September 26, 2007

Mr. Christopher M. Crane
President and Chief Executive Officer
Amergen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING RELOCATION OF TECHNICAL SPECIFICATION REQUIREMENTS FOR REFUELING AND SPENT FUEL POOL AREA RADIATION MONITORS (TAC NO. MD3783)

Dear Mr. Crane:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 260 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated December 12, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML063540168), as supplemented by letters dated May 31, 2007 (ADAMS Accession No. ML071520229) and July 11, 2007 (ADAMS Accession No. ML071920354).

The amendment revises the TMI-1 technical specifications (TS) by removing reference to the refueling area and spent fuel storage area radiation monitors. This will allow the relocation of the administrative operability requirements for these monitors from the TS to the Updated Final Safety Analysis Report and plant procedures. To support this change the current TMI-1 fuel handling accident in the fuel handling building has been reanalyzed without credit for actuation of the fuel handling building ventilation exhaust filtration system on a high radiation signal. The amendment also establishes compliance with criticality accident monitoring requirements in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68(b) rather than those contained in 10 CFR 70.24.

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 J. Bamford, Project Manager
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 260 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

September 26, 2007

Mr. Christopher M. Crane
President and Chief Executive Officer
Amergen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING RELOCATION OF TECHNICAL SPECIFICATION REQUIREMENTS FOR REFUELING AND SPENT FUEL POOL AREA RADIATION MONITORS (TAC NO. MD3783)

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cc w/encls: See next page

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Official Record Copy

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 260
License No. DPR-50

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

   A. The application for amendment by AmerGen Energy Company, LLC (the licensee), dated December 12, 2006, as supplemented by letters dated May 31, 2007, and July 11, 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 Title 10 of the *Code of Federal Regulations* (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-50 is hereby amended to read as follows:

   (2) Technical Specifications

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

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance. Implementation of the amendment shall include updating the UFSAR in accordance with 10CFR50.71(e). This update shall include the relocated operability requirements for the refueling area and spent fuel storage area radiation monitors as well as an update regarding criticality monitor requirements in accordance with 10 CFR 50.68(b)(8).

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

Harold K. Chernoff, Chief
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License No. DPR-50 and the
Technical Specifications

Date of Issuance: September 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 260

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace page 3 of Facility Operating License No. DPR-50 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 pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Remove	Insert
3-44	3-44
4-5a	4-5a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 260 TO FACILITY OPERATING LICENSE NO. DPR-50

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated December 12, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML063540168), as supplemented by letters dated May 31, 2007 (ADAMS Accession No. ML071520229) and July 11, 2007 (ADAMS Accession No. ML071920354), AmerGen Energy Company, LLC (AmerGen or the licensee), requested changes to the technical specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). 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 staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 3, 2007 (72 FR 36521).

The proposed changes would revise the TMI-1 TS to relocate the reactor building refueling area and spent fuel storage area radiation monitor operability requirements to the Updated Final Safety Analysis Report (UFSAR) and plant procedures based on the assertion that these radiation monitors do not meet the criteria for inclusion in the TS as presented in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(2)(ii). To support the proposed change, the licensee revised the TMI-1 fuel handling accident (FHA) in the fuel handling building (FHB), described in UFSAR Section 14.2.2.1.b.1, without taking credit for FHB ventilation exhaust filtration. The updated analysis of the FHA in the FHB incorporates the alternative source term (AST) methodology, pursuant to 10 CFR 50.67, as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." These proposed changes are consistent with standard TS as described in NUREG-1430, Revision 3.

2.0 REGULATORY EVALUATION

Section 50.36 of 10 CFR provides requirements for the inclusion of a limiting condition for operation in the TSs for operational constraints, which are used as an input in a design-basis accident (DBA) based on the following four criteria:

(A) Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

For this license amendment request (LAR) the applicable DBA for the evaluation of the cited 10 CFR 50.36 criteria is the FHA. In Amendment No. 257, dated October 13, 2005, (ADAMS Accession No. ML052660397), the licensee incorporated the AST methodology to evaluate the FHA in the reactor building (RB) without credit for isolation of the RB. To support the current LAR, the licensee performed a revised calculation that analyzes both the FHA in the RB and the FHA in the FHB using the AST methodology. The licensee made no changes to the analysis of the FHA in the RB as previously found acceptable to the staff in Amendment No. 257. The references to the FHA in the RB are included in this safety evaluation (SE) for information only.

The Nuclear Regulatory Commission (NRC or Commission) staff evaluated the radiological consequences of the FHA in the FHB using AST methodology as proposed by the licensee against the dose criteria specified in 10 CFR 50.67(b)(2). These criteria are: 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release; 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release; and 5 rem TEDE for access and occupancy of the control room (CR) for the duration of the postulated fission product release.

In addition to DBA considerations, 10 CFR 20.1501(a) requires licensees to “make or cause to be made surveys that ... (2) are reasonable under the circumstances to evaluate ... the magnitude and extent of radiation levels... and the potential radiological hazards.” The term “surveys” includes fixed radiation monitoring devices that continuously monitor radiation levels in selected areas of the plant such as areas where fuel is stored and/or transported, where the potential exists for changing radiological conditions. This ensures that appropriate action can be taken to protect personnel from excessive radiation.

Criticality monitor requirements for power reactors are specified in 10 CFR 70.24 and 10 CFR 50.68(b). Previous to this amendment, the TMI-1 licensing basis contained an exemption to 10 CFR 70.24, granted on July 3, 1997 (ADAMS Accession No. ML003765710). Based on the optional criteria specified in 10 CFR 50.68(a), compliance with 10 CFR 50.68(b) is not required if a licensee chose to operate under the requirements of 10 CFR 70.24 and any applicable exemptions. This is documented in the rulemaking comments for 10 CFR 50.68 published in the *Federal Register* on November 12, 1998 (63 FR 63127). As a result of this amendment, the licensing basis for the criticality monitors is being changed such that compliance with 10 CFR 50.68(b) is now required and the exemption to 10 CFR 70.24 is no longer applicable.

This SE addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. It also evaluates compliance with requirements for surveys and monitoring, as well as criticality accident monitoring. For DBA considerations, the regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and in Standard Review Plan (SRP) 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- 10 CFR Part 50.67, “Accident source term.”
- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Rev. 2, March 1978.
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Rev. 0, July 2000.
- NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability Systems,” Rev. 2, July 1981.
- NUREG-0800, “Standard Review Plan,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Rev. 0, July 2000.
- 10 CFR Part 20, Subpart F - “Surveys and Monitoring.”
- 10 CFR 50.68, “Criticality accident requirements.”

3.0 TECHNICAL EVALUATION

3.1 Background

Radiation monitors RM-G6 and RM-G7 are area gamma monitors currently located on the RB fuel handling bridges. Isolation of the RB is not credited in the fuel handling accident analysis for TMI-1, as described in existing UFSAR Section 14.2.2.1.b.2. RM-G6 and RM-G7 alarm as a result of detecting any excessive radiation in the vicinity of the refueling water surface and provide the CR operators sufficient information to initiate evacuation and closure of the RB in the event of a fuel handling accident.

Radiation monitor RM-G9 is an area gamma monitor currently located in the FHB on the east wall in the vicinity of the spent fuel pool (SFP) water surface. The current TS state that if RM-G6, RM-G7, or RM-G9 become inoperable, portable survey instrumentation having appropriate ranges and sensitivities to fully protect individuals involved in refueling operations, will be used until permanent instrumentation is returned to service. The above described functions and operability requirements are currently described in the TMI-1 UFSAR Section 11.4.2. With approval of this LAR, the licensee will revise the TMI-1 UFSAR and plant procedures to incorporate the relocated TS channel check, test, and calibration surveillance requirements for RM-G6, RM-G7, and RM-G9.

Radiation monitors RM-G6 and RM-G7 provide no interlock functions. Area gamma monitor RM-G9 interlocks are designed to trip and isolate the normal FHB ventilation system upon detection of a high radiation signal. The current TMI-1 FHA in the FHB analysis assumes the FHB is ventilated and discharges through 90 percent efficient charcoal filters to the unit vent. The RM-G9 interlock function is redundant to the interlock function provided by the FHB exhaust ventilation duct atmospheric radiation monitor RM-A4. The surveillance and operability requirements for RM-A4 are specified in the TMI-1 Offsite Dose Calculation Manual (ODCM) and are not described in the current TS. With the approval of TS Amendment No. 248, dated December 12, 2003, (ADAMS Accession No. ML033140383), the licensee committed to continue to test the normal FHB ventilation system fan stop and damper interlocks as part of the monthly and quarterly surveillance requirements associated with RM-G9 and RM-A4 (FHB exhaust radiation monitor) to provide assurance that the system will isolate. The licensee has not proposed to make any changes to this commitment.

In support of this LAR, the licensee performed an analysis of the FHA in the FHB without credit for FHB ventilation exhaust filtration and without the assumption that the RM-G9 interlock functions to isolate the normal FHB ventilation system. The licensee analyzed the FHA in the FHB in accordance with 10 CFR 50.67, "Accident source term," using the methodology described in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." In Amendment No. 235, dated September 19, 2001, (ADAMS Accession No. ML012480262), the NRC approved a full scope implementation of the AST for TMI-1, by approving the AST loss-of-coolant accident analysis. In Amendment No. 257, dated October 13, 2005, (ADAMS Accession No. ML052660397), the licensee incorporated the AST methodology to evaluate the FHA in the RB.

3.1.2 Fuel Handling Accident in the Fuel Handling Building

To support the current LAR, the licensee provided the staff with a revised calculation that analyzes the FHA in the FHB using the AST methodology. The FHA in the FHB, as described in Section 14.2.2.1.b.1 of the TMI-1 UFSAR, consists of mechanical damage during transfer operations to a single fuel assembly in the FHB. The UFSAR description of the FHA in the FHB specifies that as a result of the drop, the entire outer row of the fuel rods in one assembly, 56 of 208, experience mechanical damage to the cladding, releasing the gap activity. In addition, a minimum water level of 23 feet is maintained above the spent fuel stored in the spent fuel storage racks. The current licensing basis FHA in the FHB credits FHB filtration. To support the current LAR, the licensee performed two separate analyses to evaluate the dose consequence of the FHA in the FHB with and without credit for FHB filtration. In accordance with RG 1.183, the licensee assumed that the release to the environment from the FHA occurs over a 2-hour period.

3.1.2.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. The licensee’s evaluation of the inventory of fission products that are available for release from the gap of the damaged fuel is based on the maximum power level of 2619 megawatts thermal (MWt), which is 1.02 times the current licensed rated thermal power of 2568 MWt.

The licensee applied the maximum core radial peaking factor of 1.70, consistent with the existing licensing basis. The licensee indicated that there have previously been approximately 8 to 16 fuel assemblies evaluated to exceed the RG 1.183 footnote 11 value of 6.3 kilowatt per foot (kW/ft) peak rod average power for burnups exceeding 54 gigawatt days per metric ton of uranium (GWD/MTU). The licensee stated that the current core design does not contain any assemblies that exceed the 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU. However, to account for the possibility of the use of such assemblies in the future, the licensee doubled the RG 1.183 Table 3 gap release fractions for noble gas and halogens. The licensee asserts and the staff agrees that this is a conservative approach. This conservative assumption for the evaluation of the gap fractions for the FHA in the FHB is consistent with the approach used to evaluate the FHA in the RB, as approved in license Amendment No. 257.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumed that: the chemical form of radioiodine released from the fuel to the SFP consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide; the CsI released from the fuel is assumed to completely dissociate in the pool water; and because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine in the pool water. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

As corrected by item 8 of Regulatory Issue Summary (RIS) 2006-04 (ADAMS Accession No. ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read, “If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 70% elemental and 30% organic species.”

In accordance with the guidance in RG 1.183, the licensee applied an overall iodine decontamination factor (DF) of 200 for a water cover depth of 23 feet. Consistent with RG 1.183, the licensee credited an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity. In accordance with RG 1.183, the licensee did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

3.1.2.2 Transport

As prescribed in RG 1.183, the TMI-1 FHA is analyzed based on the assumption that 100 percent of the fission products released from the SFP are released to the environment over a 2-hour period. For the FHA in the FHB, the licensee evaluated two cases. The first case assumes that all of the activity released from the pool water would be released to the environment with no credit for the FHB filtration system. The second case assumes the activity released from the pool water would pass through the FHB filtration system prior to being released to the environment. The licensee analyzed the second case assuming that the activity released from the pool water would be processed through the FHB filtration system with elemental and organic iodine filter efficiencies credited at 90 percent. In accordance with the guidance in RG 1.183 the licensee assumed that all particulate and aerosol activity released as a result of the FHA remains in the SFP water.

3.1.2.3 Control Room Habitability for the Fuel Handling Accident

To eliminate the effects of radiation monitor detection delays, the licensee conservatively assumed that the TMI-1 control room would be isolated manually by a CR operator 30 minutes after an FHA. During this 30-minute period, the licensee assumed a total unfiltered intake of 61,000 cubic feet per minute (cfm) in the CR habitability analysis. The licensee asserts, and the staff agrees, that this value conservatively produces a CR environment that is in equilibrium with the outside air for the initial 30-minute time period.

After the initial 30 minutes, the licensee credited the safety related CR emergency filtration system for dose mitigation. In the emergency mode, the CR ventilation consists of 8000 cfm of filtered makeup airflow and 28,000 cfm of filtered recirculation flow. For the FHA CR habitability analysis, the licensee assumed an unfiltered inleakage of 1000 cfm after CR isolation. A charcoal filter efficiency of 90 percent is assured by the applicable TS. However, for added conservatism, the licensee assumed a 75 percent charcoal efficiency for the CR emergency filtration system.

The licensee evaluated the radiological consequences resulting from a postulated FHA in the FHB and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Tables 2 through 5 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.2 Radiation Monitoring

The relocation of the operability requirements for RM-G6, RM-G7 and RM-G9 from the TS to licensee controlled documents is administrative in nature and does not, by itself, involve a change to the ability of the radiation detectors to monitor and alarm at an appropriate level to provide personnel protection for radiological conditions in the refueling and SFP areas. The licensee has stated that this TS change or any future changes in monitor location will not adversely impact the ability of the monitors to evaluate the magnitude and extent of radiological hazards in the area and adequately warn personnel of off-standard conditions. Since the requested licensing action will not impact the ability of the monitors to perform their survey function as required by 10 CFR Part 20, the staff finds the change acceptable.

3.3 Criticality Monitor

The licensee, in its supplement dated May 31, 2007, has established their intent to comply with the requirements of 10 CFR 50.68(b) in lieu of operation under the previous exemption to 10 CFR 70.24. This includes compliance with the monitoring function of 10 CFR 50.68(b)(6), which states, "Radiation monitors are provided in storage and associated handling areas where fuel is present to detect excessive radiation levels and to initiate appropriate safety actions." Relocating the operability requirements from the TS to licensee controlled documents does not impact the ability of the monitors to perform this function. The licensee has stated that the monitors will continue to be located in the fuel handling areas as specified in General Design Criteria (GDC) 63 and as stated in the previous 10 CFR 70.24 exemption. Since the monitors will remain in the fuel handling areas when fuel is present and those monitors meet the requirements of GDC 63 and 10 CFR 50.68(b), this change is acceptable to the staff. The licensee will update its UFSAR in accordance with 10 CFR 50.68(b)(8) to reflect this licensing basis change, as required by 10 CFR 50.71(e).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official from the Bureau of Radiation Protection 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 (72 FR 36521). 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: John Parillo
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Date: September 26, 2007