UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 2009

Mr. Keith J. Polson
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: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NO. MD9538)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 130 to Renewed Facility Operating License (FOL) No. NPF-69 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 14, 2008.

The amendment revises (1) the NMP2 TS Surveillance Requirement (SR) frequency in TS 3.1.1, "Control Rod Operability," and (2) Example 1.4-3 in TS Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The proposed changes are consistent with the Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," and NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 3.0. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the *Federal Register* on November 13, 2007 (72 FR 63935).

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

Sincerely,

Richard V. Guzman

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 130 to NPF-69
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
Renewed 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 14, 2008, 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 Renewed Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications

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. 130, 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

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: February 23, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 130

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
4	4

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
1.4-4	1.4-4
1.4-5	1.4-5
3.1-3-2	3.1-3-2
3.1-3-4	3.1-3-4

(1) Maximum Power Level

Nine Mile Point Nuclear Station, LLC is authorized to operate the facility at reactor core power levels not in excess of 3467 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(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. 130 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) Fuel Storage and Handling (Section 9.1, SSER 4)*

a. Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three containers high.

b. When not in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility.

c. The above three fuel assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

d. The New Fuel Storage Vault shall have no more than ten fresh fuel assemblies uncovered at any one time.

(4) Turbine System Maintenance Program (Section 3.5.1.3.10, SER)

The operating licensee shall submit for NRC approval by October 31, 1989, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities. (Submitted by NMPC letter dated October 30, 1989 from C.D. Terry and approved by NRC letter dated March 15, 1990 from Robert Martin to Mr. Lawrence Burkhardt, III).

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the license condition is discussed.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be met in MODE 1. --- Verify leakage rates are within limits.	 24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND	
	A.4 Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 ------------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -------------------------------	
	Fully insert inoperable control rod.	3 hours
	AND	
	C.2 Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	Deleted	
SR 3.1.3.3	------------------------------ NOTE ------------------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 05 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO

RENEWED 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 14, 2008 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML082270515), Nine Mile Point Nuclear Station, LLC (NMPNS or the licensee) submitted a license amendment request (LAR) for changes to the Nine Mile Point Nuclear Station Unit No. 1 (NMP1) Technical Specifications (TSs) and the Renewed Facility Operating License (FOL). The proposed amendment would revise the TS Surveillance Requirement (SR) frequency in TS 3.1.3, "Control Rod Operability," and Example 1.4-3 in TS Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

The proposed change is consistent with the Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action, and NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 3.0. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the *Federal Register* on November 13, 2007 (72 FR 63935).

NMPNS is proposing editorial variations from the applicable TS changes described in the modified TSTF-475, Revision 1, and the NRC staff's model safety evaluation dated November 13, 2007. These deviations do not affect the applicability of either the safety evaluation (SE) or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIIP.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(c)(3), states that TSs shall contain SRs "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." As discussed in Section 3.0 of this SE, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.0 TECHNICAL EVALUATION

Background of TSTF-475, Revision 1 (in reference to model SE dated November 13, 2007)

The Control Rod Drive (CRD) System is the primary reactivity control system for the reactor. The CRD System, in conjunction with the reactor protection system, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRD System that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

The CRD System consists of a control rod drive mechanism (CRDM) by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consist of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

CRT cracking was first discovered in 1975. It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time, hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

Subsequently, many boiling-water reactors have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods was still performed weekly. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods. Current operating experience now provides justification to reduce the notch test frequency for the fully withdrawn control rods as well. A review of industry operating experience did not identify any incidents of stuck control rods identified via performance of a rod notch surveillance for either partially or fully withdrawn control rods. Therefore, increasing the CRD notch testing frequency for fully withdrawn control rods from weekly to monthly will have minimal impact on the reliability of the CRD System.

Although not a basis for approving the frequency extension of notch testing for partially withdrawn control rods, General Electric (GE) Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," provides additional insight as to why a review of industry operating experience may not have identified any incidents of stuck control rods identified via performance of rod notch surveillance. The GE report is discussed in TSTF-475, Revision 1.

The GE report provides a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the TSs. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks haven been observed in the final improved CRT design. No collet housing failures have been noted since 1975.

Intergranular Stress-Corrosion Cracking (IGSCC) growth rates were evaluated using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress-corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

In addition to notch testing, other SRs are performed to verify the operability of the CRD System. Scram-time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. Also, the HCUs, CRD drives, and control rods are also tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. As a result, increasing the CRD notch testing frequency of fully withdrawn rods from weekly to monthly will have minimal impact on the reliability of the CRD System since additional SR are performed that verify the operability of the system.

It should be noted that approval to relax the SR frequency for notch testing of each fully withdrawn control rod is based on, in part, operational experience that has demonstrated no known CRD failures having been detected during the notch testing SR. Should the SR frequency relaxation result in a noticeable trend in failures, the licensee is expected to consider the need for revising the TS to include a more conservative testing frequency in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety." Administrative Letter 98-10 states that "Occasionally, as a result of licensee design-basis reconstitution efforts or NRC inspection efforts, licensees determine that specific values or required actions in TS may not assure safety. When this occurs, licensees typically conduct an evaluation and, if necessary, institute administrative controls that instruct the operators to maintain a more restrictive value for a particular parameter or to take a more conservative required action." Administrative Letter 98-10 also goes on to state that "Imposing administrative controls in response to an improper or inadequate TS is considered an acceptable short-term corrective action. The staff expects that, following the imposition of administrative controls, an amendment to the TS, with appropriate justification and schedule, will be submitted in a timely fashion."

NRC Staff Evaluation

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4 "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Notes of the "Surveillance" column. The NRC staff finds this change acceptable since the revision clarifies the example to make it consistent with the definition of specified "Frequency" provided in the second paragraph of Section 1.4 which states that the specified Frequency is referred to throughout this section and each of the Specifications of Section 3.0, SR Applicability. The specified Frequency consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

The licensee stated in their application that they have reviewed the basis for the NRC staff's acceptance of TSTF-475, Revision 1, and concluded that the basis is applicable to NMP2, and supports their adoption of the TSTF-475 changes. The licensee has proposed the following TS editorial changes which differ from those TS changes described in TSTF-475, Revision 1, and the model SE dated November 13, 2007. NMP2 chooses to designate SR 3.1.3.2 as "Deleted" and retain current SR 3.1.3.3, SR 3.1.3.4, and SR 3.1.3.5. This proposed variation will alleviate the requirement to make editorial changes listed in TSTF-475, Revision 1, for TS 3.1.3, 3.1.4 and associated TS Bases.

Also, the licensee states that during the NMP2 conversion to the STS (NUREG-1433) for TS 3.3.1.2, "Source Range Monitor (SRM) Instrumentation" by Amendment 91 dated February 15, 2000, Required Action E.2 was revised and presently reads "Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies." The term "fully" was also included in TS Bases 3.3.1.2. Therefore, these TSTF-475, Revision 1 changes are not necessary for NMP2.

The NRC staff has reviewed the licensee's proposal to amend NMP2 TS to revise the TS SR frequency in TS 3.1.3, "Control Rod Operability," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the high reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the staff concludes that the amendment request is acceptable.

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 amendments change 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 change the surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (73 FR 62567). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Grover
V. Cusumano
A. Lewin

Date: February 23, 2009

DATED: February 23, 2009

AMENDMENT NO. 130 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69 NINE MILE POINT, UNIT NO. 2

PUBLIC	
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SLittle	RidsNrrLASLittle
RGuzman	RidsNrrPMRGuzman
MDavid	RidsNrrDpr
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RElliott	RidsNrrDirsItsb
ACRS	RidsNrrAcrsAcnw&mMailCenter
GDentel, RI	RidsRgn1MailCenter
RGrover	
ALewin	
VCusumano	

February 23, 2009

Mr. Keith J. Polson
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: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NO. MD9538)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 130 to Renewed Facility Operating License (FOL) No. NPF-69 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 14, 2008.

The amendment revises (1) the NMP2 TS Surveillance Requirement (SR) frequency in TS 3.1.1, "Control Rod Operability," and (2) Example 1.4-3 in TS Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The proposed changes are consistent with the Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," and NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 3.0. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the *Federal Register* on November 13, 2007 (72 FR 63935).

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

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 130 to NPF-69
2. Safety Evaluation

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ADAMS Accession No.: ML090420301 * SE input provided by memo. No substantial changes made. NRR-058.

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