October 27, 2003

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. 1 - ISSUANCE OF AMENDMENT RE: PRESSURE-TEMPERATURE LIMIT CURVES AND TABLES (TAC NO. MB6687)

Dear Mr. Katz:

The Commission has issued the enclosed Amendment No. 183 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 15, 2002, as supplemented by letters dated January 15, July 31, and September 15, 2003.

The amendment revised the reactor coolant system pressure-temperature limit curves and tables in Section 3/4.2.2, "Minimum Reactor Vessel Temperature for Pressurization," of the TSs. The revised curves and tables are effective up to 28 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. 50-220

Enclosures: 1. Amendment No. 183 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

October 28, 2003

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Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
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Accession Number: **ML032760696**

OFFICE	PDI-1\PM	PDI-1\LA	EMCB/SC	SRXB/SC	OGC	PDI-1/SC	
NAME	PTam	SLittle	SCoffin*	JUhle*	LZaccari	RLaufer	
DATE	10/2/03	10/2/03	8/11/03	9/9/03	10/7/03	10/24/03	

*SE transmitted by memo on these dates.

OFFICIAL RECORD COPY

DATED: October 27, 2003

AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT UNIT NO. 1

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)

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
License No. DPR-63

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 November 15, 2002, January 15, 2003, July 31, 2003, and 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. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 183, is hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 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: October 27, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 183

TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

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
84	84
85	85
86	86
87	87
88	88
89	89
90	90
91	91
92	92
93	93
94	94
94a	94a
94b	--
94c	--
94d	--

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-63

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated November 15, 2002, Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), submitted changes related to the reactor pressure vessel (RPV) pressure-temperature (P-T) limits in the NMP1 Technical Specifications (TSs). The licensee proposed to revise P-T limits which would be effective through 28 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. Simultaneously, the licensee requested an exemption from the requirements of Appendix G to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) in order to utilize ASME Code Case N-640; the licensee withdrew this exemption request by letter dated August 19, 2003, on the basis that Code Case N-640 had been found unconditionally acceptable in Regulatory Guide (RG) 1.147, Revision 13.

To support the proposed amendment, and in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) dated May 6, 2003, the licensee submitted additional information by letters dated January 15, July 31, and September 15, 2003. These letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The 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 P-T limit curves based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; General Design Criterion (GDC) 14, 30, and 31 of Appendix A of 10 CFR Part 50; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; RG 1.99, Revision 2; RG 1.190; and Standard Review Plan (SRP) Section 5.3.2. 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 irradiation. GL 92-01, Rev. 1, requested that licensees submit their RPV data for their plants 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 review of P-T limit curves. 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). The licensee has incorporated the 1989 Edition of the ASME Code into the NMP1 licensing basis for defining the ASME Code requirements which apply to the facility's ASME Code, Section XI program. Hence, with respect to the requirements of Appendix G to 10 CFR Part 50, it is the 1989 Edition of Appendix G to Section XI of the ASME Code which currently applies to the P-T limits in the NMP1 TS.

3.0 TECHNICAL EVALUATION

3.1 Background

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 to Section XI of the ASME Code 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 on stress intensities resulting from hydrostatic testing. Appendix G to Section XI of the ASME Code also requires a safety factor of 1.0 on 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 six 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 requires 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 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 term.

Guidance on the determination of ΔRT_{NDT} and the margin term is given in RG 1.99, Rev. 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, Rev. 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, Rev. 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.

3.2 New P-T Limits Curves

The licensee requested, pursuant to 10 CFR 50.90, an amendment to the NMP1 TS to revise the RPV P-T limit 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. 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 NMP1, as given in the TSs, includes Figures 3.2.2.a through 3.2.2.g, and associated Tables 3.2.2.a through 3.2.2.g. These figures and tables 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 $\leq$100 °F/hour, as well as for leak/hydrostatic test conditions. The proposed TS changes replace Figures 3.2.2.a through 3.2.2.e and associated Tables 3.2.2.a through 3.2.2.e with new figures and tables, providing the revised P-T limits for the above heatup and cooldown conditions, and leak/hydro test conditions. The licensee proposed that the revised P-T limits be effective for 28 EFPYs of facility operation. The licensee proposed to delete Figures 3.2.2.f and 3.2.2.g, and associated Tables 3.2.2.f and 3.2.2.g.

The licensee submitted ART calculations and P-T limit curves valid for up to 28 EFPY of facility operation. The licensee's ART calculations were based upon determinations of ΔRT_{NDT} and the margin term, which were calculated using values for the CF taken from RG 1.99, Rev. 2. The licensee determined that the most limiting beltline materials at the 1/4T and 3/4T locations were the upper shell plates (plate G-307-4/5). The ART values for the limiting upper shell plates at the 1/4T and 3/4T locations at 28 EFPY were determined as follows:

	1/4T Location	3/4T Location
Fluence	0.1759 X 10^{19} n/cm^2	0.07479 X 10^{19} n/cm^2
Chemistry Factor	173.85°F	173.85°F
ΔRT_{NDT}	93.7°F	62.8°F
Initial RT_{NDT}	40°F	40°F
Margin	34°F	34°F
ART	167.7°F	136.8°F

The licensee submitted information regarding the detailed fracture mechanics evaluations performed to establish the proposed NMP1 RPV P-T limits. The licensee carried out an initial benchmark evaluation using the ASME Code, Section XI, Appendix G methodology, without application of ASME Code Case N-640, for the purpose of reproducing the original P-T curves for the pressure test and heatup/cooldown conditions. This initial benchmark evaluation was carried out to ensure that the methodology used for developing the revised P-T curves would be completely consistent with the methodology applied to the P-T curves in the current TSs, with the exception of the use of Code Case N-640. Based on the results of this initial benchmark evaluation, the licensee concluded that the resulting P-T limits produced without the application of ASME Code Case N-640 were essentially identical to the P-T curves in the current TSs. Therefore, the licensee concluded that the revised P-T curve methodology, based on the Code Case N-640 modification to the requirements in Appendix G to Section XI of the ASME Code, would be otherwise consistent with the previous P-T curve methodology.

The licensee's submittal contained information on the throughwall temperature gradients resulting from heatup and cooldown transients and their determination of the applied stress intensity at the flaw tip, for both the 1/4T and 3/4T postulated flaws, due to thermal loading (i.e., K_{IT}). Values for the throughwall temperature gradients were obtained from the heat transfer analysis performed for the calculation of the original P-T limits, and these values were unchanged from those used for calculating the original P-T limits. 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. In accordance with Appendix G to Section XI of the ASME Code, 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 for the proposed amendment, 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 NMP1 P-T limit curves. Use of the lower bound K_{IC} curve in the development of P-T operating limits is technically correct. 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. The NRC staff concluded 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, the NRC staff agreed that implementation of the proposed P-T curves, as defined by the technical basis supported by ASME Code Case N-640, will maintain adequate margins of safety in protecting the RPV from brittle failure.

The NRC staff performed independent calculations of the ART values for all of the NMP1 beltline materials using the methodology in RG 1.99, Revision 2. Based on these calculations, the NRC staff verified that the licensee's limiting beltline materials for the RPV are the upper shell plates (plate G-307-4/5). Furthermore, the NRC staff verified that the materials represented in the surveillance program for NMP1 do not include the limiting beltline material. This confirmed that the licensee correctly utilized Table 2 of RG 1.99, Revision 2, as the means for obtaining values for the limiting plates' chemistry factor. Finally, the NRC staff confirmed that the licensee used a margin term that was appropriate based on its use of CF values from Table 2 of RG 1.99, Revision 2, and a material-specific value of the initial RT_{NDT}. Therefore, 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 28 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 NMP1 RPV P-T limit curves. The NRC staff verified that the licensee's proposed P-T limit curves satisfied the requirements

in Section IV.A.2 of Appendix G to 10 CFR Part 50. Specifically, the NRC staff concluded 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 the NRC staff's 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, which 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 40 °F for the non-beltline flange material, 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 this independent assessment, the NRC staff concluded that the licensee's proposed P-T limit curves were acceptable for operation of the NMP1 RPV through 28 EFPYs of facility operation.

3.3 Neutron Transport Calculations

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.

3.3.1 Benchmarking

The licensee cited the measured results of the Pool Critical Assembly (PCA) pressure vessel simulator, where well-characterized pressure vessel simulation experiments were conducted (see NUREG/CR-6454 (ORNL/TM-13205), "Pool Critical Assembly Pressure Vessel Facility Benchmark," July 1997). Measurements were made in a variety of arrangements simulating both pressurized- and boiling-water reactors (PWR and BWR) geometries. The measurements were performed with a wide range of dosimeters to enable spectral verification. The measurements and their associated uncertainties were reported.

The arrangement analyzed has a 12-cm gap between the core and the thermal shield, and a 13 cm gap between the thermal shield and the vessel. However, the core-to-shroud distances in NMP1 range from 10 to about 40 cm, and the shroud-to-vessel distance is about 40 cm. The arrangement chosen is not representative of a BWR geometry. In addition, the test excluded the rhodium-103 and uranium-238 detectors. The licensee's justification was that the rhodium-103 detector is not commonly used due to its short half-life, and, therefore, the cross sections are not well known. The licensee's basis for rejecting the uranium-238 dosimeter was due to difficulties in the generation of appropriate cross sections. In its May 6, 2003, RAI, the NRC

staff disagreed with this justification. The NRC staff stated that the purpose of benchmarking is to demonstrate the ability of the methodology to integrate all of the calculational activities; the above benchmark was missing information for the NRC staff to make a determination on its adequacy. By its July 31, 2003, letter the licensee provided detailed justification for the choice of geometry, and presented additional data for the rhodium-103 and uranium-238 dosimeters. The NRC staff reviewed the supplemental information in the licensee's July 31, 2003, letter and found the licensee's benchmarking to satisfy the criteria in RG 1.190, and is, therefore, acceptable.

3.3.2 Results of the Calculational Benchmark

NUREG/CR-6115 (BNL-NUREG-52395), "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," dated 1997 documents a BWR computational standard designed to be used to demonstrate that a Code has all the necessary routines, and that they are properly assembled. There are 6 dosimeters to be analyzed: 2 fission dosimeters (uranium-238 and neptunium-237), 2 in the over 1.0 MeV range (iron-54 and nickel-i58), and 2 in the upper end of the spectrum (copper-63 and titanium-46).

The licensee carried out its neutron transport analyses using the same Code as in NUREG/CR-6115 (reference: TORT-DORT-PC, Two- and Three-Dimensional Discrete Ordinates Transport, Version 2.7.3, RSIC Computer Code Collection, CCC-543, Oak Ridge National Laboratory, Oak Ridge TN, June 1996), but used DOTSOR instead of MESH for the conversion (x, y) → (r, θ) and BUGLE-96 instead of BUGLE-93, which was used in the standard. These changes would justify a few percent difference in the results obtained by the licensee compared to those in the standard.

The licensee's results are in good agreement with the results reported in NUREG/CR-6115, thus successfully demonstrating that the Code is working properly.

3.3.3 NMP1 Surveillance Capsule at 210-Degree Azimuthal

At the end of Fuel Cycle 12, the licensee removed and analyzed this surveillance capsule. At the same time, the licensee extracted two "boat" samples from the shroud to determine the fluence on the shroud welds. Because of an inconsistency between the iron and nickel dosimeter readings in the boat samples, the licensee investigated the details of power distribution for Fuel Cycle 12 in order to explain the dosimeter discrepancies. In its July 31, 2003, letter, the licensee reported its calculated surveillance capsule average fluence to be within 20% of the measured value, which is within the recommended margin of RG 1.190.

The NRC staff obtained and reviewed the NMP1 210-degree surveillance capsule report (MPM-398675, "Nine Mile Point Unit 1 210 Degree Surveillance Capsule Report," by MPM Technologies, Incorporated, March 1998). This capsule was removed at the end of Fuel Cycle 12 at 16.81 EFPYs. The report does not include any information on the methodology for the neutronic calculations, and in addition, indicates substantial discrepancies between the copper, iron, and nickel dosimeters. The report recommends the fluence value based on the copper results; yet copper activation represents a small part of the spectrum compared to iron and nickel. Iron and nickel results are also available. Because the neutron transport methodology was not presented and the very large discrepancies in the dosimeter response, the staff stated, in its May 6, 2003 RAI, that the NMP1 210-degree surveillance capsule results did not support the benchmark of the fluence calculational methodology.

The licensee's July 31, 2003, submittal addressed the discrepancies identified above. The licensee reanalyzed the NMP1 210-degree dosimetry and identified a power distribution discrepancy for Fuel Cycle 12. Likewise, Fuel Cycles 9 through 11 were also analyzed for revised flux distribution. The average copper and iron dosimeter measurement to calculation ratio (M/C) is 0.937 with a 1σ uncertainty of 9%. This is well within the RG 1.190 limits, and therefore, is acceptable. In conclusion, the analysis of the NMP1 210-degree surveillance capsule led to an acceptable benchmark.

3.3.4 Nine Mile Point Unit 2 (NMP2) Surveillance Capsule at 3-Degree Azimuthal

The licensee removed and measured the NMP2, 3-degree surveillance capsule at the end of Fuel Cycle 7. The capsule had been irradiated for 8.72 EFPYs (MPM-1200676, "Nine Mile Point Unit 2 3-Degree Surveillance Capsule Report," by MPM Technologies, Incorporated, December 2000). While NMP2 is equipped with jet pumps, the NRC staff notes that the 3-degree location surveillance capsule is not affected by the jet pumps. Neutronic analysis is included in this report and the methodology adheres to the guidance in RG 1.190. (At the time the report was prepared, RG 1.190 had not been issued; however, the licensee used DG 1053, which was essentially identical to the final guide). The capsule used copper and iron dosimeters. The results are within the guidelines of RG 1.190 after the implementation of a correction to the position of the dosimeters within the capsule. The NRC staff finds the 3-degree capsule benchmark acceptable because the licensee's methodology adheres to the guidance in RG 1.190, and the copper and iron dosimetry results, in terms of M/C values, are within the uncertainty bounds of RG 1.190.

3.3.5 Surveillance Capsules Used in the Proposed Benchmarking

The licensee removed, tested, and analyzed a total of five surveillance capsules from both NMP1 and NMP2. However, only two were used in the November 15, 2002, submittal for benchmarking analysis. As stated in RG 1.190, one of the objectives of benchmarking is to determine potential bias in the calculation of the best estimate which requires that all of the existing data be used. The November 15, 2002, application ignored the existence of three capsules and made no effort to determine the existence of a bias. The staff found that the benchmarking effort was incomplete in its RAI of May 6, 2003.

In response, the licensee's July 31, 2003, submittal accounted for all of the NMP1 and NMP2 surveillance capsules. The M/C values range from 0.937 to 1.073, which are well within the guidance of RG 1.190, and therefore, are acceptable.

3.3.6 Acceptability of Neutron Transport Calculation

Based on review of the licensee's submittals set forth above, the NRC staff finds that the licensee used acceptable methodology to derive the applicable fluence values. The licensee's September 15, 2003, letter submitted the revised version of the benchmarking report ("Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations," MPM-402781 (Revision 1), September, 2003, MPM Technologies, Inc.) to update the data, discussions, and conclusions contained in the licensee's July 31, 2003, letter. Accordingly, the NRC staff determines that the proposed P-T limit curves and tables are acceptable up to 28 EFPYs.

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 (67 FR 75882). 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: October 27, 2003

Nine Mile Point Nuclear Station, Unit No. 1

cc:

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