October 20, 2003

Mr. John L. Skolds, Chairman
and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1), RE: CYCLE 15 CORE RELOAD ANALYSIS (TAC NO. MB7270)

Dear Mr. Skolds:

The Commission has issued the enclosed Amendment No. 247 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated January 16, 2003, as supplemented June 11, 2003.

The amendment revises the technical specifications to incorporate changes associated with the TMI-1 Cycle 15 core reload design analysis. The TMI-1 Cycle 15 core reload design implements the Framatome ANP statistical core design methodology . This amendment permits the licensee to determine the minimum departure from nucleate boiling ratio using an NRC-approved methodology based on statistical analysis of operational and design uncertainties.

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/

Donna M. Skay, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 247 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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ACCESSION NO.: ML032970039 (Package)
ACCESSION NO.: ML032600017 (Letter) *SE provided. No substantive changes.
ACCESSION NO.: ML032960496 (TS) **See previous concurrence

OFFICE	PDI-1\PM	PDI-2\LA	SRXB/SC	IROB	OGC	PDI-1\SC
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DATE	10/17/03	10/07/03	9/3/2003	9/30/03	10/10/03	10/17/03

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. 247
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 January 16, 2003, as supplemented June 11, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: October 20, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 247

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove	Insert
2-1	2-1
2-2	2-2
2-3	2-3
2-4a	2-4a
2-4c	2-4c
2-6	2-6
2-7	2-7
2-8	2-8
2-10	2-10
2-11	2-11
3-29	3-29
4-4	4-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 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 letter dated January 16, 2003 (Reference 1), as supplemented by letter dated June 11, 2003 (Reference 2), AmerGen Energy Company (AmerGen or licensee) requested approval of a license amendment for Three Mile Island, Unit 1 (TMI-1). The supplement dated June 11, 2003, 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 March 18, 2003 (68 FR 12948).

The proposed changes would revise Technical Specification (TS) Section 2.1, "Safety Limits - Reactor Core," Section 2.2, "Safety Limits, Reactor System Pressure," Section 2.3, "Limiting Safety System Settings, Protection Instrumentation," Table 2.3-1, "Reactor Protection System Trip Setting Limits," Table 3.5-1, "Instruments Operating Conditions," and Table 4.4-1, "Instrument Surveillance Requirements." Specifically, the proposed changes would permit the licensee to determine the minimum departure from nucleate boiling ratio using an Nuclear Regulatory Commission (NRC)-approved methodology based on statistical analysis of operational and design uncertainties.

2.0 REGULATORY EVALUATION

2.1 Background

The TMI-1 TSs are currently based on a deterministic approach which considers all of the worst-case uncertainties in calculating the cycle-specific minimum departure from nucleate boiling ratio (DNBR). This analytical method results in a conservative minimum DNBR for each cycle. However, it is highly unlikely that these uncertainties would occur simultaneously. Therefore, the NRC approved the Framatome Topical Report BAW-10187P-A, "Statistical Core Design for B&W [Babcock and Wilcox Co.]-Designed 177 FA Plants" (Reference 3). This methodology provides additional design and operational flexibility by increasing available margin to the core safety limits.

The implementation of BAW-10187P-A requires the licensee to make a number of changes to the TMI-1 TSs. Specifically, the licensee requests the TSs be amended to change the core protection safety limits, reactor protection system (RPS) trip limits, RPS instrument channels, and instrumentation surveillance requirements. A detailed description of the proposed TS changes is contained in Section 2.3 of this safety evaluation (SE).

The Framatome ANP Statistical Core Design (SCD) methodology (BAW-10187P-A) has been implemented at all other B&W 177 FA plants for which Framatome ANP performs reload licensing. Therefore, AmerGen performed its TMI-1 Cycle 15 reload analysis using the SCD methodology and has proposed TS changes consistent with this analysis.

2.2 Regulatory Requirements and Review Documents

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50 Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," (Reference 4), provides a list of the minimum design requirements for nuclear power plants. According to GDC 10, "Reactor design," the core and its protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs). The staff reviewed the amendment request to ensure that the licensee complied with GDC 10.

To identify other regulatory requirements for its review, the NRC staff used Standard Review Plan (SRP), Sections 4.2, "Fuel System Design," 4.3, "Nuclear Design," and 4.4, "Thermal and Hydraulic Design," (References 5, 6, and 7). Each SRP section provided a detailed list of potentially affected GDC requirements. The staff reviewed the acceptance criteria of each SRP section to ensure the licensee's amendment request was reviewed against the appropriate GDCs.

From the staff's review of the aforementioned SRP sections, it identified that the following GDCs, in addition to GDC 10, are appropriate review criteria for the licensee's amendment request:

1. GDC 13, "Instrumentation and control," requires the provision of instrumentation and controls to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

2. GDC 20, "Protection system functions," requires automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and to assure automatic operation of systems and components important to safety under accident conditions.

3. GDC 29, "Protection against anticipated operational occurrences," requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

These four GDCs (10, 13, 20, and 29) served as the basis for the staff's review of the proposed TMI-1 TS changes.

2.3 Description of Proposed TS Changes

AmerGen proposed the following changes to the TMI-1 TSs:

1. TS Figure 2.1-1, "Core Protection Safety Limit:" The figure is revised to incorporate the Cycle 15 limits based on the SCD methodology. Additionally, Bases Figure 2.1-3, "Core Protection Safety Bases," is similarly revised.

2. TS Table 2.3-1, "Reactor Protection System Trip Setting Limits:" This table is revised to add the variable low reactor coolant system pressure trip (VLPT) setpoint. Additionally, TS Bases Figure 2.3-1, "Protection System Maximum Allowable Setpoints," was revised to incorporate the VLPT setpoint.

3. TS Table 3.5-1, "Instruments Operating Conditions," and Table 4.4-1, "Instrument Surveillance Requirements:" These tables were revised to incorporate operational and surveillance requirements for the pressure-temperature instrument channels associated with the VLPT reactor protection system (RPS) function.

The staff used the information provided in the revised bases pages as supporting information for its review of the TS changes and did not review the changes to the TS bases for the purpose of their approval.

3.0 <u>TECHNICAL EVALUATION</u>

In determining the acceptability of AmerGen's amendment request, the NRC staff reviewed three aspects of the licensee's analyses: 1) the implementation of the SCD methodology; 2) the results of the Cycle 15 Departure from Nucleate Boiling (DNB) limiting transients; and 3) the incorporation of the VLPT setpoint for Cycle 15 operations. For each part of the review, the staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins were developed in accordance with NRC regulations and could be maintained during Cycle 15 operation at TMI-1.

3.1 TMI-1 Implementation of SCD Methodology

The TMI-1 Cycle 15 core reload design implements the Framatome ANP SCD methodology with a higher design radial-local peaking factor, and allowances for a potential future Appendix K power uprate application. The Framatome ANP SCD methodology (BAW-10187P-A, Reference 3) is a thermal-hydraulic analysis technique that provides additional DNBR margin by statistically combining core and fuel element uncertainties, while retaining the criterion that the core is designed to avoid DNB. Prior to the implementation of the SCD methodology, TMI-1 treated uncertainties by assuming the worst-case for each uncertainty simultaneously. This results in an exceptionally conservative prediction of the cycle-specific DNBR limit. The NRC approved the SCD methodology in March 1994.

The licensee stated that its use of the SCD methodology for Cycle 15 is acceptable for the following reasons:

1. The values and ranges for the state parameters and uncertainty parameters described in Tables 3-4 and 3-6 of BAW-10187P-A that were used in developing the Statistical Design Limit (SDL) are all applicable to TMI-1.

2. The fuel designs utilized at TMI-1 are the Mark-B designs, for which the Babcock & Wilcox correlation (BWC) critical heat flux (CHF) correlation has been approved. The Mark-B fuel design and the BWC CHF correlation were assumed in the development of the SDL of 1.313.

3. The LYNXT core thermal-hydraulic code is used for core DNB calculations

4. Core state variables that were not explicitly included in the SCD are input to thermal-hydraulic computer codes at their most adverse allowable level.

5. Cycle-specific evaluations will be performed for each reload to determine if the bounding assembly-wise power distribution assumed in the core-wide SDL calculation bounds the expected operating power distributions.

At the staff's request, the licensee provided additional information (Reference 2) describing how it met the limitations delineated in the staff's safety evaluation approving the SCD methodology. The NRC staff has reviewed the licensee's response and finds that the licensee's use of the SCD methodology is within the limitations listed in the staff's approval, and is, therefore, acceptable for use in the Cycle 15 reload design.

The licensee also assumed the following TMI-1-specific values in the SCD analysis.

1. The licensee assumed a nominal reactor coolant system (RCS) flow rate of 104.5 percent of design flow in all core DNB analyses supporting the proposed amendment. This design flow is based on TMI-1's approved ability to operate with up to 20 percent average once-through steam generator tube plugging and minimum RCS flow rate of 102 percent of design flow including a 2.5-percent measurement uncertainty.

2. The licensee assumed a design radial-local peaking factor of 1.80 in all core DNB analyses supporting the proposed amendment.

3. The licensee assumed a rated power level of 2612 MWt (i.e. 1.7 percent power uprate from 2568 MWt) in all core DNB analyses supporting the proposed amendment.

The staff reviewed these plant-specific values against those approved by the NRC for use in the SCD methodology. The staff found that each of these values is within the limits previously approved in BAW-10179P-A (Reference 8), and is, therefore, acceptable.

Because the licensee's inputs to the SCD methodology are within the approved ranges, the licensee is justified in using the SDL provided by the topical report. The licensee added additional retained margin to the SDL to offset effects not treated in the SDL development, such as transition core effects, deviations in uncertainty values from those incorporated in the SDL, or other cycle-specific emergent issues. The licensee stated that for Cycle 15 the retained margin, in the form of a Thermal Design Limit (TDL), is used to offset transition core effects from co-resident Mark-B10 and Mark-B12 fuel assemblies. For Cycle 14, the licensee operated with a commitment to maintain a higher RCS flow rate to offset the effects of the transition core penalty. The licensee will remove this commitment prior to Cycle 15 since the TDL now specifically contains added margin to account for the penalty. The staff reviewed the licensee's methodology for determining the TDL and determined that it contains sufficient margin to account for transition core effects as well as other cycle-specific requirements. The staff finds that the licensee has determined an appropriately conservative TDL and, therefore, removal of the commitment is acceptable.

Finally, the licensee revised core protection safety limits and bases based on the NRC-approved reload methodology (BAW-10179P-A, Reference 8) and the NRC-approved core thermal-hydraulic code LYNXT (BAW-10156P-A, Reference 9). The staff finds the licensee's use of these methodologies acceptable for determining the core protection safety limits.

3.2 DNB Limiting Transients

To demonstrate that the TDL determined from the SCD methodology provides adequate protection against violating SAFDLs, the licensee analyzed the most limiting DNB transients for TMI-1. These events are the following:

1. Single reactor coolant pump coastdown (4-to-3 pumps)
2. Four reactor coolant pump coastdown (4-to-0 pumps)
3. Single reactor coolant pump locked rotor (4-to-3 pumps)

The licensee analyzed these events using the LYNXT code and used nominal state parameters as input, consistent with the SCD methodology. The licensee used the higher nominal radial peaking factor of 1.7341 (1.80 when 3.8 percent uncertainty is included) and higher rated thermal power level of 2612 MWt. Finally, the licensee input the higher design RCS flow rate of 104.5 percent to account for up to 20 percent tube plugging.

In its analysis of each of the DNB limiting transients, the licensee compared the TDL for Cycle 15 with the minimum DNBR calculated using the LYNXT code. In each case, the licensee determined that considerable margin existed between the minimum DNBR and the TDL. For example, in its analysis of the locked rotor event, an event for which a limited amount of fuel failure is acceptable, the licensee determined that the minimum predicted DNBR would remain above the TDL for Cycle 15. Therefore, the licensee has determined that no fuel failure would occur.

The proposed amendment changes could also impact fuel handling accidents (FHAs) since the design radial-local peaking factor is being increased from 1.714 to 1.80. The higher peaking factor could potentially increase the source term (i.e. isotopic inventory) of an average power fuel assembly. The isotopic inventory is highly dependent on the steady state power level of the reactor. Transient events have a short-lived and limited effect on the isotopic inventory. The licensee determined that the steady-state radial-local peaking factor is 1.64 for Cycle 15. Since this value is below the 1.70 limit used in the current FHA analyses, the licensee concluded that the FHA for Cycle 15 is bounded by the previous analysis. The staff reviewed the licensee's assumptions and conclusions and agrees that the Cycle 15 FHA analyses are bounded by the current analyses.

In addition to analyzing the DNB limiting transients, the licensee's reload methodology (Reference 8) requires it to verify that all other transients and accidents are bounded by the current safety analyses. Therefore, the staff has determined that the licensee's implementation of the SCD methodology and its Cycle 15 operating and design conditions will not result in transients or accidents which would violate the DNB safety limits.

3.3 Variable Low Pressure Trip

In performing its reload analysis, the licensee determined that the pressure-temperature safety limits for the upcoming cycle could not be solely protected by the low pressure and high reactor coolant temperature trips. The licensee determined that it needed to credit the variable low pressure trip (VLPT) to prevent exceeding the safety limit during normal and transient conditions. The licensee stated that the proposed VLPT setpoint, in conjunction with the existing reactor coolant low pressure and high temperature trips, will ensure that power operation will be restricted to temperature/pressure conditions that meet the DNB safety criterion.

To determine the appropriate VLPT setpoint, the licensee applied the NRC-approved reload methodology, BAW-10179P-A (Reference 8). The methodology for determining the VLPT setpoint is described in Section 7.6 of this topical report. Because the licensee used the NRC-approved methodology and has adhered to the limitations described therein, the staff has determined that the licensee's calculation of the VLPT setpoint for Cycle 15 will provide adequate protection against violation of the DNB safety limit during normal and transient conditions.

In order to reintroduce the VLPT into the reactor protection system, the licensee proposed to amend TS Tables 3.5-1 and 4.1-1. Table 3.5-1, "Instruments Operating Conditions," describes the minimum operating channels, minimum degree of redundancy, and operator actions if either of the other conditions are not met for each of the reactor protection system trips. Table 4.1-1 provides the surveillance requirements for each of the reactor protection system trips. The amended tables contain the minimum requirements for the VLPT setpoint regarding both the operating conditions and surveillance requirements. The staff compared the licensee's proposed changes in Tables 3.5-1 and 4.1-1 to the B&W Standard Technical Specifications, NUREG-1430 (Reference 10). The staff found that the licensee's proposed changes meet or exceed the requirements listed in NUREG-1430. Since the changes are consistent with the requirements previously reviewed and approved by the staff for Cycle 7 operations (the last cycle for which the VLPT was credited at TMI-1) and are consistent with NUREG-1430, the staff finds the amended tables acceptable.

In its review of the reintroduction of the VLPT setpoint into the TMI-1 TSs, the staff evaluated the licensee's methodology for determining the appropriate setpoint and its controls for ensuring the trip would provide adequate protection against violating the DNB safety limit. The staff found that the licensee used an approved methodology to determine the appropriate setpoint. Additionally, the staff determined that the proposed operating conditions and surveillance requirements were consistent with both those previously approved and used at TMI-1 prior to Cycle 7 and NUREG-1430. Therefore, the staff concludes there is reasonable assurance that the proposed VLPT setpoint, minimum operating conditions, and surveillance requirements would provide adequate safety margin.

3.4 Conclusion

The staff reviewed the effects of the proposed changes using the appropriate requirements of 10 CFR, Part 50, Appendix A, and the associated guidance of SRP Sections 4.2, 4.3, and 4.4. The staff determined that the licensee's proposed TS changes would meet the requirements of GDCs 10, 13, 20, and 29. The staff found that the licensee's amendment request provided reasonable assurance that under both normal and transient conditions the licensee would be able to safely operate the plant and comply with the NRC's regulations. Therefore, the staff finds AmerGen's amendment request acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (68 FR 12948). 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.

7.0 REFERENCES

1. Letter from M. P. Gallagher, AmerGen, to NRC, "Technical Specification Change Request No. 316 - Cycle 15 Core Reload Design," dated January 16, 2003, ADAMS Accession No. ML030280516.

2. Letter from R. J. DeGregorio, AmerGen, to NRC, "Response to Request for Additional Information - Technical Specification Change Request No. 316, Cycle 15 Core Reload Design (TAC NO. MB7270)," dated June 11, 2003, ADAMS Accession No. ML031690396.

3. BAW-10187P-A, “Statistical Core Design for B&W-Designed 177 FA Plants,” B&W Fuel Company, Lynchburg, Virginia, March 1994.

4. Title 10, *Code of Federal Regulations*, Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Revised as of January 1, 2003.

5. NUREG-0800, Standard Review Plan, Section 4.2, “Fuel System Design,” Draft Revision 3, April 1996.

6. NUREG-0800, Standard Review Plan, Section 4.3, “Nuclear Design,” Draft Revision 3, April 1996.

7. NUREG-0800, Standard Review Plan, Section 4.4, “Thermal and Hydraulic Design,” Draft Revision 2, April 1996.

8. BAW-10179P-A, Revision 4, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses,” B&W Fuel Company, Lynchburg, Virginia, August 1993.

9. BAW-10156P-A, Revision 1, “LYNXT Thermal-Hydraulics Code,” Framatome Cogema Fuels, Lynchburg, Virginia, February 1996.

10. NUREG-1430, Revision 2, “Standard Technical Specifications, Babcock and Wilcox Plants,” Volume 1, NRC, Washington, DC, June 2001.

Principal Contributor: R. Taylor

Date: October 20, 2003