March 27, 2003

Mr. John T. Conway
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: SHUTDOWN MARGIN REQUIREMENTS (TAC NO. MB6940)

Dear Mr. Conway:

The Commission has issued the enclosed Amendment No. 180 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 December 19, 2002.

The amendment revised the NMP1 TSs to add the definition of shutdown margin (SDM), incorporate new, more restrictive SDM limits, add the associated limiting condition for operation actions and completion times for each applicable operating condition if the SDM is not met, and add surveillance requirements for verifying SDM.

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. 180 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: **ML030860122**

OFFICE	PDI-1\PM	PDI-1\LA	RORP/SC	SRXB/SC	OGC	PDI-1/SC
NAME	PTam	SLittle	RDennig*	RCaruso*	RWeisman	RLaufer
DATE	3/14/03	3/13/03	1/28/03*	3/20/03*	3/26/03	3/27/03

*SE transmitted by memo on date shown.

OFFICIAL RECORD COPY

DATED: March 27, 2003

AMENDMENT NO. 180 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
KKavanagh
EKendrick
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. 180
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 December 19, 2002, 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. 180, 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: March 27, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 180

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
8	8
29	29
–	29a
30	30
31	31
–	31a
36	36

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 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 December 19, 2002, Nine Mile Point Nuclear Station, LLC (the licensee) submitted a request to revise portions of the Technical Specifications (TSs) for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The proposed changes would add the definition of shutdown margin (SDM), incorporate new, more restrictive SDM limits, add the associated limiting condition for operation (LCO) actions and completion times for each applicable operating condition if the SDM is not met, and add surveillance requirements (SRs) for verifying SDM. The proposed amendment also eliminates the restriction requiring SDM demonstration in the cold shutdown condition. However, the option for SDM demonstration in the cold shutdown condition is retained consistent with the existing special test exception. Accordingly the Nuclear Regulatory Commission (NRC) staff reviewed the licensee's proposed changes to the TSs.

2.0 REGULATORY EVALUATION

Shutdown margin requirements are specified to ensure: (1) the reactor can be made subcritical from all operating conditions, transients, and design basis events; (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. These requirements are satisfied by the control rods, as described in General Design Criteria (GDC) 26 of Appendix A, Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions. [NMP1 was constructed before the GDC were promulgated. However, when the licensee petitioned the Atomic Energy Commission to convert its provisional operating license to full-term operating license, the licensee stated (reference Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License, dated July 1972) that the unit meets the intended safety function of GDC-26, "Reactivity control system redundancy and capability."]

The control rod drop accident (CRDA) analysis assumes the core is subcritical with the highest worth control rod withdrawn. Also, SDM is assumed as an initial condition for the control rod withdrawal error during refueling. Prevention or mitigation of reactivity insertion events, such as those described above, is necessary to limit energy deposition in the fuel to prevent significant fuel damage. Adequate SDM provides assurance that inadvertent criticality events and potential CRDAs involving high worth rods will not cause significant fuel damage. Accordingly, SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion into the plant TSs.

3.0 TECHNICAL EVALUATION

3.1 TSs Section 1.32

The current NMP1 TSs do not define SDM. The licensee proposed to add a new Section 1.32 regarding SDM with the following wording:

> SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:
>
> a. The reactor is xenon free,
>
> b. The moderator temperature is 68°F, and
>
> c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

The proposed definition of SDM provides a clear understanding of the assumptions related to the calculation of SDM. The definition of SDM will also be applicable throughout the TSs and associated bases. Therefore, the NRC staff concludes that the addition of the proposed definition of SDM acceptable. In addition, the proposed wording is consistent with the definition of SDM in NUREG-1433, Revision 2, "Standard Technical Specifications, General Electric Plants, BWR [Boiling Water Reactor]/4."

3.2 TSs Sections 3.1.1a(1)(a) and 3.1.1a(1)(b)

LCO 3.1.1a(1) currently requires only that "the core loading shall be limited to that which can be made subcritical in the most reactive condition in the operating cycle with the strongest control rod in its full-out position and all other operable rods fully inserted." The associated SR 4.1.1a(1) currently requires that "[s]ufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent $\Delta k/k$ that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted." The licensee's proposed wording would require that the SDM under all operational conditions shall be equal to or greater than 0.38 percent $\Delta k/k$ with the highest worth control rod analytically determined, or 0.28 percent $\Delta k/k$, with the highest worth control rod determined by test.

This proposed change would incorporate new, more restrictive, SDM limits without requiring a specific SDM test in the cold shutdown condition. Consistent with the approved fuel vendor's methodology, the SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or the highest worth rod is determined by testing. An SDM demonstration relying on an analytical determination of the high worth rod necessitates the inclusion of additional margin to account for the calculation uncertainty. If one or more control rods are determined to be inoperable, as defined in TSs

Section 3.1.1a(2), while in the power operating condition, then a determination of whether Section 3.1.1a(1)(a) is met must be made within 6 hours or TSs Section 3.1.1a(1)(c) applies. In addition, the licensee proposed to change Section 4.1.1a(1) to require SDM to be verified both during the fuel loading sequence, and during the plant startup process following a refueling outage. Because this proposed requirement is more restrictive than the current requirement, and provides acceptable methods of determining SDM, the NRC staff finds it acceptable.

3.3 TSs Sections 3.1.1a(1)(c), 3.1.1a(1)(d), and 3.1.1a(1)(e)

NMP1 currently does not have any required actions or completion times associated with the SDM limits not being met. The addition of required actions and completion times would be a more restrictive change to the current licensing basis. The licensee proposed to revise Section 3.1.1a(1) to provide required actions and associated completion times if the SDM is not met under all operating conditions defined in TSs Section 1.1.

For power operating conditions, the licensee proposed a new Section 3.1.1a(1)(c), which would require restoration of compliance with Section 3.1.1a(1)(a) within 6 hours or be in a shutdown condition within the following 10 hours. "Power operating condition" is defined in Section 1.1 as "Reactor mode switch is in startup or run position," or the "Reactor is critical or criticality is possible due to control rod withdrawal." Therefore, SDM must be maintained within limits specified in Section 3.1.1a(1)(a) during startup and full-power operation, or the required actions of Section 3.1.1a(1)(c) must be completed. The allowed completion time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval. If the SDM cannot be restored, the plant must be brought to a shutdown condition in 10 hours to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed completion time of 10 hours is reasonable, based on operating experience, to reach a shutdown condition from full-power conditions in an orderly manner and without challenging plant systems. The proposed Section 3.1.1a(1)(c) is also consistent with the required actions and completion times for conditions A and B of LCO 3.1.1 in NUREG-1433, Revision 2.

For hot shutdown condition or cold shutdown condition, the licensee proposed a new Section 3.1.1a(1)(d) with the following conditions if Section 3.1.1a(1)(a) cannot be met:

> Immediately initiate action to fully insert all insertable control rods, and
>
> Initiate action within 1 hour to restore secondary containment to operable status, and
>
> Initiate action within 1 hour to restore one emergency ventilation system to operable status, and
>
> Initiate action within 1 hour to restore isolation capability in each required secondary containment penetration flow path not isolated.

With SDM not within the limits specified in Section 3.1.1a(1)(a) while in the hot shutdown condition or the cold shutdown condition, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one emergency ventilation system is OPERABLE; and secondary containment isolation capability in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. Because this proposed requirement would result in the least reactive condition for the core, and provides means for control of potential radioactive releases, the NRC staff finds the addition of Section 3.1.1a(1)(d) to be acceptable. In addition, Section 3.1.1a(1)(d) is consistent with the required actions and completion times for conditions C and D of LCO 3.1.1 in NUREG-1433, Revision 2.

For refueling, the licensee proposed a new Section 3.1.1a(1)(e) with the following conditions if Section 3.1.1a(1)(a) cannot be met:

> Immediately suspend core alterations, except for fuel assembly removal, <u>and</u>
>
> Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

With SDM not within limits specified in Section 3.1.1a(1)(a) while in the refueling condition, the operator would immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities would not preclude completion of movement of a component to a safe condition. NMP1 TSs Section 1.13, “Core Alterations,” does not define control rod movement with the control rod drive hydraulic system as a core alteration. Therefore, control rod insertion is not a core alteration. Inserting control rods or removing fuel from the core will reduce the total reactivity and are, therefore, excluded from the proposed wording.

The proposed wording would require that action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This will continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted. In addition, the NRC staff notes that a means for control of potential radioactive releases is provided since secondary containment integrity (TS Section 1.12) is currently required to be fully operable in the refueling condition in accordance with Sections 3.4.1 through 3.4.5. Based on the discussion above, the NRC staff finds the addition of Section 3.1.1a(1)(e) to be acceptable. In addition, Section 3.1.1a(1)(e) is consistent with the required actions and completion times for condition E of LCO 3.1.1 in NUREG-1433, Revision 2.

3.4 <u>TS Section 4.1.1a(1)</u>

The licensee proposed to modify this SR to state that “the SDM shall be verified within limits: (a) Prior to each in vessel fuel movement during the fuel loading sequence, <u>and</u> (b) Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement.” The proposed SR provides the conditions under which the SDM must be demonstrated. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or fuel shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. The frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification. Based on the discussion above, the NRC staff finds the revision to Section 4.1.1a(1) to be acceptable. In addition, the revised Section 4.1.1a(1) is consistent with the SR 3.1.1.1 in NUREG-1433, Revision 2.

3.5 Administrative and Associated Bases Changes

The licensee proposed editorial changes to Section 3.1.1a(2), 3.1.1b(2), and 3.1.1f to incorporate cross-referencing corrections based on the technical changes discussed above. These proposed changes involve no technical information, are purely administrative, and are thus acceptable.

The licensee proposed revising the associated bases for Section 3.1.1 and 4.1.1 to reflect the proposed changes discussed above. Since the NRC staff has found the technical changes acceptable, as set forth above, and the associated TS bases changes appear consistent with the TS changes, the NRC staff has no objection to the associated bases changes.

The licensee proposed changes to the bases for 3.7.1 and 4.7.1, Shutdown Margin Demonstration. Currently the bases state that "[t]he shutdown margin demonstration has to be performed prior to power operation." The bases further state that "the shutdown margin demonstration will be performed in the cold shutdown condition with the vessel head in place. The shutdown margin demonstration will be performed prior to the reactor coolant system pressure and control rod scram time tests following refueling outages when core alterations are performed." The licensee proposed to revise these sentences to state that compliance with this special test exception is optional. In addition, Section 3/4.7.1 is retained to continue to provide the special testing requirements for performing an SDM demonstration in the cold shutdown condition. Because the proposed changes to the bases of Sections 3.7.1 and 4.7.1 are consistent with the TS changes the NRC has found acceptable, as described above, the NRC staff has no objection to them. Furthermore, the proposed changes to the bases of Sections 3.7.1 and 4.7.1 are consistent with the intent of the LCO 3.0.7 and 3.10.8 bases in NUREG-1433, Revision 2.

3.6 Summary of Technical Review

The NRC staff has reviewed the licensee's application with the supporting documentation. Based on its review, the NRC staff concludes that the proposed revised requirements are acceptable because they are more restrictive than those in the current TSs, and provide appropriate verifications to assure that the SDM is maintained within specified limits.

The NRC staff reviewed the licensee's proposed TS bases that reflect the proposed TS changes. The TS bases changes are consistent with the licensee's proposed TS changes, and the NRC staff has no objections to the bases changes presented in the licensee's application.

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 2806). 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: K. Kavanagh
E. Kendrick

Date: March 27, 2003

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