January 27, 2004

Mr. Peter E. Katz
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: PRESSURE-TEMPERATURE LIMIT CURVES (TAC NO. MC0331)

Dear Mr. Katz:

The Commission has issued the enclosed Amendment No. 240 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated August 15, 2003, as supplemented by letter on September 15, 2003.

The amendment revised the reactor coolant system pressure-temperature limit curves in Section 3.4.11, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," of the TSs. The revised curves are effective up to 22 effective full-power years.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. 240 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: **ML040220584**

OFFICE	PDI-1\PM	PDI-1\LA	EMCB/SC	SRXB/SC	OGC	PDI-1/SC	
NAME	PTam	SLittle	SCoffin*	JUhle*	RWeisman	RLaufer	
DATE	1/8/04	1/8/04	12/23/03*	9/8/03*	1/22/04	1/22/04	

*SE transmitted by memo on these dates.

OFFICIAL RECORD COPY

DATED: <u>January 27, 2004</u>

AMENDMENT NO. 240 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT UNIT NO. 2

PUBLIC
PDI R/F
RLaufer
SLittle
PTam
SRichards
OGC
GHill (2)
WBeckner
CSydnor
LLois
ACRS
BPlatchek, RI

cc: Plant Service list

NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

LONG ISLAND LIGHTING COMPANY

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 240
License No. NPF-69

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

   A. The application for amendment by Nine Mile Point Nuclear Station, LLC (the licensee) dated August 15, 2003, as supplemented by letter on September 15, 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. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 240 are hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 27, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 240

TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following pages of 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 Pages	Insert Pages
3.4.11-6	3.4.11-6
3.4.11-7	3.4.11-7
3.4.11-8	3.4.11-8
3.4.11-9	3.4.11-9
3.4.11-10	3.4.11-10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 240 TO FACILITY OPERATING LICENSE NO. NPF-69

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated August 15, 2003, Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP2), submitted changes related to the reactor pressure vessel (RPV) pressure-temperature (P-T) limits in the NMP2 Technical Specifications (TSs). The licensee proposed to revise P-T limits which would be effective through 22 effective full-power years (EFPYs) of facility operation. The proposed changes to the P-T limits were based, in part, on the use of American Society of Mechanical Engineers (ASME) Code Case N-640.

To support the proposed amendment, the licensee submitted additional information by a letter dated September 15, 2003. This letter provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the *Federal Register*, and did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates P-T limit curves based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; General Design Criteria (GDC) 14, 30, and 31 of Appendix A of 10 CFR Part 50; Generic Letter (GL) 88-11; Regulatory Guide (RG) 1.99, Revision 2; RG 1.190; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code). GL 88-11 advised licensees that the NRC staff would use RG 1.99, Revision 2 to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) in RPV material resulting from neutron irradiation. GL 92-01, Revision 1, requested that licensees submit RPV data for review. GL 92-01, Revision 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 as the basis for the review of P-T limit curves. 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 the methodology of Appendix G to Section XI of the ASME Code is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Appendix G to Section XI of the ASME Code specifies a safety factor of 2.0 for stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for stress intensities resulting from hydrostatic testing. Appendix G to Section XI of the ASME Code also specifies a safety factor of 1.0 for stress intensities resulting from thermal loads for normal and transient operating conditions as well as for hydrostatic testing. 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 (i.e., of axial orientation). This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 6 times its depth. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit 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 methodology found in Appendix G to Section XI of the ASME Code provides that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) at the 1/4T and 3/4T locations. The ART is defined as the sum 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 term. Guidance on the determination of ΔRT_{NDT} and the margin term is given in RG 1.99, Revision 2. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001, provides guidance regarding acceptable methods for the benchmarking of vessel fluence methodologies based on the requirements of GDC 31, and in part on GDCs 14 and 30.

3.0 TECHNICAL EVALUATION

3.1 The Proposed Pressure-Temperature Limit Curves

The licensee requested, pursuant to 10 CFR 50.90, an amendment to the TSs revising the RPV P-T limit curves utilizing the modification of ASME Code Case N-640. ASME Code Case N-640 permits application of the lower bound static initiation fracture toughness (K_{IC}) curve as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness (K_{IA}) curve, which is invoked by Appendix G to Section XI of the ASME Code. All other aspects of the ASME Code, Section XI, Appendix G process for determining P-T limit curves remain unchanged in the licensee’s evaluation.

The licensing basis for the P-T limit curves at NMP2 is stated in the TSs as Figures 3.4.11-1 through 3.4.11-5. These figures provide P-T limits for normal reactor operation, including heatup and cooldown for core critical and core not critical conditions, with the respective heating and cooling rates ≤ 100°F/Hour, as well as for leak/hydrostatic test conditions. The proposed amendment would revise Figures 3.4.11-1 through 3.4.11-5, providing the revised P-T limits for the above heatup and cooldown conditions, and leak/hydrostatic test conditions. The revised P-T limits would be effective through 22 EFPYs of NMP2 operation.

The licensee submitted ART calculations and revised P-T limit curves valid for up to 22 EFPYs of facility operation. The licensee reported that, to date, only one surveillance capsule has been pulled for the materials represented in the NMP2 surveillance program. Consequently, only one credible surveillance capsule data point exists for the NMP2 RPV surveillance program materials. Therefore, the ART calculations were based on the use of a CF value that was derived from Table 2 of RG 1.99, Revision 2, as set forth in Regulatory Position 1.1 of this RG. Using this method, the licensee derived a CF value through linear interpolation between data points in Table 2. Based on these methods, the licensee found that the limiting beltline materials differed for the 1/4T and 3/4T locations. The licensee determined that the most limiting beltline materials at the 1/4T location were plates C3147-1 and C3147-2. The licensee determined that plate C3065-2 was the most limiting beltline material at the 3/4T location. For the 1/4T location, the licensee determined that plates C3147-1 and C3147-2 had identical chemistry factors, and would therefore exhibit equally limiting fracture toughness properties. The ART values for the limiting materials at the 1/4T and 3/4T locations at 22 EFPY were determined as follows:

	1/4T Location	3/4T Location
Limiting Material ID	Plates C3147-1 and -2	Plate C3065-2
Fluence	3.77×10^{17} n/cm^2	1.79×10^{17} n/cm^2
Chemistry Factor	74.5	37
ΔRT_{NDT}	18.6°F	5.9°F
Initial RT_{NDT}	0°F	10°F
Margin	34.5°F	29.6°F
ART	53.1°F	45.5°F

The equations for the thermal stress intensity factor (K_{IT}) and the stress intensity factor due to pressure loads (K_{IP}) were developed in accordance with the provisions of Appendix G to Section XI of the ASME Code. Accordingly, the P-T curves were generated by correlating the stress intensity factors due to thermal and pressure loads (K_{IT} and K_{IP}) with the reference fracture toughness curve, which was derived using the ART values cited above. In calculating the revised P-T limit curves, the licensee invoked the ASME Code Case N-640 modification to the ASME Code, Section XI, Appendix G procedures by using the lower bound K_{IC} fracture toughness curve in lieu of the lower bound K_{IA} fracture toughness curve.

The NRC staff used this information to evaluate the acceptability of the proposed NMP2 P-T limit curves.

3.2 Evaluation of Proposed P-T Limits Curves

ASME Code Case N-640 permits application of the lower bound static initiation fracture toughness (K_{IC}) curve as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness (K_{IA}) curve, which is invoked by Appendix G to Section XI of the ASME Code. Use of the lower bound K_{IC} curve in the development of P-T operating limits is technically correct, because the lower bound K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process for an RPV. Specifically, the NRC staff has determined that P-T curves based on the K_{IC} fracture toughness curve, as referenced by ASME Code Case N-640, will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operation. In addition, implementation of the proposed P-T curves, as defined by the technical basis supported by ASME Code Case N-640, maintains adequate margins of safety in protecting the RPV from brittle failure. This code case has been approved for use without conditions in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." In addition, the code case has been incorporated into the 1998 version of the Code, which has been endorsed by the NRC staff in 10 CFR 50.55a.

The NRC staff performed independent calculations of the ART values for all of the NMP2 beltline materials using the methodology in RG 1.99, Revision 2. The NRC staff verified that, to date, only one surveillance capsule has been pulled for the NMP2 RPV, resulting in only one credible surveillance capsule data point for the materials represented in the NMP2 surveillance program. According to the guidance in RG 1.99, Revision 2, a minimum of two credible surveillance capsule data sets should be available for the reactor in question if the ART values are determined using surveillance data. Therefore, the licensee correctly utilized the methodology of regulatory position 1.1 in RG 1.99, Revision 2, Table 2 as the means for obtaining values for the limiting plates' chemistry factor. Based on independent calculations, the NRC staff verified that the licensee's limiting beltline materials for the NMP2 RPV differed for the 1/4T and 3/4T locations, due to neutron fluence attenuation through the beltline thickness. The NRC staff verified that plates C3147-1 and C3147-2 were the most limiting beltline materials at the 1/4T location and plate C3065-2 was the most limiting beltline material at the 3/4T location. For the 1/4T location, the NRC staff verified that plates C3147-1 and C3147-2 had identical properties, and were therefore, equally limiting. The NRC staff confirmed that the licensee used values for the margin term that were appropriate based on the licensee's use of CF values from Table 2 of RG 1.99, Revision 2 and a material-specific value of the initial RT_{NDT}. Finally, the NRC staff's calculated ART values for the limiting beltline plates agreed with the licensee's calculated ART values.

Given the acceptability of the licensee's calculated ART value for the limiting beltline material to 22 EFPY, the NRC staff evaluated the licensee's revised P-T limit curves for acceptability by performing a finite set of check calculations based on information submitted by the licensee and by using the methodologies referenced in the ASME Code (as indicated in SRP 5.3.2). The NRC staff's independent calculations confirmed the licensee's determination regarding how the limiting RPV beltline material contributed to the definition of the NMP2 RPV P-T limit curves. The NRC staff verified that the licensee's proposed P-T limit curves satisfy the requirements in Section IV.A.2 of Appendix G to 10 CFR Part 50. Specifically, the NRC staff concludes that the P-T limit curves submitted by the licensee appropriately accounted for the limiting conditions defined by the material properties of the limiting beltline materials and were at least as conservative as those that would be generated by application of the methodology specified in

Appendix G to Section XI of the ASME Code, as modified by ASME Code Case N-640. In addition, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure flange region based on the reference temperature for the flange material. Section IV.A.2 of Appendix G to 10 CFR Part 50 states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature at the closure flange region that is highly stressed by the bolt preload must exceed the reference temperature of the material in that region by at least 160 °F for core critical operation, 120 °F for normal, non-critical core operation, and by 90 °F for hydrostatic pressure tests and leak tests. The NRC staff confirmed the licensee's limiting RT_{NDT} value of 10 °F for the flange material, which is limiting, based on information previously reported by the licensee and documented in the NRC staff's Reactor Vessel Integrity Database, as well as the acceptability of this value for the original P-T limits. Based on this limiting flange reference temperature, the NRC staff has determined that the proposed P-T limits have satisfied the above requirements for the closure flange region during all modes of normal operation and for hydrostatic pressure and leak testing.

Based on the above, the NRC staff concludes that the proposed P-T limit curves for the pressure test, core not critical, and core critical conditions satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code, as modified by ASME Code Case N-640. Therefore, the proposed P-T limit curves are acceptable for incorporation into the NMP2 TSs and shall be valid for 22 EFPYs of facility operation.

3.3 Evaluation of Fluence Calculations

The NRC staff previously found the licensee's vessel fluence methodology acceptable for both Nine Mile Point units (see safety evaluation supporting Unit No. 1 Amendment No. 183, dated October 27, 2003, Accession No. 032760696). Therefore, this review is limited to the application of that methodology to the NMP2 vessel fluence calculation for 22 EFPYs of operation.

The licensee's calculations were carried out using the DORT Code ("Two- and Three-Dimensional Discrete Ordinates Transport Version 2.7.3," Computer Code Collection CCC-543, Radiation Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN, June 1996) and the BUGLE-96 cross section library ("BUGLE-96, Coupled 47 Neutron, 20 Gamma Ray Group Cross Section Library Derived from ENDF/B-VI," Radiation Safety Information Computation Center, Oak Ridge National Laboratory, Oak Ridge, TN, March 1996). The licensee used (r, θ) and (r, z) modes to synthesize three-dimensional flux distributions from the core through the pressure vessel. The fast neutron scattering was treated with a P_3 approximation, and the angular quadrature with a S_8 approximation. The code, the cross sections, and the (r, θ) and (r, z) meshes are within the recommendations of RG 1.190, and therefore, are acceptable.

The licensee used the ORIGEN 2.1 Code to calculate the effects of burnup on the neutron source ("ORIGEN 2.1, Isotope Generation and Depletion Code Matrix Exponential Method," Radiation Safety Computational Center, Oak Ridge, TN, May 1999). This is an unusual practice because burnup information is available from the cycle-reload calculations. However, the licensee stated that the CASMO-SIMULATE data were not available. In the process of benchmarking the methodology, the licensee presented information that demonstrated that the ORIGEN and the refueling data for the fission source are nearly identical. The ORIGEN code, among others, calculates the fissionable isotope fractions and the average number of neutrons per fission, ν, and the average energy per fission, κ. The neutron source is then derived from

ν/κ; the value of this ratio changes with burnup. Use of the ORIGEN code for source calculation is acceptable because it was benchmarked to cycle-specific data. The peak vessel fluence calculated by the licensee for 22 EFPYs is 5.71×10^{17} n/cm^2.

The NRC staff reviewed the information submitted by the licensee to support a request for new P-T limit curves applicable to 22 EFPYs, and found the proposed fluence calculation acceptable. This finding was based on the fact that the methodology has been previously approved by the NRC staff for both Nine Mile Point units, and the computer code, neutron cross sections, and approximations used are within the guidelines of RG 1.190.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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 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 52235). 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 staff 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: C. Sydnor
L. Lois

Date: January 27, 2004

cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
Nine Mile Point Nuclear Station
P.O. Box 126
Lycoming, NY 13093

Mr. James R. Evans
LIPA
P.O. Box 129
Lycoming, NY 10393

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

Mr. Paul D. Eddy
Electric Division
NYS Department of Public Service
Agency Building 3
Empire State Plaza
Albany, NY 12223

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

C. Adrienne Rhodes
Chairman and Executive Director
State Consumer Protection Board
5 Empire State Plaza, Suite 2101
Albany, NY 12223-1556

Mark J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street, NW.
Washington, DC 20005-3502

Mr. Michael J. Wallace
President
Nine Mile Point Nuclear Station, LLC
c/o Constellation Energy Group
750 East Pratt Street
Baltimore, MD 21202

James M. Petro, Jr., Esquire
Counsel
Constellation Energy Group, Inc.
750 East Pratt Street, 5th Floor
Baltimore, MD 21202

Mr. Peter R. Smith, Acting President
New York State Energy, Research,
  and Development Authority
17 Columbia Circle
Albany, NY 12203-6399