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

October 30, 2014

Mr. Christopher Costanzo
Vice President Nine Mile Point
Exelon Generation Company, LLC
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION OF TECHNICAL SPECIFICATIONS TO ADOPT TSTF-535, "REVISE SHUTDOWN MARGIN DEFINITION TO ADDRESS ADVANCED FUEL DESIGNS" (TAC NOS. MF2927 & MF2928)

Dear Mr. Costanzo:

The Commission has issued the enclosed Amendment No. 216 to Renewed Facility Operating License No. DPR-63 and Amendment No. 146 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated October 7, 2013, as supplemented by letter dated June 18, 2014.

The amendments modified the Nine Mile Point Units 1 and 2 Technical Specifications (TSs) definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68 °F or a higher temperature that represents the most reactive state throughout the operating cycle. This change is needed to address new Boiling Water Reactor fuel designs, which may be more reactive at shutdown temperatures above 68 °F.

The U. S. Nuclear Regulatory (NRC) staff issued a notice of opportunity for comment in the *Federal Register* on November 19, 2012; 77 FR 69507, on possible amendments to revise the plant specific TS, to modify the TS definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68 °F or a higher temperature that represents the most reactive state throughout the operating cycle, including a model safety evaluation and model NSHC determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the *Federal Register* on February 26, 2013 (78 FR 13100).

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,

Bhalchandra K. Vaidya, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-220 and 50-410

Enclosures:
1. Amendment No. 216 to DPR-63
2. Amendment No. 146 to NPF-69
3. Safety Evaluation

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

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 216
Renewed License No. DPR-63

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated October 7, 2013, as supplemented by letter dated June 18, 2014, 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 1;

   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 and Environmental Protection Plan

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 216, are hereby incorporated in the license. Exelon Generation, the licensee 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

Benjamin G. Beasley, 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: October 30, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 216

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
Page 3	Page 3

Replace the following pages of the 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
Page 8	Page 8

(2) Exelon Generation pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Exelon Generation 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) Exelon Generation 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) Exelon Generation 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. is hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(3) Deleted

1.28 (Deleted)

1.29 (Deleted)

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

1.32 Shutdown Margin (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

a. The reactor is xenon free,

b. The moderator temperature is ≥ 68° F, corresponding to the most reactive state, 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.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 146
Renewed License No. NPF-69

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated October 7, 2013, as supplemented by letter dated June 18, 2014, 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 1;

   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) <u>Technical Specifications and Environmental Protection Plan</u>

   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. 146, are hereby incorporated into this license. Exelon Generation Company, 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

Benjamin G. Beasley

Benjamin G. Beasley, 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: October 30, 2014

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at reactor core power levels not in excess of 3988 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 146 are hereby incorporated into this license. Exelon Generation 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.

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; 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.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

ATTACHMENT TO LICENSE AMENDMENT NO. 146

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
Page 4	Page 4

Replace the following pages of the 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
TS 1.1-6	TS 1.1-6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENTS NOS. 216 AND 146

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-63 AND NPF-69

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-220 AND 50-410

NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2

(TAC NOS. MF2927 & MF2928)

1.0 INTRODUCTION

By letter dated October 7, 2013, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13294A094), supplemented by letter dated June 18, 2014, (ADAMS Accession No. ML14195A005), Exelon Generation, (the Licensee) submitted a License Amendment Request (LAR) which proposed changes to the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, (NMP or NMPNS) Units 1 and 2. The LAR requests the adoption of approved Technical Specification Task Force (TSTF) traveler TSTF-535, Revision 0, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs," which is an approved change to the Standard Technical Specifications (STS). The proposed changes modify the TS definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68 °F or a higher temperature that represents the most reactive state throughout the operating cycle.

The supplemental letter dated June 18, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (November 12, 2013, (78 FR 67411)).

The licensee stated that the license amendment request is consistent with NRC-approved TSTF Traveler TSTF-535. The availability of this TS improvement was announced in the Federal Register on February 26, 2013, 78 FR 13100 as part of the consolidated line item improvement process.

2.0 REGULATORY EVALUATION

2.1 Background

In water-moderated reactors, water is used to slow down, or moderate, high-energy fast neutrons to low energy thermal neutrons through multiple scattering interactions. The low energy thermal neutrons are much more likely to cause fission when absorbed by the fuel. However, not all of the thermal neutrons are absorbed by the fuel; a portion of them are instead absorbed by the water moderator. The amount of moderator and fuel that is present in the core heavily influences the fractions of thermal neutrons that are absorbed in each.

Water-moderated reactors are designed such that they tend to operate in what is known as an under-moderated condition. In this condition, the ratio of the moderator-to-fuel in the core is small enough that the overall effectiveness of water as a moderator decreases with increasing temperature; fewer neutrons are absorbed in the moderator due to the decrease in its density, however, this is overshadowed by the reduction in the number of neutrons that moderate from high fission energy to the lower energy level needed to cause fission. The result is a decrease in power and temperature: a negative reactivity feedback effect where the reactor becomes self-regulating. However, if the amount of moderator becomes too large with respect to the amount of fuel, the reactor can enter an over-moderated condition. In this condition, the overall effectiveness of water as a moderator increases with increasing temperature and the reduction in the number of neutrons absorbed in the moderator outweighs the loss in neutrons reaching lower energies. This causes an increase in power that leads to a further increase in temperature creating a potentially dangerous positive reactivity feedback cycle.

As practical examples in support of the proposed changes to the definition of SDM, TSTF-535 discussed SDM with regards to GE14 and GNF2 fuels. TSTF-535 indicated that for historical fuel products through GE14, the maximum reactivity condition for SDM always occurred at a moderator temperature of 68 °F because these fuel products were designed so that the core is always under-moderated when all control rods are inserted, except for the single most reactive rod. In cores with GNF2 fuel, TSTF-535 stated that it is expected that the maximum reactivity condition at beginning of cycle will remain at 68 °F, but that later in cycle the most limiting SDM may occur at a temperature greater than this, indicating that with this fuel design the core could potentially achieve an over-moderated condition.

The licensee stated in its letter dated October 7, 2013:

> Nine Mile Point Nuclear Station, LLC (NMPNS) has reviewed the model safety evaluation dated February 19, 2013, as part of the Federal Register Notice of Availability (Reference 1). This review included a review of the NRC's evaluation, as well as the information provided in TSTF-535 (Reference 2). As described in the subsequent paragraphs, NMPNS has concluded that the justifications presented in the TSTF-535 proposal and the model safety evaluation prepared by the NRC are applicable to Nine Mile Point Unit 1 (NMP1) and Nine Mile Point Unit 2 (NMP2) and justify this amendment for the incorporation of the changes to the plant TS.

2.2 Technical Specification Changes

The licensee's adoption of TSTF-535 for NMPNS proposes to revise the TS definition of SDM to require calculation of SDM at the reactor moderator temperature corresponding to the most reactive state throughout the operating cycle (68 °F or higher).

The current definition of SDM in Section 1.1, "Definitions," of the [PLANT] TS is:

> 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 the 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 licensee proposes the following changes (shown in bold) to the NMP1 and NMP2 TS definition of SDM in accordance with TSTF-535:

> SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical **throughout the operating cycle** assuming that:
>
> a. The reactor is xenon free
>
> b. The moderator temperature is ≥ 68°F, and **corresponding to the most reactive state**; and
>
> c. All control rods are fully inserted except for the single control rod of the 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.

2.3 Regulatory Review

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criteria (GDC) 26, "Reactivity control system redundancy and capability," and GDC 27, "Combined reactivity control systems capability," respectively require that reactivity within the core be controllable to ensure subcriticality is achievable and maintainable under cold conditions, with appropriate margin for stuck rods; and that reactivity within the core be controllable to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Among other things, 10 CFR 50.36(c)(2)(ii)(B) requires the establishment of a limiting condition for operation (LCO) for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The TS definition of SDM and the LCOs placed on SDM serve, in part, to satisfy GDCs 26 and 27 by ensuring there is always sufficient negative reactivity worth available to offset the positive reactivity worth of changes in moderator and fuel temperature, the decay of fission product poisons, the failure of a control rod to insert, and reactivity insertion accidents. Given this margin, the core can be held subcritical for conditions of normal operation, including anticipated operational occurrences.

The following provides the licensee's explanation regarding meeting the GDCs discussed above.

- The licensee's Evaluation Against Criterion 26 and 27 for NMP1

  By letter dated May 19, 2014, ADAMS Accession No. ML14139A274, the NRC Staff requested the licensee provide additional information regarding a comparison of its plant-specific assessment of applicable regulatory requirements versus those specified in the model Safety Evaluation (SE) (ADAMS Accession No. ML12355A772) for the TSTF. Specifically, the staff's letter pointed to the licensee's statement in its application dated October 7, 2013, which stated:

  > The traveler and model safety evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). NMPI was not licensed to the 10 CFR 50, Appendix A GDC, while NMP2 was licensed to the GDC. The NMPI Updated Final Safety Analysis Report (UFSAR) provides an assessment against the GDC in Table I-1. This UFSAR table refers to the NMP1 Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License, July 1972, for the details of the assessment against the GDC current at that time. A review has determined that the plant-specific requirements for NMP1 are sufficiently similar to the Appendix A GDC as related to the proposed change.

  The Staff's letter also stated (from the licensee's letter dated June 18, 2014):

  > The Staff's review of the NMPNS UFSAR Table I-1 determined that this one page Table merely points to certain historical files. The Table does not provide adequate plant-specific licensing basis comparison to those specific General Design Criteria discussed above. Please explain how your plant licensing basis compares to the GDC discussed above, and how it demonstrates an adequate technical basis for adopting TSTF-535.

  In response to the staff's letter of May 19, 2014, the licensee's letter dated June 18, 2014, ADAMS Accession No. ML14195A005, provided the following additional information:

The Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License, July 1972, provided an analysis of plant design criteria for Nine Mile Point Unit No. 1 to the GDC criteria. The following is an excerpt from the Technical Supplement containing the analysis provided for GDC criterion 26 and 27:

Criterion 26 - Reactivity Control System Redundancy and Capability

The Station contains a control rod system and a liquid-poison system for the control of reactivity. These systems are based on different design principles and are independent. The control rod system, in conjunction with the use of burnable poison in the fuel and reactor coolant recirculation system flow control, has the capability of controlling reactivity changes resulting from load changes, long-term reactivity changes, xenon burnout and fuel burnup. Reactor shutdown by the control rod system, in conjunction with the reactor protection system, is sufficiently rapid to prevent fuel damage limits from being exceeded during any anticipated operational transients. The control rod system is designed with a positive means of insertion and is capable of maintaining the reactor subcritical under hot or cold conditions with the highest worth control rod in the fully withdrawn position. The liquid poison system is capable of bringing and maintaining the reactor core subcritical either in its hot or in its most reactive condition (cold, xenon-free) independent of the control rod system. The control rod system is described in the FSAR. The liquid poison system is described in the same document.

Criterion 27 - Combined Reactivity Control Systems Capability

As stated in the FSAR Volume I, Section VII, the liquid poison system is provided to bring the reactor to a cold shutdown condition at any time in core life independent of the control rod system capabilities. The most severe requirement imposed on the liquid poison system is to shut down the reactor from a full-power operating condition, assuming complete failure of the withdrawn control rods to respond to an insertion signal. The rate of negative reactivity insertion provided by the liquid poison system is designed to exceed the rate of reactivity gain associated with reactor cool down from the full power condition. The liquid poison and the emergency core cooling systems are separate and independent systems. However, when operated simultaneously they accomplish the dual function under postulated accident conditions of both controlling reactivity changes with appropriate margin for stuck rods and maintaining adequate core cooling.

The licensee further states (in its June 18, 2014 letter):

> . . .the Technical Supplement was submitted to the NRC in July 1972.
>
> Based on the analysis performed in the Technical Supplement, NMPNS believes that the plant specific requirements for NMP1 are sufficiently similar to the Appendix A GDC and represent an adequate technical basis for adopting TSTF-535.

The NRC Staff, in part, based upon the technical supplement mentioned above, granted the full term operating license to NMP1 on December 26, 1974. The NRC staff, therefore, concludes that the NMP1 licensing basis i.e., UFSAR documents that the NMP1 plant specific criteria requirements are an acceptable equivalent to the 10 CFR Part 50, Appendix A GDCs 26 and 27.

- Regulatory Evaluation of Applicability of TSTF-535 to NMP1

In the Enclosure, Section 2.2 of its Application dated October 7, 2013, the Licensee stated:

> The NMP1 TS are custom TS, and therefore the current SDM definition format and numbering varies slightly from the NRC Standard Technical Specifications (STS) (NUREG-1433) shown in TSTF-535, Revision 0, and the applicable parts of the NRC's model safety evaluation. The minor variations are administrative and do not affect the applicability of TSTF-535 to the NMP1 TS.

Based on its independent review of the NMP1 TS and the applicable TS Bases, the NRC staff finds that NMP1 TS and TS Bases are indeed very similar to the Standard Technical Specifications in NUREG-1433 and that the minor differences are administrative. Therefore, the NRC staff finds that NMP1 can adopt TSTF-535 with the consolidated line-item improvement process and refer to the model NSHDC and model SE from the Notice of Availability.

- The licensee's Evaluation Against Criterion 26 for NMP2

NMP2 USAR (Revision 20) provides the following specifics in meeting the intent of GDC 26:

> Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies containing boron carbide (B4C) or B4C and hafnium. Positive insertion of these control rods is provided by means of the CRD [control rod drive] hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup, and shutdown) via operator-controlled insertions and withdrawals. Control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.
>
> Circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. Separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting scram circuitry. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown. Design of the control rod system includes appropriate margin for malfunctions such as stuck rods in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are

selected prior to operation to achieve optimum core performance and, simultaneously, low individual rod worths. Operating procedures to accomplish such patterns are supplemented by the rod worth minimizer (RWM), which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved by the insertion of a small number of many independent control rods. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods does not hinder the capability of the control rod system to render the core subcritical.

The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to effect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the unlikely event that reactor flow is suddenly increased to its maximum value (pump runout), the core will not exceed fuel design limits because the power flow map defines the allowable initial operating states in such a way that the pump runout does not violate these limits.
The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing fuel with burnable poison (Gd_2O_3) to control high reactivity of fresh fuel. In addition, the standby liquid control system (SLCS) is available to add soluble boron to the core and render it subcritical, as discussed in Sections 3.1.2.27 and 9.3.5.

The USAR further states,

> ...redundancy and capabilities of the reactivity control systems satisfy the requirements of Criterion 26.

- <u>The licensee's Evaluation Against Criterion 27 for NMP2</u>

NMP2 USAR (Revision 20) provides the following specifics in meeting the intent of GDC 27:

There is no credible event applicable to the BWR [boiling-water reactor] which requires combined capability of the control rod system and poison additions by the ECCS [emergency core cooling system]. The BWR design is capable of maintaining the reactor core subcritical, including allowance for a stuck rod, without the addition of any poison to the reactor coolant. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and if protection system limits are reached, corrective action is initiated through an automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the CRD system. Response by the RPS [Reactor Program System] is

> prompt and total scram time is short.
>
> In the unlikely event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by the control rods alone as the reactor cools down subsequent to initial shutdown, the SLCS is activated manually to inject soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; hence, only decay heat is generated by the core, which can be removed by the RHR [Residual Heat Removal] system, ensuring that the core is always coolable.
>
> Design of the reactivity control systems assures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods.

The USAR further states:

> The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

3.0 TECHNICAL EVALUATION

3.1 Current Definition of Shutdown Margin

In BWR plants, the control rods are used to hold the reactor core subcritical under cold conditions. The control rod negative reactivity worth must be sufficient to ensure the core is subcritical by a margin known as the SDM. It is the additional amount of negative reactivity worth needed to maintain the core subcritical by offsetting the positive reactivity worth that can occur during the operating cycle due to changes in moderator and fuel temperature, the decay of fission product poisons, the failure of a control rod to insert, and reactivity insertion accidents. Specifically, Section 1.1, "Definitions," of the STS defines SDM as the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that the reactor is (1) xenon free, (2) the moderator is 68°F, and (3) all control rods are fully inserted except for the rod of highest worth, which is assumed to be fully withdrawn.

The three criteria provided in the definition help exemplify what has traditionally been the most reactive design condition for a reactor core. Xenon is a neutron poison produced by fission product decay and its presence in the core adds negative reactivity worth. Assuming the core is xenon free removes a positive reactivity offset and is representative of fresh fuel at the BOC [beginning of cycle]. The minimum temperature the reactor moderator is anticipated to experience is 68°F, making it the point at which the moderator will be at its densest and therefore capable of providing the highest positive reactivity worth. By assuming the highest worth rod is fully withdrawn, the core can be designed with adequate shutdown margin to ensure it remains safely shutdown even in the event of a stuck control rod, as required by GDCs 26 and 27.

Determination of the SDM under the aforementioned conditions yields a conservative result that, along with the requirements set forth in Section 3.1.1 of the TS, helps ensure:

a. the reactor can be made subcritical from all operating conditions and transients and design basis events,

b. the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and

c. the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

3.2 Proposed Definition of Shutdown Margin

The specified moderator temperature of 68 °F facilitates the maximum reactivity condition only if the core exists in an under-moderated condition. In addition to burnable poisons, many modern fuel designs also incorporate partial length rods for increased neutron economy, which are employed in order to extend the operating cycle. Both of these affect the ratio of moderator to fuel. The strong local absorption effects of the burnable poisons in fresh fuel make the core under-moderated. As burnable poisons are depleted during the fuel cycle, the core becomes less under-moderated, potentially leading to a slightly over-moderated condition wherein the core will be more reactive at a moderator temperature higher than the 68°F specified in the SDM definition. Thus, the maximum core reactivity condition and the most limiting SDM may occur later in the fuel cycle at a temperature greater than 68 °F. Consequently, calculation of the SDM at the currently defined moderator temperature of 68 °F may not accurately determine the available margin.

TSTF-535 therefore proposed a change to the definition of SDM to enable calculation of the SDM at a reactor moderator temperature of 68°F or a higher temperature corresponding to the most reactive state throughout the operating cycle. SDM would be calculated using the appropriate limiting conditions for all fuel types at any time in core life.

In support of the proposed change, TSTF-535 cited Topical Report NEDO-24011-A, Revision 18, "General Electric Standard Application for Reactor Fuel (GESTAR II)," dated April 20, 19, and 30, 2011 (ADAMS Accession Nos. ML111120040, ML111120044, ML111120046, and ML111120047, respectively). Section 3.2.4.1 of the Enclosure 3 GESTAR II (ADAMS Accession No. ML111120046) states:

> The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted.

The Traveler also cited SRP Section 4.3, which states the following concerning the review of control systems and SDM:

> The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation. The applicant shall discuss shutdown margins (SDM). Shutdown margins need to be demonstrated by the applicant throughout the fuel cycle.

Although the licensing basis for SDM in GESTAR II are only applicable for cores licensed with Global Nuclear Fuels methods, they are consistent with the review procedures set forth in the SRP, which are provided to help ensure compliance with GDCs 26 and 27. TSTF-535 stated that while the SRP does not prescribe the temperature at which the minimum SDM should be determined, the requirement of shutting down the reactor and maintaining it in a shutdown condition "at any time during operation" suggests that considering a range of thermal and exposure conditions would be appropriate in the determination of the minimum SDM. Because newer fuel designs employ elements such as partial length rods and burnable absorbers, which may cause the maximum core reactivity conditions and the most limiting SDM to occur later in the fuel cycle at a temperature greater than 68°F, the NRC staff agrees with the topical report assessment in this regard. Additionally, the NRC staff finds that allowing calculation of the SDM at the most limiting core reactivity condition is prudent with respect to ensuring compliance with GDCs 26 and 27 or their plant-specific equivalent, and concludes that the proposed changes to the NMPNS TSs are acceptable.

The impetus for TSTF-535 was to provide for a more broadly applicable SDM definition in recognition of modern fuel designs, for which the core may not be in its most reactive condition at 68°F. The proposed language will require the licensee to consider all temperatures equal to or exceeding 68°F, and all times in the operating cycle. This change places an additional responsibility on the licensee to identify the most limiting time-in-cycle and temperature, a change that is more conservative than the current definition and will ensure the licensee maintains adequate SDM as required by their current licensing basis. Therefore, the change is acceptable for NMPNS. The NRC staff also finds that the revised definition is consistent with the 10 CFR 50.36 requirements pertaining to LCOs, because it ensures that the LCOs for SDM consider a broadly conservative range of potential initial conditions in the anticipated operational occurrence analyses.

3.3 <u>Summary</u>

The NRC staff has reviewed the licensee's implementation of TSTF-535 proposed revisions to the definition of SDM. Based on the considerations discussed above, the NRC staff concludes that the proposed revisions are acceptable and will provide a conservative and improved approach to the calculation of SDM that ensures use of the appropriate limiting conditions for all fuel types at any time in the life of the core. Additionally, the NRC staff concludes the proposed changes to the definition of SDM will require the licensee to calculate SDM in consideration of the most limiting conditions in the core. Therefore, the revised SDM definition is acceptable for use with any current fuel design.

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. 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 (November 12, 2013, (78 FR 67411)). 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) there is reasonable assurance that 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 Contributor: Ravi Grover

Date: October 30, 2014

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/

Bhalchandra K. Vaidya, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-220 and 50-410

Enclosures:
1. Amendment No. 216 to DPR-63
2. Amendment No. 146 to NPF-69
3. Safety Evaluation

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