Required Analyses" in RG 1.183 states that for full implementation of an AST, a complete DBA LOCA analysis should be performed, as a minimum. Any implementation of an AST should be supported by evaluations of all significant radiological impacts of the proposed changes. AmerGen proposed only to incorporate the AST methodology in accordance with 10 CFR 50.67. Other than changes to emergency operating procedures and technical specifications related to the implementation of the AST and the proposed post-LOCA use of the standby liquid control system for suppression pool pH control, no other changes to the plant design or operation were proposed. Because none of the proposed changes impact the assumptions and inputs to the remainder of Oyster Creek's DBAs and the radiological consequences of the LOCA are bounding for the other DBAs, the NRC staff finds that only the analysis of the LOCA is required to support implementation of the AST at Oyster Creek.

Oyster Creek is a General Electric GE 2 boiling-water reactor (BWR) with a Mark I containment. The basic set of DBAs for BWRs for purposes of DBA radiological consequence analysis are the LOCA, main steam line break, control rod drop accident, and fuel-handling accident (FHA). The current Oyster Creek radiological consequence analyses based on the traditional source term in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," for the main steam line break, control rod drop accident and FHA remain as part of the design basis until the analyses are required to be revised. The licensee's current analyses for equipment qualification remain bounding and will continue to use the TID-14844 source term upon implementation of the AST, which is acceptable to the staff as discussed in RG 1.183.

AmerGen used the guidance in RG 1.183, including Appendix A, to develop the Oyster Creek DBA LOCA radiological consequence analysis. Tables A and B of Attachment 1 to the licensee's March 28, 2005, submittal present the licensee's determination of conformance with the positions in RG 1.183. In response to NRC staff question, some analysis assumptions and inputs were changed by licensee's supplemental letters dated January 24, February 2, March 23 and March 28, 2007, however, the conformance of the licensee's analysis to the applicable guidance was maintained. The NRC staff determined that the licensee's analysis assumptions and modeling are consistent with the guidance in RG 1.183, with acceptable exceptions discussed below.

The licensee evaluated the radiological consequences offsite and in the control room for the DBA LOCA. The following pathways for release to the environment were analyzed:

- Primary containment leakage
- Secondary containment bypass
- Main steam isolation valve (MSIV) leakage
- Leakage from engineered safety feature (ESF) systems

The radiological consequences of the LOCA are the sum of the total effective dose equivalent (TEDE) for the above pathways. The licensee calculated the dose to the maximally exposed individual at the exclusion area boundary (EAB) for the worst 2-hour period, at the low population zone (LPZ) outer boundary for the duration of the event, and in the control room for the duration of the event. AmerGen used dose conversion factors from Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," and FGR 12, "External

Exposure to Radionuclides in Air, Water, and Soil." Use of the dose conversion factors in FGR 11 and FGR 12 are acceptable to the NRC staff, as discussed in RG 1.183.

The licensee used the proprietary STARDOSE dose analysis code to perform the calculations to support implementation of an AST at Oyster Creek. In addition, the licensee performed analyses using RADTRAD Version 3.03 to check the STARDOSE calculations. Because the NRC staff uses RADTRAD version 3.03 to perform confirmatory analyses, the information provided by the licensee with regard to their use of RADTRAD was also utilized by the NRC staff in its independent analysis.

3.1.1 Accident Source Term

For the core inventory fraction release and timing, the licensee used the BWR assumptions in RG 1.183, Tables 1 and 4. The radionuclides considered and the chemical form of the radioiodine assumed in the licensee's analysis are consistent with RG 1.183. The licensee used an isotope generation and depletion code to determine the core isotopic inventory, based on a 2-year fuel cycle with a nominal 690 effective full power days per cycle. The DBA analysis is based on a power level of 1969 thermal megawatts (MWt), which is equivalent to the licensed rated power of 1930 MWt with a 2% power uncertainty. The staff finds that the licensee's assumptions for the source term and power level are consistent with the guidance in RG 1.183.

RG 1.183, Appendix A, states that the if the suppression pool pH is controlled at values of 7 or greater, the radioiodine species are assumed to be 95% cesium iodide, 4.85% elemental iodine and 0.15% organic iodide. AmerGen proposes to use the standby liquid control (SLC) system to inject sodium pentaborate to the vessel post-LOCA for the purpose of providing a buffering agent for the suppression pool water.

In the licensee's supplemental letter dated March 23, 2007, Appendix A provides a discussion of the projected iodine re-evolution in the wetwell, Appendix A. The licensee increased the calculated dose due to elemental iodine to account for this projected iodine re-evolution. The NRC staff considers this to be a conservative change because AmerGen determined that the suppression pool pH is shown to be controlled above 7 for the duration of the accident. Thus, any re-evolution of elemental iodine would not be significant. Review of the suppression pool pH is contained in Section 3.3 of this SE.

The NRC staff reviewed the SLC system with respect to its role in delivery of sodium pentaborate to the suppression pool for pH control. Control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged. As such, the new role being assigned to the SLC is a safety-related role. The licensee stated that the SLC is designated a safety-related system.

Additionally, the NRC staff reviewed the licensee's submittals and the responses to requests for additional information (RAIs) on the use of the SLC system for the safety-related function. From the licensee's supplements to the application, the NRC staff has concluded the following:

The SLC system is designated a safety-related system. As such, the SLC system as designed and installed is a high quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation. Specifically:

1. The system components required for reactivity control and new suppression pool pH control functions are seismically qualified.

2. The system is provided with emergency power with the capability to supply power from the emergency diesel generators.

3. The system is subject to American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection requirements as required by 10 CFR 50.55a, Codes and standards."

4. The system is within the scope of the 10 CFR 50.65 Maintenance Rule.

5. Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The licensee identified seven items which are susceptible to single failure. These are discussed below under single failure review.

6. Emergency Operating Procedures (EOP's) direct the activation of the SLC following a LOCA when reactor water level cannot be maintained above the top of active fuel. Manual initiation of SLC is also directed in the severe accident management guidelines (SAMGs), which are entered when adequate core cooling cannot be maintained. Procedures will be updated to specifically direct boron injection without dilution until the required amount of boron is injected for pH control.

7. Training will be provided on the new SLC injection function as part of operator re-qualification training and EOP and SAMG training.

The NRC staff considered components that could be subject to single failure. The licensee identified seven components as not being redundant. Two of the components, the containment isolation check valves, are installed in series. The failure of either check valve to open could prevent SLC injection. The check valves are Velan Valves Model 2A34B valves mounted in the injection line. In the periodic inspections and testing of these valves, Oyster Creek has not experienced any failures of these valves to open on demand. A review of the industry databases, EPIX and NPRDS, was performed and no failures of check valves of this type and manufacture failing to open were identified. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement, periodic testing and inspection, and historical performance of the component. The staff finds that the use of a single penetration of the containment with the identified check valves as described by the licensee to be acceptable.

Additionally, the NRC staff reviewed the remaining five items identified by the licensee as non-redundant. Although failure of the Liquid Poison Control Switch could delay injection, the switch is in an accessible location where it could be easily replaced or bypassed (jumpered) with no significant impact on the injection timing or objectives. The other four items: the control for the liquid poison pump suction line heat coil, the liquid poison pump suction line heater, the poison tank temperature indicating control switch, and the liquid poison tank immersion heater are in periodic service and subject to operator surveillance each shift. Failure prior to an accident would be observed and corrected. Failure at the time of an accident would not interfere with SLC injection since the contents of the tank would be at the right temperature for transport. The NRC staff finds that these five non-redundant components are acceptable based on their intended service.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the bottom of the reactor vessel. The transport of reactor vessel contents including the sodium pentaborate to the suppression pool is by flow through the break (assumed to be a large recirculation pipe break) to the drains that feed the suppression pool. Core spray (CS) systems and low-pressure coolant injection (LPCI) systems are used to maintain water level and assure core cooling after a LOCA accident.

Using the LPCI in this mode for suppression pool cooling also provides mixing. The NRC staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution. As discussed previously, the NRC staff review of the licensee's analysis of suppression pool pH is contained in Section 3.3 of this SE.

3.1.2 Primary Containment Leakage

The Mark I primary containment is composed of a drywell and a wetwell/torus which includes the suppression pool. The radioactivity released from the reactor coolant system to the containment is assumed to instantaneously and homogeneously mix throughout the drywell air space. Mixing between the drywell and wetwell air space is modeled at 9180 cubic feet per minute (cfm) from 1.129 hours to 1.296 hours, based on the expected steam flow from the drywell to the wetwell from analysis with the MAAP4 thermal-hydraulics code. The drywell and wetwell air space volumes are assumed well mixed by transfer of 10 drywell volumes per hour after 2.008 hours until the end of the event, due to expected opening of the vacuum breakers between the wetwell and drywell. The NRC staff finds this formulation to be reasonably conservative and the modeling of the primary containment volume acceptable.

The primary containment is assumed to leak at its design leakage rate of 1 percent of its contents by weight per day for the first 24 hours and then at 0.5 percent for the remainder of the 30-day accident duration. RG 1.183, Appendix A, Section 3.7 states that for BWRs, primary containment leakage may be reduced after the first 24 hours, if supported by plant configuration and analysis, to a value not less than 50% of the TS leak rate. The licensee stated that they performed an evaluation using the MAAP4 thermal-hydraulics code that justifies the reduction in assumed containment leakage after 24 hours based on the calculated reduction in drywell pressure.

MAAP4 is acceptable for calculating primary containment since it adequately represents the suppression pool, containment passive heat sinks and sprays (drywell and wetwell). Additionally, Since MAAP4 calculates that the drywell pressure is reduced by more than 50% after 24 hours, it is acceptable, based on the guidance of RG 1.183, to assume that the leak rate has been reduced by 50%.

The licensee's analysis took credit for removal of iodine by the drywell sprays and by natural deposition. Iodine removal is not modeled after 24 hours. The licensee calculated the aerosol and elemental iodine removal rates due to drywell spray and due to natural deposition. Natural deposition removal rates are applied during the times that the sprays are not operating. For time periods when the sprays are operating, the spray removal is much more effective and the spray removal rates are much larger than would be calculated for natural deposition.

The Oyster Creek drywell spray system is designated safety related, has been designed to work in post-LOCA containment conditions and its availability is governed by the TS. Drywell spray manual initiation is called for in the plant procedures, and the sprays do not operate continuously. The licensee's dose consequence analysis assumes that one pump in one loop of the drywell spray system is in operation with a flow rate of 3000 gallons per minute. The drywell sprays are manually initiated at 10 minutes, then are operated intermittently by manual operation until nearly 8 hours. The licensee calculated the aerosol elemental iodine spray and natural deposition removal coefficients using the proprietary STARNAUA code. The STARNAUA code, based on first principle aerosol physics, is capable of calculating the transport of particulate and its time-dependent airborne concentration in the presence of various removal mechanisms like sprays, natural deposition (gravitational sedimentation), containment leakage and steam condensation on the heat sinks (diffusiophoresis). The NRC staff did not review this code in detail, however, based on similarities to aerosol codes currently used in severe accident analysis and advanced reactor licensing, the NRC staff concludes that the employed aerosol models are scientifically sound. By letter dated January 24, 2007, the licensee described the methodology used in the STARNAUA code and compared it to the models found acceptable to the staff in RG 1.183. For spray removal, STARNAUA uses the same equation to calculate the removal coefficient as that in SRP 6.5.2, which is identified in RG 1.183 as being an acceptable model to the staff. STARNAUA differs from the SRP 6.5.2 model in the quantification of the collection efficiency to the average spray droplet diameter, which is an input to the equation. The licensee provided a figure comparing the spray removal efficiency calculated by STARNAUA and compared it to the accepted models in RG 1.183 and experimental data, as an example of the differences. The licensee stated that the best way to characterize the spray removal rates calculated by STARNAUA is that they are comparable to those derived from NRC methods, but not as conservative, although still within acceptance criteria. The STARNAUA removal rates are calculated based on similar aerosol characteristics as those used in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," which is an acceptable model to the staff, as noted in RG 1.183. The licensee also used STARNAUA to calculate the aerosol natural deposition rate due to sedimentation. Natural deposition of aerosol is a removal mechanism that is accepted by the staff, as noted in RG 1.183. The natural deposition removal rates calculated by the licensee are consistent with those that would be calculated by the model in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," which is a model acceptable to the staff. Based on the above discussion, the NRC staff finds the licensee's aerosol removal rates are acceptable.

The licensee assumed that the elemental iodine is removed at the same rate as the aerosols in the drywell. This approach is based on an assumption that "..elemental iodine, being reactive, will adhere to the aerosol." The NRC staff did not find either theoretical or empirical evidence in support of such a claim. Therefore, the NRC staff does not accept the physical rationale for this assumption. However, the NRC staff does recognize that elemental iodine is removed by sprays, as discussed in SRP 6.5.2. In response to NRC staff questions, and by letter dated

January 24, 2007, the licensee provided a discussion comparing the elemental iodine removal assumed by the licensee to that which would be calculated by using the SRP 6.5.2 model. The licensee stated that on average the non-limited elemental removal rate would exceed the aerosol removal rate. If the elemental removal rate is limited to 20 per hour as directed in SRP 6.5.2, the licensee stated that the integrated fraction of the core's elemental iodine airborne during spray operation would be essentially the same using the aerosol removal rates as compared to the 20 per hour removal rate, considering that the elemental iodine is only 4.85% of the total iodine release from the core. The NRC staff considers the demonstrated insensitivity to assumed removal rate changes to be a reasonable justification for use of the aerosol removal rate values for elemental removal. Therefore, the NRC staff finds the licensee's elemental iodine removal rates acceptable.

Leakage from the primary containment will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or bypass the secondary containment. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the primary containment is collected and processed by the SGTS. The SGTS exhaust is processed through charcoal filter media prior to release to the environment via the site's elevated stack. Oyster Creek does not credit dilution or holdup of leakage in the secondary containment. In accordance with the Oyster Creek current licensing basis as found previously acceptable to the staff, no drawdown time was modeled for the secondary containment. With the implementation of the AST methodology, the full melt source term is not assumed to enter the primary containment until approximately one half hour after the initiation of the LOCA, which ensures that the release to the secondary containment and then to the environment prior to drawdown would not be significant. Given that the differential pressure will ensure that leakage is processed by the SGTS, the NRC staff finds the licensee's assumptions with respect to primary containment to be acceptable.

3.1.3 Secondary Containment Bypass

Although the majority of the leakage from the primary containment is captured in the secondary containment and filtered by the SGTS, the licensee identified and modeled other paths that bypass secondary containment to release containment atmosphere to the environment.

In addition to leakage past the MSIVs, which is discussed below in Section 3.1.4, there are some secondary containment bypass pathways that terminate in the turbine building and some that terminate on the east wall of the reactor building. Those that terminate on the east wall of the reactor building use the control room atmospheric dispersion factor denoted as "yard" in the licensee's analysis. Deposition in the horizontal sections of the piping is credited in a similar manner as credited for the MSIV leakage piping, discussed below. Each pathway leaks at the design leak rate for the first 24 hours, then at half the design leak rate for the remainder of the event, based on reduction of containment pressure.

The 2-inch and 8-inch nitrogen (N_2) pathways are lines that terminate on the east wall of the reactor building. The total leak rate is 13 standard cubic feet per hour (scfh) for the 2-inch line and 8.5 scfh for the 8-inch line. An aerosol removal effective efficiency of 91.6% and elemental iodine removal efficiency of 50%, to account for 50% iodine re-evolution, are assumed. The traversing in-core probe (TIP) purge line is a 0.5-inch flowpath that ultimately connects to the

2-inch N_2 line discussed above. Additionally, the design leak rate is 0.05 scfh.

The instrument air line pathway, isolation condenser vent pathways and drywell spray test line pathway all terminate in the turbine building. During the period when sprays are operating, it is unlikely that the drywell spray test line would provide a containment bypass pathway, however the licensee conservatively assumed it would leak. The administrative leak rate for each pathway is 2 scfh. An aerosol removal effective efficiency of 96.5% and elemental iodine removal efficiency of 50%, to account for 50% iodine re-evolution, are assumed. The NRC staff finds these assumptions to be representative of the plant configuration and therefore acceptable.

3.1.4 MSIV Leakage

A major source of containment leakage that bypasses the secondary containment is MSIV leakage. Oyster Creek has two main steam lines, each with two MSIVs. An MSIV leakage limit of 11.9 scfh per valve at a test pressure greater than or equal to 20 pounds per square foot gauge (psig) is contained in the Oyster Creek TSs. Since the allowable TS leakage is assessed in units of scfh, and the steam lines are not at standard conditions of temperature and pressure, AmerGen adjusted the assigned flow rates for the assumed accident conditions.

In the March 28, 2005, submittal, the licensee proposed MSIV leakage rates and other bypass leakage rates that varied over time, based on changes in pressure in the primary containment over the course of the accident. In response to NRC staff questions regarding this assumption, the licensee changed the assumption to be less variable by assuming the MSIVs leak at the TS allowable rate of 11.9 scfh for the first 24 hours and at half that rate for the remainder of the accident, based on the calculated containment pressure reduction. The licensee's analysis of the LOCA MSIV leakage pathway used the same activity concentration that was modeled for the containment leakage pathway as the source of radioactive materials to be leaked through the MSIVs. This revised modeling is consistent with the guidance in RG 1.183.

The licensee's analysis credited the time delay for the leakage to travel from the outboard MSIV to the turbine/condenser before being released to the environment. The licensee assumed that one MSIV on one of the two steam lines fails to close at the time of the LOCA. The travel time for the leakage in the steam line with one MSIV failed open is 8.7 hours, and for the steam line with two closed MSIVs is 13 hours. These assumptions are the same as in the current licensing basis for Oyster Creek. No changes to the plant operation are proposed, nor do the characteristics of the proposed alternative source term impact the assumptions on travel time for the release through the main steam lines for the LOCA, therefore, the NRC staff finds the assumptions discussed above remain acceptable.

The licensee's analysis models aerosol and elemental iodine removal in the main steam line piping. AmerGen used the proprietary STARNAUA code to model the aerosol removal through sedimentation in the main steam line with both MSIVs closed. STARNAUA, discussed above with respect to iodine removal in containment, includes detailed aerosol physics models as used in severe accident analysis.

The licensee's aerosol removal model assumes the sedimentation volumes are well-mixed, only takes credit for horizontal lengths of seismically qualified piping, and does not take credit for piping beyond the outboard MSIV. No credit for sedimentation was taken for the steam line with one MSIV failed open. The licensee's calculation of aerosol removal rates in the steam line piping with two closed MSIVs uses detailed flow rates based on temperature and pressure through the MSIVs. Although detailed flow rates were used as input to the aerosol removal rate calculation, the leakage through the MSIVs in the dose calculation were constant at the TS value of 11.9 scfh for the first 24 hours, then decreased to a constant value of 50 percent of the TS value for the remainder of the accident. The more detailed flow rate modeling in the aerosol removal rate calculation gives smaller removal rate values than if a constant TS flow rate had been used, as was used in the revised release flow rate in the dose analysis. STARNAUA also models aerosol impaction on the MSIVs. This effect is included in the calculated effective aerosol removal rates. The staff compared the effective aerosol removal rate values used in the licensee's analysis to those that would be calculated as median values for aerosol sedimentation by the NRC report AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term." Several licensing precedents have found the use of median values calculated by appropriate use of AEB 98-03 to be acceptable. The effective aerosol removal rates calculated by the licensee are less than would be calculated as the median value for the credited volume of piping by the aerosol sedimentation removal model in AEB 98-03. Although the NRC staff has not generally found credit for removal by impaction on the MSIVs to be acceptable, the NRC staff does acknowledge that some amount of impaction may occur. Given that the effective aerosol removal rates used by the licensee are generally less than those the NRC staff would find acceptable for just the removal mechanism of sedimentation, the NRC staff finds the aerosol removal values used by the licensee to be acceptable.

The licensee's analysis assumed removal of 50% of the elemental iodine in both steam lines. The licensee stated this removal is consistent with assuming that the elemental iodine plates out on the aerosols with 50% of the plated out iodine re-evolving into the containment, and is also consistent with the proprietary STARNAUA model. As discussed above with regard to the elemental iodine removal in containment, the staff does not agree with the assumption that the elemental iodine will adhere to the aerosols. However, the NRC staff does find the elemental iodine removal rate in the steam lines to be reasonable because the assumption of 50% elemental iodine removal is similar to the amount of removal that accepted models, such as those referred to in RG 1.183, Appendix A would calculate.

3.1.5 ESF Leakage

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the reactor coolant system (RCS) and by spray removal processes. Post-LOCA, the suppression pool is a source of water for the ESF systems. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from ESF systems, the licensee assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. This source term assumption is conservative in that all of the radioiodine released from the fuel is assumed to be in both the primary containment atmosphere leakage and the ESF leakage concurrently. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time. Noble gases released from the fuel are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to become airborne on release from the ESF, they are not included in the ESF source term. These assumptions are consistent with the guidance in RG 1.183.

The analysis considers the equivalent of one gallon per minute (gpm) ESF leakage starting at the onset of the LOCA. This leakage rate includes a factor of 2 multiplier over the TS limit, in accordance with guidance in RG 1.183. AmerGen assumes 10% of the iodine in the ECCS leakage becomes airborne and is available for release. No credit was assumed for holdup and dilution in the secondary containment. The leakage is assumed to enter the environment via the SGTS as a filtered elevated release. The NRC staff finds these assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.1.6 Control Room Modeling

The Oyster Creek control room heating, ventilation, and air conditioning (HVAC) system is safety-related and consists of two independent trains. The control room HVAC system does not contain high efficiency particulate air filters or charcoal adsorbers. The system has four manual operating modes designated as normal, purge, partial recirculation and full recirculation. The normal mode automatically maintains a comfortable temperature and a slightly higher than atmospheric pressure in the control room. The purge mode brings in outside air at 14,000 cfm per train to clear smoke or fumes from the control room. The partial recirculation mode minimizes the intake of outside air to 2,000 cfm per train while maintaining a positive pressure of at least 0.125 inches water gauge in the control room. The full recirculation mode fully isolates the control room from outside air. All intake to and recirculation within the Oyster Creek control room is unfiltered, therefore no credit was taken for filtration in the licensee's radiological consequence analysis. The licensee's analysis used the maximum outside air intake flow rate of 14,000 cfm that is associated with one train operating in the purge mode.

The licensee's sensitivity studies show that the dose calculated in the control room is lower if the system is placed in the partial recirculation mode, which is the expected control room HVAC operation mode for an accident such as the DBA LOCA. The licensee stated that given the control room volume of 27,500 ft^3 and the volumetric exchange rate with the outside environment, the radionuclide concentration in the control room would be the same as that in the release plume at the air intake. The licensee did not model additional unfiltered inleakage through the control room envelope, and has not performed a tracer gas test to determine such leakage. By letter dated November 17, 2005, AmerGen responded to the request for information on the response for Oyster Creek to Generic Letter (GL) 2003-01, "Control Room Habitability." In that response, the licensee presented the results of a study to determine the impact on the control room dose of additional unfiltered outside air intake to the control room. The licensee determined that for the largest dose contributor pathway, the TEDE in the control room increased by less than 1/100th of a rem as the unfiltered outside air intake rate increased from 4,000 cfm to 80,000 cfm. The maximum forced air flow into the control room in the purge mode of operation is approximately 28,000 cfm with both fans running. Because the licensee used a bounding intake flow rate for the expected operation of the control room, and because their sensitivity study shows no impact on the dose results for additional unfiltered intake, the NRC staff finds the licensee's modeling of the control room HVAC system acceptable for the dose analysis.

The licensee used non-standard control room occupancy factors of 1 for the first 24 hours and 0.25 for the remainder of the accident. The basis for the licensee's occupancy factors is described in the licensee's letter dated March 28, 2007. The occupancy factors are based on the Oyster Creek staffing plan and are the same as in the current licensing basis. Therefore, the licensee's control room occupancy factors are acceptable to the NRC staff for use at the Oyster Creek Generating Station.

The licensee's control room dose results include the external dose due to gamma radiation shine from sources outside the control room in addition to the dose in the control room due to the outside air intake. The licensee conservatively estimated the external shine dose in the control room due to the outside plume by assuming it is 10% of the external dose due to the activity in the control room. The current design basis value for the shine dose of 0.6 rem due to other external sources continues to be bounding for the AST source term. The NRC staff finds the licensee's estimation of the shine doses to be acceptable.

3.1.7 Atmospheric Dispersion Estimates

AmerGen used meteorological data collected at the nearby Forked River Plant from 1995 through 1999 to generate new control room atmospheric dispersion factors (χ/Q values) for postulated releases from the plant yard as well as all EAB, and LPZ χ/Q values for the Oyster Creek site. The resulting χ/Q values represent a change from the χ/Q values used in the current Oyster Creek Updated Final Safety Analysis Report, Chapter 15, "Accident Analysis." The licensee used previously generated χ/Q values for postulated releases from the turbine building and plant stack to the control room air intake.

3.1.7.1 Meteorological Data

AmerGen generated new χ/Q values for Oyster Creek ground level and elevated release dose assessments using meteorological data collected from 1995-1999 on the Forked River Plant meteorological tower which is about 770 meters west northwest of the Oyster Creek stack. These data were provided for NRC staff review in the form of hourly meteorological data files in the format suitable for input into the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The NRC staff performed a review of the 1995-1999 hourly meteorological databases using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets.

Wind speed and wind direction data were measured on the meteorological tower at heights of 10.1 meters, 45.7 meters, and 115.8 meters above the ground. Temperature difference data, which were used to determine atmospheric stability class, were measured between two intervals, the 45.7-meter and 10.1-meter levels and the 115.8-meter and 10.1-meter levels. The combined data recovery of the wind speed, wind direction, and stability data was generally in the upper 90 percentiles at all three levels during each of the 5 years which meets the data recovery recommendation of RG 1.23, "Onsite Meteorological Programs."

With respect to reported atmospheric stability measurements between the 45.7-meter and 10.1-meter levels, there was a relatively high occurrence of extremely unstable and extremely stable conditions and low occurrence of neutral stability conditions in comparison to the occurrence of other stability conditions. This, however, is generally consistent with measurements made over a smaller (e.g., less than 50-meter) measurement interval near the ground and should result in conservative χ/Q value estimates with respect to the dose assessment described above. As expected for a more than 100-meter measurement interval, the atmospheric stability measurements between the 115.8-meter and 10.1-meter levels reported a high occurrence of neutral conditions and low occurrence of unstable conditions. There was a higher occurrence of extremely stable conditions in comparison to extremely unstable conditions. For both measurement intervals, the diurnal occurrence of stable and unstable conditions was as expected with stable and neutral atmospheric conditions generally reported to occur at night and unstable and neutral conditions during the day.

Wind direction frequency distributions for each measurement channel were reasonably similar from year to year among the three heights. Wind speed frequency distributions were also reasonably similar from year to year and among the measurement levels except in 1998 when the frequencies of light winds were moderately higher at the 10.1-meter and 45.7-meter levels.

A comparison of joint frequency distributions, those derived by the NRC staff from the ARCON96 hourly data with those developed by the licensee for input into the PAVAN atmospheric dispersion model (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”), showed good agreement for both the 10.1-meter and 45.7-meter level data and the 10.1-meter and 115.8-meter level data.

For the reasons cited above, the NRC staff finds that the 1995 through 1999 meteorological data provide an acceptable basis for making atmospheric dispersion estimates for use in the Oyster Creek dose assessments performed in support of this license amendment request.

3.1.7.2 Control Room Atmospheric Dispersion Factors

As part of the current license amendment request, AmerGen performed a radiological consequence analysis for only the most limiting Oyster Creek DBA, the LOCA. This dose assessment utilized three sets of control room χ/Q values. Postulated releases from the turbine building and yard were modeled as ground level releases. A third release was postulated from the Oyster Creek stack. With regard to other potential release points, AmerGen stated by letter dated March 16, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML070850816) that the control rod drop and FHA release points are via the standby gas treatment system and main stack which is the same as for the containment and engineered safety feature leakage during a design basis LOCA. The main steam line break release uses the same release location as for the turbine building in the LOCA analysis. Thus, the NRC staff finds that the licensee substantiated that no additional release points (e.g., for other DBAs) other than those assessed in the current license amendment request were postulated.

AmerGen generated new control room χ/Q values for postulated ground level releases from the Oyster Creek yard using the ARCON96 computer code and meteorological measurements between the 10.1-meter and 115.8-meter levels. RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of this license amendment request. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice in performing these types of calculations, with the following two exceptions. NRC guidance recommends use of meteorological measurements representative of the heights of release. Since the modeled heights of release from both the turbine building and yard were 10 meters, use of measurements between the 10.1-meter and 45.7-meter levels would generally be preferable in order to use the more representative atmospheric stability interval. Therefore, the NRC staff made comparison calculations using the data between the 10.1-meter and 45.7-meter levels and has concluded that AmerGen's use of the 10.1-meter and 115.8-meter levels is acceptable in this specific case. Second, with respect to the yard release calculation, AmerGen estimated two direction values, 75° and 255°, and selected 255° as the input, the direction from the release point to the intake. However, 75° is the correct input to the ARCON96 code which is written to use the direction from the intake back to the release point, i.e., the wind direction with respect to the intake. The NRC staff performed a calculation using 75° and has concluded that the χ/Q values calculated by AmerGen are acceptable for use in the dose assessment described above because they are more limiting.

For postulated releases from the turbine building, the licensee used χ/Q values calculated in support of Oyster Creek License Amendment No. 225 dated February 7, 2002 (ML020320579) and addressed in the associated SE. For the stack release dose assessment, the licensee used χ/Q values associated with Oyster Creek Amendment No. 105 dated July 15, 1986 (8607230486) and addressed in the associated SE.

On the basis of the review summarized above, the NRC staff has concluded that the χ/Q values for the Oyster Creek LOCA releases to the control room as presented in Table 3 are acceptable for use in the DBA control room dose assessment performed in support of this license amendment request.

3.1.7.3 Offsite Atmospheric Dispersion Factors

AmerGen generated ground level and elevated release χ/Q values for the EAB and LPZ using the methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the PAVAN computer code. AmerGen input EAB and LPZ distances of 414 meters and 3218 meters, respectively.

For postulated ground level releases, the licensee used the joint frequency distribution derived from the 1995-1999 10.1-meter wind data. Stability class was calculated using the temperature difference between the 45.7-meter and 10.1-meter levels on the meteorological tower. AmerGen also assumed a building minimum cross-sectional area of 1913 square meters and a reactor building height of 44.8 meters.

Elevated release χ/Q values were generated from the joint frequency distribution derived from the 115.8-meter wind data with stability class calculated using the temperature difference between the 115.8-meter and 10.1-meter levels. AmerGen used the elevated release mode option of the PAVAN computer code to model the release from the 112-meter free-standing stack at Oyster Creek, assuming fumigation during the 0-2 hour time period for the EAB and 0-4 hour time period for the LPZ as specified in RG 1.145 for coastal sites.

The NRC staff qualitatively reviewed the inputs and assumptions to the PAVAN computer calculations and found them generally consistent with NRC guidance. The staff also performed an evaluation of the resulting atmospheric dispersion estimates for the postulated ground level and elevated releases using PAVAN and found the AmerGen calculations acceptable. On the basis of this review, the staff has concluded that the resulting EAB and LPZ χ/Q values presented in Table 4 are acceptable for use in the accident dose assessments performed in support of this license amendment request. The licensee provided EAB X/Q estimates for time periods of longer than 2 hours duration, since such estimates were not used in this EAB dose assessment, they were not included in this review. Therefore, these values should not be considered approved as part of this license amendment.

3.1.7.4 Secondary Containment Drawdown - Meteorology

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states that the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. AmerGen estimated the relevant Oyster Creek 95% wind speed as about 21 miles per hour (mph) and stated that a wind speed of greater than 35 mph would be needed before secondary containment pressures would be positive when compared with the outside air on the downwind side of the reactor enclosure. The NRC staff confirmed the estimate of 21 mph from the 1995-1999 wind data measured at the 45.7-meter level. In addition, climatic wind data (Weather Data Viewer, Version 3, 2005 American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc., Atlanta, GA) provides a 95% wind speed estimate for Atlantic City, New Jersey, of about 19 mph, which is approximately 35 miles south of the Oyster Creek site.

3.1.8 Dose Analysis Results

The 30-day dose results for each pathway described above were added together to give the TEDE at the LPZ and in the control room. To determine the most limiting 2-hour period for the EAB dose, the pathway analyses were calculated holding the atmospheric dispersion factors constant for the 30-day accident duration to determine the greatest rate of dose increase. Using this method, the licensee determined that the 2-hour period with the highest dose at the EAB was the period between 1.5 and 3.5 hours. The licensee's calculated radiological consequences of the DBA LOCA meet the dose reference values in 10 CFR 50.67 for doses at the EAB, LPZ, and control room. Using the licensee's assumptions and inputs found acceptable to the NRC staff, as discussed above, the staff performed independent analyses which confirmed the licensee's dose results, including the worst 2-hour time period. The NRC staff used the DBA dose analysis code RADTRAD, version 3.03. Table 1 lists the licensee's calculated dose results for the DBA LOCA and Table 2 lists the licensee's inputs and assumptions found acceptable to the NRC staff.

3.2 Technical Specifications Revisions

The licensee proposed changes to TS 3.2.C.1, "Standby Liquid Control System" and its bases that extends existing operability requirements for the standby liquid control system to all plant operating conditions other than cold shutdown. The proposed change ensures that the standby liquid control system is available to maintain the suppression pool pH above 7.0, consistent with the radiological consequences analysis assumptions necessary to implement an AST using the guidance in RG 1.183. The licensee's radiological consequence analysis and conclusions rely on the ability to control the suppression pool pH for the LOCA, therefore the proposed change is acceptable with respect to the radiological consequences of DBAs. With regard to the proposed Bases changes, the NRC staff finds the licensee's bases control program to be the appropriate method for implementing a revision. Therefore, the proposed bases changes are not discussed in this SE.

3.3 Prevention of Iodine Re-Evolution

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," contains the NRC staff position that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95% cesium iodide (CsI), with no more that 5% of iodine (I) and hydriodic acid (HI). The radiation induced conversion of iodide in water into elemental iodine (I2) is strongly dependent on the pH. Without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be re-evolved into the containment atmosphere if the pH is less than 7 (i.e., acidic). Conversely, if the pH is maintained above 7 (i.e., basic), less than 1 percent of the dissolved iodine will be converted to elemental iodine.

AmerGen's methodology used to calculate the post-accident suppression pool water pH is based on a proprietary computer code and NUREG/CR-5950, "Iodine Evolution and pH Control." The factors included in the pH calculation are the time-dependent radiation level in the drywell and suppression pool; the drywell and suppression pool volume; the mass, type and dimensions of electrical cables; and the minimum mass of sodium pentaborate injected into the core from the SLC system.

AmerGen's calculation of the time-dependent pH of the suppression pool showed that the pH remains above 8.0 for the majority of the accident duration, decreasing to 7.9 at 30 days. The NRC staff performed an independent calculation of the suppression pool pH which confirmed the licensee's submitted calculation.

Given that AmerGen's calculation was supported by the NRC staff's independent calculation and shows that the suppression pool pH remains basic for the duration of the accident, re-evolution of iodine is not expected during a DBA. Therefore, the NRC staff finds the proposed change to be acceptable with respect to the prevention of iodine re-evolution.

3.4 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by AmerGen to assess the radiological impacts of implementing an alternative source term at Oyster Creek. A summary of the major inputs to the NRC staff's independent calculation is contained in Tables 1, 2, and 3. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by AmerGen to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed implementation of an AST is acceptable with regard to the radiological consequences of postulated DBAs.

This licensing action is considered to be a full implementation of the AST. With this approval, the previous accident source term in the Oyster Creek design basis is superceded by the AST proposed by AmerGen. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid and skin doses are superceded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the Oyster Creek design basis.

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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 24646). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: M. Hart, L. Brown, C. Lauron

Date: April 26, 2007

Table 1

Licensee Calculated Radiological Consequences of Design Basis Loss-of-Coolant Accident

Dose Receptor	Time Period (hr)	Calculated TEDE (rem)	10 CFR 50.67 TEDE Criterion (rem)
EAB	1.5 - 3.5	1.91	25
LPZ	0 - 720	0.59	25
Control Room	0 - 720	4.63	5

Table 2

Oyster Creek LOCA Radiological Consequences Analysis
Licensee Inputs and Assumptions

Reactor power (1930 x 1.02), MWt	1969
Core release fractions and timing	RG 1.183, Table 1
Dose conversion factors	FGR11/FGR12
Breathing rate - offsite, m^3/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
>24 hours	2.32E-4
Breathing rate - control room, m^3/s	3.47E-4
<u>Containment Leakage Pathway</u>	
Iodine species distribution	
Elemental	0.95
Organic	0.0485
Particulate	0.0015
Drywell volume, ft^3	180,000
Wetwell volume including pool, ft^3	210,000
Suppression pool volume, ft^3	
Maximum	92,000
Minimum	82,000
Reactor building volume, ft^3	1,800,000
Primary containment leak rate, % volume /day	
0- 24 hours	1
Greater than 24 hours	0.5
SGTS filter efficiency for all iodine species, %	90
Drywell-wetwell mixing rate, cfm	
0 - 1.129 hrs	0
1.129 - 1.296 hrs	9,180
1.296 - 2.008 hrs	0
2.008 - 720 hrs (assumed to be well mixed)	30,000
Drywell spray and natural deposition removal for elemental and particulate iodine	
0 - 24 hrs	Variable*
24 - 720 hrs	Not credited
*Rates are listed in Item 4.2 in Enclosure 2 to March 23, 2007 licensee letter	
Containment leakage pathway elevated release point	Stack

MSIV Leakage Pathway

Activity source same as for containment leakage case	
Number of main steam lines	2
MSIV leak rate, scfh*	
0 - 24 hr	11.9
24 - 720 hr	5.95
Volume between MSIVs, ft^3	32.4
Aerosol removal in closed steam line, hr^{-1}	
0 - 0.512 hrs	1.36
0.512 - 1.009 hrs	2.54
1.009 - 2.239 hrs	2.41
2.239 - 2.803 hrs	2.59
2.803 - 3.088 hrs	2.11
3.088 - 5.041 hrs	1.39
5.041 - 9.871 hrs	0.66
9.871 - 14.12 hrs	0.4
14.12 - 24 hrs	0.37
24 -720 hrs	0
Elemental iodine removal in closed steam line, %	50
Aerosol and elemental iodine removal in steam line with one MSIV failed open, %	50
Organic iodine removal in steam lines, %	0
MSIV leakage pathway ground level release point	Turbine building

ESF Leakage Pathway

Iodine species fraction	
Elemental	0.97
Organic	0.03
Particulate	0
Suppression pool liquid volume, ft^3	82,000
Total ESF leakage, gpm	1
Iodine flashing fraction	0.1
SGTS filtration start delay, min	0
SGTS iodine filtration efficiency, %	90
ESF leakage pathway elevated release point	Stack

Other Secondary Containment Bypass Pathways

Design basis leak rates at 35 psig, scfh	
Instrument air lines	2
Isolation condenser vents, total	2
Drywell spray test line	2
2" N_2 lines, total	13
8" N_2 lines, total	8.5
TIP purge line	0.05
Aerosol removal effective efficiency, %	
2"/8" N_2 lines	91.6
Remainder of bypass paths	96.5
Elemental iodine removal effective efficiency, %	
2"/8" N_2 lines	50
Remainder of bypass paths	50
Release locations	
2"/8" N_2 lines	East wall of reactor building (yard)
Remainder of bypass paths	Turbine building

Control Room Assumptions

Control room intake flow, cfm	14,000
Control room volume, ft^3	27,500
Control room occupancy factor	
0-24 hrs	1.0
1-30 days	0.25

No control room isolation

No control room intake or recirculation filtration

Atmospheric Dispersion Factors

Control Room	Table 3
Offsite	Table 4

Table 3

Oyster Creek
Control Room Atmospheric Dispersion Factors (X/Q in sec/m^3)

Time Interval	Stack	Yard	Turbine Building
0-2 hours	1.80×10^{-4}	2.88×10^{-3}	3.73×10^{-3}
2-8 hours	1.80×10^{-4}	2.49×10^{-3}	2.37×10^{-3}
0-8 hours	1.80×10^{-4}	2.59×10^{-3}	2.71×10^{-3}
8-24 hours	9.67×10^{-5}	1.15×10^{-3}	8.76×10^{-4}
24-96 hours	2.50×10^{-5}	8.44×10^{-4}	8.63×10^{-4}
96-720 hours	3.60×10^{-6}	7.18×10^{-4}	8.45×10^{-4}

Table 4

Oyster Creek
Offsite Atmospheric Dispersion Factors (X/Q in sec/m^3)

Time Interval	EAB - Ground	EAB - Stack	LPZ - Ground	LPZ - Stack
Fumigation 0-2 hours*	---	1.07×10^{-4}	---	---
Fumigation 0-4 hours*	---	---	---	1.68×10^{-5}
0-2 hours	1.41×10^{-3}	---	1.35×10^{-4}	1.83×10^{-6}
0-8 hours	---	---	6.23×10^{-5}	8.88×10^{-7}
8-24 hours	---	---	4.23×10^{-5}	6.19×10^{-7}
24-96 hours	---	---	1.82×10^{-5}	2.83×10^{-7}
96-720 hours	---	---	5.43×10^{-6}	9.16×10^{-8}

* Per Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"

Oyster Creek Nuclear Generating Station

Site Vice President - Oyster Creek
Nuclear Generating Station
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

Senior Vice President of
Operations
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Kathryn M. Sutton, Esquire
Morgan, Lewis, & Bockius LLP
1111 Pennsylvania Avenue, NW
Washington, DC 20004

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

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

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 and Regulatory Affairs
AmerGen Energy Company, LLC
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

Manager Licensing - Oyster Creek
Exelon Generation Company, LLC
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

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

Assistant General Counsel
AmerGen Energy Company, LLC
200 Exelon Way
Kennett Square, PA 19348

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

Correspondence Control Desk
AmerGen Energy Company, LLC
200 Exelon Way, KSA 1-–1
Kennett Square, PA 19348

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