September 6, 2001

Mr. Mark E. Warner
Vice President - TMI Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - AMENDMENT RE: REVISION OF PRESSURE-TEMPERATURE LIMITS OF TECHNICAL SPECIFICATION 3.1.2 AND LOW-TEMPERATURE OVERPRESSURE PROTECTION REQUIREMENTS OF TECHNICAL SPECIFICATIONS 3.1.12 AND 4.5.2 (TAC NO. MB1584)

Dear Mr. Warner

The Commission has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated March 29, 2001, as supplemented by information contained in letters dated June 27, 2001, and July 24, 2001.

The amendment revises the reactor coolant system heatup, cooldown, and inservice leak hydrostatic test limitations for the reactor coolant system to a maximum of 29 effective full power years in accordance with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix G. These pressure-temperature (P-T) limits are contained in TMI-1 Technical Specification (TS) 3.1.2. In addition, the amendment revises the low-temperature overpressure protection (LTOP) requirements in TSs 3.1.12 and 4.5.2 to reflect the revised P-T limits. These changes will allow operation of two reactor coolant pumps in a single loop during LTOP conditions.

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/

Timothy G. Colburn, 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. 234 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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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. 234
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 March 29, 2001, as supplemented by letters dated June 27, 2001, and July 24, 2001, 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. 234 , 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/

Patrick Milano, Acting 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: September 6, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 234

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
ii	ii
vii	vii
3-3	3-3
3-4	3-4
3-5	3-5
3-5a	3-5a
3-5b	3-5b
3-18d	3-18d
3-18e	3-18e
3-18f	3-18f
4-41	4-41

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 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 March 29, 2001, as supplemented by letters dated June 27, 2001, and July 24, 2001, AmerGen Energy Company, LLC (the licensee), submitted a request for approval of changes to the Three Mile Island Nuclear Station, Unit 1 (TMI-1), Technical Specifications (TSs). The July 24, 2001, letter provided additional clarifying information which did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice (66 FR 38758). An additional letter dated August 20, 2001, transmitted the revised camera-ready TS pages.

The requested changes revise the reactor coolant system heatup, cooldown, and inservice leak hydrostatic test limitations for the reactor coolant system to a maximum of 29 effective full power years (EFPY) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix G. These pressure-temperature (P-T) limits are contained in TMI-1 TS 3.1.2. In addition, the amendment revises the low temperature overpressure protection (LTOP) requirements in TSs 3.1.12 and 4.5.2 to reflect the revised P-T limits. These changes will allow operation of two reactor coolant pumps (RCPs) in a single loop during LTOP conditions. A related exemption was issued on August 30, 2001.

2.0 BACKGROUND

2.1 Pressure Vessel Fluence, Pressure-Temperature (P-T) Limits and Pressurized Thermal Shock

The Nuclear Regulatory Commission (NRC) has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11, “NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations,” dated July 12, 1988; GL 92-01, Revision 1 (Rev. 1), “Reactor Vessel Structural Integrity, 10 CFR 50.54(f),” dated February 28, 1992; GL 92-01, Rev. 1, Supplement 1, dated May 19, 1995; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2) “Radiation Embrittlement of Reactor Vessel Materials,” dated May 1988; NRC RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” published in March of 2001; and Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2 contained in NRC NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” published in July of 1981.

GL 88-11 advised licensees that the NRC staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the NRC staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the NRC staff as the basis for the staff's review of P-T limit curves, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires, in part, that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves.

Pursuant to 10 CFR 50.60 and 10 CFR 50.61, licensees are required to demonstrate that the materials of their RPVs satisfy specific fracture toughness standards (i.e., that the effects of progressive embrittlement by neutron irradiation do not compromise the integrity of the RPV). The effects of neutron radiation must consider the radiation conditions, (i.e., the fluence) as specified in the regulations. TMI-1 participates in an integrated surveillance program (Reference 1) that satisfies the requirements of 10 CFR Part 50, Appendix H (note that Appendix H is referenced in 10 CFR 50.60). In addition, TMI-1 participated in the Babcock & Wilcox Owners Group (B&WOG) cavity dosimetry program, which has been approved by the NRC (Reference 2). TMI-1 began cavity dosimetry surveillance in Cycle 7. All of the surveillance data (in-vessel and cavity dosimetry) were used to verify the calculational results and to establish the uncertainty. The actual values were derived using the B&W methodology documented in BAW-2241PA, which has been reviewed and approved by the NRC (Reference 3).

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

[illegible]

69 In this [illegible] F (σ = 20.6 F and σ = 28 F), respectively. The [illegible] calculation is

P-T Limits

Table 1: Comparison of ART

Fluence (10^{19} n/cm^2) 1/4T	0.56	0.61	0.61	0.61	0.61	0.56	0.43	0.59	0.38	0.38	0.38	[2]
Chem. Factor (°F)	51.0	58	58	82	73	199.3	152.4	220.6	220.6	220.6	143.9	
ΔRT_{NDT} (°F)	42.7	49.9	49.9	70.5	62.8	166.8	116.4	188	162.1	162.1	N/A	
Init. RT_{NDT} (°F)	3	1	1	1	1	-26	-5	-7	-7	-7	-5	
Margin (°F)	70.7	63.6	63.6	63.6	63.6	56	68.5	69.5	69.5	69.5	N/A	
ART (°F)	116.4	114.5	114.5	135.1	127.4	196.8	179.9	250.5	224.6	224.6	N/A	

[1] Value determined using the master curve methodology

[2] Note: The "N/A" designation for LS Longitudinal weld (OD 63%) means that there is no inside surface fluence for the weld, so the RT_{PTS} value for this weld is not applicable and is not evaluated as part of this calculation.

Table 2: Comparison of

Fluence (10^{19} n/cm^2) 1/4T	0.20	0.22	0.22	0.22	0.16	0.21	0.14	[4] 0.14	N/A	0.14
Chem. Factor (°F)	51.0	58.0	58.0	82.0	73.0	199.3	152.4	220.6	220.6	220.6	143.9
ΔRT_{NDT} (°F)	29.2	34.4	34.4	48.6	43.3	114.2	77.9	128.3	107.4	N/A	70.1
Init. RT_{NDT} (°F)	3	1	1	1	1	-26 [3]	-5	-7	-7	-7	-5
Margin (°F)	68.5	63.6	63.6	63.6	63.6	56	68.5	69.5	69.5	N/A	68.5
ART (°F)	100.7	99.0	99.0	113.2	207.9	144.2	141.4	190.8	169.9	N/A	133.6

[3] Value determined using the master curve methodology

[4] The "N/A" designation for LS Longitudinal weld (ID 37%) means that there is no outside surface fluence for the weld, so the RT_{PTS} value for this weld is not applicable and is not evaluated as part of this calculation.

[illegible] for normal operation and by the NRC staff has [illegible] for hydrostatic pressure [illegible] curve

Pressurized Thermal Shock

[illegible]

Enable Temperature

[redacted]

332 F" with "indicated RCS temperature is

329 F" is acceptable because

Reactor Coolant Pump Operation

Actuation Setpoint

these [illegible] has been [illegible]

Date: September 8, 2021 Primer 8, 2021 Contributors: L. Zoi

Worsham, [illegible] Program,"

C. A. Dialee

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