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

February 11, 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. 1 - ISSUANCE OF AMENDMENT RE: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NO. MD9539)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 200 to Renewed Facility Operating License (FOL) 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 August 18, 2008.

The amendment revises the NMP1 TS Section 3/4.1.1, "Control Rod System," to increase the Surveillance Requirement (SR) frequency associated with control rod exercising. The proposed change revises the required SR frequency from once each week to once every 31 days. The proposed change is 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 3.1. 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, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures:
1. Amendment No. 200 to DPR-63
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

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. 200
Renewed 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 August 18, 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. DPR-63 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. 200 , 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 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 200

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

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
3	3

Replace the following page of Appendix A, Technical Specifications, 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
30	30

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I:

Part 20, Section 30.34 of Part 30; Section 40.41 of Part 40; Section 50.54 and 50.59 of Part 50; and Section 70.32 of Part 70. This renewed license is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect and is also subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1850 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 200, is hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Deleted

secondary containment penetration flow path not isolated.

(e) If Specification 3.1.1a(1)(a) is not met while in the refueling condition, then:

Immediately suspend core alterations, except for fuel assembly removal, and

Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

(2) Reactivity margin - stuck control rods

Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.1.1a(1)(a) is met. In no case shall the number of non-fully inserted rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the cold shutdown condition. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

(2) Reactivity margin - stuck control rods

Each withdrawn control rod shall be exercised at a frequency of 31 days after the control rod has been withdrawn and power level is greater than the low power set point of the RWM. Insert each withdrawn control rod at least one notch.

This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 200 TO

RENEWED 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 August 18, 2008 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML082320016), 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 3/4.1.1, "Control Rod System."

Specifically, the proposed change revises the required SR frequency from once each week to once every 31 days. 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 3.1. 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).

TSTF-475, Revision 1, revised the reference Standard Technical Specifications (STS) by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from 7 days after the control rod is withdrawn and THERMAL POWER is greater than the Low Power Setpoint (LPSP) of the Rod Worth Minimizer (RWM) to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" (NUREG-1433 and NUREG-1434) and (2) revising Example 1.4-3 in Section 1.4 "Frequency" to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column (NUREG-1430 through NUREG-1434).

The purpose of the surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) Mechanism (CRDM), by which the control rods are moved, are components of the CRD System (CRDS), which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube which is conducive to control rod insertion. A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test. The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the TSTF-475 change to Example 1.4-3 in Section 1.4 "Frequency" is to clarify the applicability of the 25% allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column. The 25% allowance is cited in NMP1 custom Technical Specifications in Surveillance Requirement Applicability section 4.0.2 and the associated 4.0.2 Bases. Therefore, the licensee did not request this portion of TSTF-475 in their application dated August 18, 2008.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, "Protection against anticipated operational occurrence," requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRDS to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

3.0 TECHNICAL EVALUATION

3.0.1 Background

The CRDS at NMP1 is the primary reactivity control system for the reactor. The CRDS, 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 CRDS 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 CRDS.

The CRDS consists of a 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.

3.0.2 TSTF-475, Revision 1

The NRC staff approved TSTF-475 which revised the TS SR 3.1.3.2, "Control Rod OPERABILITY" in the STS (NUREG-1433 and NUREG-1434) from 7 days to monthly, based on the following: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise, the NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRDS.

As stated in the staff's model safety evaluation for TSTF-475, Revision 1:

> According to the BWROG [Boiling Water Reactor Owners Group], at the time of the first CRT crack discovery in 1975, each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. 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.
>
> 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.
>
> Subsequently, many BWRs [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 are 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 a 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.

In response to the NRC staff RAIs [request for additional information] and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and a General Electric (GE) Nuclear Energy Report entitled, "CRD Notching Surveillance Testing for Limerick Generating Station" (CRDNST). The GE Report provided 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 have been observed in the final improved CRT design.

[Neither the BWROG nor the NRC staff were able to find evidence of a collet housing failure since 1975]. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. For instance, on a numerical basis, based on the BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6,590 rod notch tests for a BWR/4 plant and 9,284 for a BWR/6 plant. If all BWRs operating in the U.S. are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, [although not a basis for approving the frequency extension of notch testing], the IGSCC [intergranular stress-corrosion cracking] crack growth rates were evaluated at Limerick Generating Station, 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 a 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.

Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action" as described in the license amendment request.

- Due Date/Event: This commitment will be implemented within 60 days from the date of the approval of the proposed amendment.

The NRC staff has reviewed the licensee's proposal to amend existing NMP1 frequency in SR 4.1.1a.(2). The NRC staff finds that the proposed changes to the NMP1 TS are consistent with the changes approved by the staff in TSTF-475, Revision 1, and, therefore, finds these changes acceptable. Specifically, the NRC staff finds that the proposed TS revision to NMP1 Section 3/4.1.1, "Control Rod System," will have a minimal effect on the high reliability of the CRD system while reducing the opportunity for potential reactivity events; thus, meeting the requirement of 10 CFR Part 50, Appendix A, GDC 29. Therefore, the NRC staff concludes that the proposed change to NMP1 TS SR 4.1.1.a.(2) 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 62568). 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

Date: February 11, 2009

February 11, 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. 1 - ISSUANCE OF AMENDMENT RE: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NO. MD9539)

Dear Mr. Polson:

The Commission has issued the enclosed Amendment No. 200 to Renewed Facility Operating License (FOL) 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 August 18, 2008.

The amendment revises the NMP1 TS Section 3/4.1.1, "Control Rod System," to increase the Surveillance Requirement (SR) frequency associated with control rod exercising. The proposed change revises the required SR frequency from once each week to once every 31 days. The proposed change is 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 3.1. 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. 50-220

Enclosures:
1. Amendment No. 200 to DPR-63
2. Safety Evaluation

cc w/encls: Distribution via Listserv
DISTRIBUTION:
(See next page)

ADAMS Accession No.: ML090160353 * SE input provided by memo. No substantial changes made. NRR-058.

OFFICE	LPLI-1/PM	LPLI-1/LA	ITSB/BC	OGC	LPLI-1/BC
NAME	RGuzman	SLittle	RElliott*	EWilliamson	MKowal
DATE	1/22/09	1/22/09	1/21/09 SE DTD	2/10/09	2/11/09

OFFICIAL RECORD COPY