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

June 29, 2010

Mr. Samuel L. Belcher
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 REGARDING REMOVAL OF POSITION INDICATION FOR RELIEF VALVES AND SAFETY VALVES FROM TECHNICAL SPECIFICATIONS (TAC NO. ME1719)

Dear Mr. Belcher:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 205 to Renewed Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station (NMPNS), Unit No. 1 (NMP1), in response to your application dated July 2, 2009 (Agencywide Documents Access Management System (ADAMS) Accession No. ML091950415).

This amendment revises the Technical Specifications (TSs) by removing position indication for the relief valves and safety valves from TS 3.6.11, "Accident Monitoring Instrumentation." The amendment also corrects an editorial error in the title of Table 4.6.11, "Accident Monitoring Instrumentation Surveillance Requirement."

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,

Douglas V Pickett for

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

Docket No. 50-220

Enclosures:
1. Amendment No. 205 to DPR-63
2. Safety Evaluation

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

NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 205
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 July 2, 2009, 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, which is attached hereto, as revised through Amendment No. 205, 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 of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L Salgado

Nancy L. Salgado, 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: June 29, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 205

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 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 Page	Insert Page
269	269
270	270
272	272

(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. 205, is hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Deleted

TABLE 3.6.11-1

ACCIDENT MONITORING INSTRUMENTATION

	Parameters	Total Number of Channels	Minimum Number of Operable Sensors or Channels	Action (See Table 3.6.11-2)
1)	Deleted			
2)	Deleted			
3)	Reactor Vessel Water Level	2	1*	2
4)	Drywell Pressure Monitor	2	1	4
5)	Suppression Chamber Water Level	2	1*	4
6)	Deleted			
7)	Containment High Range Radiation Monitor	2	1	3
8)	Suppression Chamber Water Temperature	2	1	2

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance provided at least one Operable channel is monitoring that Parameter.

TABLE 3.6.11-2

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION - 1 Deleted

ACTION - 2

a. With the number of OPERABLE accident monitoring instrumentation channels less than the total Number of Channels shown in Table 3.6.11-1, restore the inoperable channel(s) to OPERABLE status within seven days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the minimum Channels OPERABLE requirements of Table 3.6.11-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION - 3

a. With the number of OPERABLE channels less than the total Number of Channels shown in Table 3.6.11-1, prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

   1) either restore the inoperable channel(s) to OPERABLE status within seven days of the event, or

   2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Parameter	Instrument Channel Test	Instrument Channel Calibration
(1) Deleted		
(2) Deleted		
(3) Reactor vessel water level	Once per quarter	Once during each major refueling outage
(4) Drywell Pressure Monitor	Once per month	Once during each major refueling outage
(5) Suppression Chamber Water Level Monitor	Once per quarter	Once during each major refueling outage
(6) Deleted		
(7) Containment High Range Radiation Monitor	Once per month	Once during each major refueling outage
(8) Suppression Chamber Water Temperature	Once per month	Once during each major refueling outage

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 205

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 July 2, 2009 (Agencywide Documents Access Management System (ADAMS) Accession No. ML091950415), Nine Mile Point Nuclear Station, LLC (NMPNS or the licensee) submitted a request for changes to the Nine Mile Point Unit No. 1 (NMP1) Technical Specifications (TSs). The proposed amendment would revise the TSs by removing position indication for the relief valves and safety valves from TS 3.6.11, "Accident Monitoring Instrumentation." The amendment would also correct an editorial error in the title of Table 4.6.11, "Accident Monitoring Instrumentation Surveillance Requirement."

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications" establishes the regulatory requirements for TS content. The requirements emphasize preventing accidents and mitigating accident consequences. Applicants are expected to incorporate into their TSs those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity into their TSs. In 10 CFR 50.36, the Nuclear Regulatory Commission (NRC or the Commission) requires that licensees' TSs include items in the following categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the rule does not specify the format and content for TS categories.

2.1 Improved Standard Technical Specifications Program

2.1.1 Interim Policy Statement on Technical Specification Improvements

The NRC and industry representatives sought to develop guidelines for improving nuclear power plant TS content and quality. On February 6, 1987, the Commission issued "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors"

(52 FR 3788). In September 1992, the Commission issued NUREG-1433, "Standard Technical Specifications—General Electric Plants (BWR/4)"; the NRC staff developed the Standard Technical Specifications (STS) using the guidance and criteria contained in the Commission's interim policy statement. The STS are a model for developing improved technical specifications (ITS) for General Electric plants. The interim policy statement criteria ensure that ITS would consistently reflect system configurations and operating characteristics for the boiling-water reactor (BWR)/4 design. In addition, the generic bases statements provide the basis for each of the STS requirements.

2.1.2 Final Policy Statement on Technical Specification Improvements

On July 22, 1993, the Commission issued its final policy statement indicating that by satisfying the guidance in the policy statement, licensees also satisfy section 182a. of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.36 (58 FR 39132). The final policy statement described the improved STS safety benefits and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments and for complete conversions to the improved STS. Furthermore, the final policy statement gave guidance for evaluating the required scope of the ITS and defined the guidance criteria for determining which of the LCOs and the associated surveillances should remain in the ITS. Using this approach, licensees should keep existing LCO requirements that fall within or satisfy any of the final policy statement criteria in the TSs but may relocate those LCO requirements that do not fall within or satisfy these criteria to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36(c)(2)(ii) (60 FR 36953, July 19, 1995).

2.1.3 Final Policy Statement Criteria

The final policy statement outlined the following four criteria:

(1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident [DBA] or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and

(4) a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

3.0 TECHNICAL EVALUATION

3.1 Evaluation for Deletion of Safety/Relief Valve Position Indication Technical Specifications

3.1.1 Delete Technical Specification 3.6.11, Table 3.6.11-1, Items 1 and 2, and Table 3.6.11-2, Action 1

Items 1 and 2 of Table 3.6.11-1 specify operability requirements for safety relief valve (SRV) position indication instrumentation. Action 1 of Table 3.6.11-2 specifies the associated actions for the LCOs of SRV position indication instrumentation. With less than the required minimum channels operable, the associated action statement (Action 1) requires the inoperable channel to be restored to operable status within 30 days or to be in at least hot shutdown within the next 12 hours. The required number of operable channels is one channel per valve (this is also the required minimum number of operable channels).

Accident monitoring instrumentation is provided to monitor variables and systems over their anticipated ranges for accident conditions, as appropriate, to ensure adequate safety during and following accidents. These variables are used by the control room operating personnel to perform their roles in the emergency plan for the evaluation, assessment, and monitoring of events, and execution of control room functions. Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions during and following an Accident," Revision 2, issued December 1980, listed five types (Types A–E) of variables for the purpose of aiding the designer in selecting the accident monitoring instrumentation and applicable criteria. Categories 1, 2, and 3 separate the type criteria into groups for a graded approach to requirements, depending on the importance to safety or the measurement of a specific variable.

In accordance with 10 CFR 50.36, the NRC staff requires that the TS include all plant-specific RG 1.97 Type A instruments and all RG 1.97 Category 1 instruments specified in the plant's safety evaluation report on RG 1.97. Type A instruments monitor primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBA events. Category 1 instruments are designed for full qualification, redundancy, continuous real-time display, and onsite (standby) power. Accordingly, the NRC staff's position to require that the TS include all plant-specific RG 1.97 Type A instruments and all RG 1.97 Category 1 instruments specified in the plant's safety evaluation report on RG 1.97 can be applied to NMP1.

In the application, the licensee analyzed the safety basis for SRVs using the final policy statement criteria of 10 CFR 50.36. The SRVs are a part of the primary success path in the updated final safety analysis report (UFSAR) accident analysis in that they can actuate to mitigate a DBA and, therefore, meet Criterion 3. Accordingly, TS 3.2.8, "Pressure Relief Systems—Safety Valves," and TS 3.2.9, "Pressure Relief Systems—Solenoid-Actuated Pressure Relief Valves (Overpressurization)," require their operability. However, the SRV redundant position indication is provided for each valve consisting of RG 1.97 Type D, Category 3, instrumentation (acoustic monitors and tailpipe thermocouples). Currently, TS 3.6.11, Table 3.6.11-1, requires NMP1 SRV position indication. The associated SRs appear in Table 4.6.11.

The licensee stated that SRV position indication does not detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (Criterion 1). Although the NRC staff does not agree with this statement, the staff also recognizes that SRV position indication is not the only means by which an open SRV would be detected. Additional indications would include reactor pressure vessel pressure, reactor water level, containment pressure, and suppression pool parameters, including level, temperature, and pressure. SRV position indication is not a process variable, design feature, or operating restriction that is an

initial condition of a DBA or transient analysis (Criterion 2). Although the function of SRVs is part of the primary success path and the SRVs actuate to mitigate a DBA or transient, position indication for the SRVs does not form a part of the primary success path because the UFSAR accident analysis assumes that the SRVs function as designed (i.e., the accident analysis assumes no operator action based on SRV position for the SRVs to perform their primary success path function (Criterion 3)). Furthermore, the licensee stated that the loss of this instrumentation has no effect on the probabilistic risk assessment and has not been shown to be significant to public health and safety (Criterion 4).

Finally, failure of SRV position indication would not pose a significant challenge to the ability of the operating NRC staff to respond to a DBA or transient because the emergency operating procedures provide symptom-based instruction to the NRC staff in mitigating an upset condition of the plant (i.e., level, pressure, and temperature provide emergency operating procedure direction regardless of SRV status). In addition, the licensee stated that NUREG-1433, Revision 3, issued June 2004, does not list SRV position indication in STS 3.3.3.1, "Post-Accident Monitoring (PAM) Instrumentation."

The NRC staff reviewed the licensee's technical analysis of the proposed changes to the accident monitoring instrumentation section of the TS against the requirements in 10 CFR 50.36. Based on the discussion above, the NRC staff determined that the SRV position indication does not meet any of the four screening criteria for TS inclusion listed in 10 CFR 50.36(c)(2)(ii). In addition, the NRC staff also determined that the proposed changes are consistent with the requirements in the STS. Therefore, the NRC staff finds that the deletion of the proposed SRV position indication in the accident monitoring instrumentation of the NMP1 TS is acceptable.

3.2 Evaluation for Relocating Safety/Relief Valve Surveillance Requirements

3.2.1 Delete Technical Specification 3.6.11, Table 4.6.11, Items 1 and 2

Items 1 and 2 of Table 4.6.11 specify the SRs for SRV position indication instrumentation. A quarterly channel test and refueling interval channel calibration are required.

In Section 2.0 of the application, the licensee stated that the continued functionality of the position indicator instrumentation will be maintained by the existing surveillance and preventative maintenance procedures. Changes to these procedures are subject to the controls of 10 CFR 50.59, "Changes, tests, and experiments." In addition, the current classification of the position indicators for the SRVs as RG 1.97 Type D, Category 3, variables will continue to be reflected in the NMP1 UFSAR. Type D instruments provide information to indicate the operation of individual safety systems and other systems important to safety. Category 3 provides for high-quality commercial grade equipment that should be selected to withstand the specified service environment.

The licensee uses NEDO-33160-A, Revision 1, "Regulatory Relaxation for the Post Accident SRV Position Indication System," issued October 2006, for its technical evaluation. NEDO-33160-A, Revision 1, a topical report prepared by the BWR Owner's Group, was previously reviewed and approved by the NRC staff (ADAMS Accession No. ML062910165). NEDO-33160-A states that, for BWRs, reactor pressure vessel (RPV) pressure and suppression pool temperature instrumentation in combination with other instruments (e.g., RPV water level,

suppression pool level, and containment pressure) satisfy the accident detection and boundary integrity indication purpose as specified in RG 1.97 for the SRV position variable. This alternate instrumentation either meets or exceeds the Category 2 criteria. SRV position indication instrumentation provides backup information and does not need to be classified as a Category 2 variable. Therefore, NEDO-33160-A recommends that, for BWRs, SRV position indication be reclassified as a Type D, Category 3 variable. The operator would use SRV position indication as a confirmation of SRV opening; however, this position indication would be a backup to the main safety function of the automatic depressurized system to reduce RPV pressure and to enable the use of low-pressure emergency core cooling systems.

The NRC staff's evaluation of the licensing design basis finds that relocating the SRV position indicator SRs is consistent with the Commission's final policy statement criteria.

The NRC staff reviewed the licensee's technical analysis of the proposed changes to the SRV position indication TS and SRs against the requirements in 10 CFR 50.36. Based on the discussion above, the NRC staff determined that the SRV position indication does not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii) for items required to be maintained in the TS. In addition, the NRC staff also determined that the proposed changes are consistent with the requirements in the STS and the guidelines of RG 1.97. Therefore, the NRC staff finds that the deletions of the proposed SRV position indication in the TS and the SRs for NMP1 are acceptable.

3.3 Evaluation of Title Change to TS Table 4.6.11

The licensee has identified an editorial error in the title of TS Table 4.6.11, "Accident Monitoring Instrumentation Surveillance Requirement." The licensee has proposed to revise the singular form of the word "requirement" to its plural form of "requirements." Thus, the licensee proposes to rename the title of TS Table 4.6.11 to "Accident Monitoring Instrumentation Surveillance Requirements." The NRC staff finds this change to be editorial in nature and, therefore, acceptable.

3.4 Conclusion

The proposed changes remove the position indication for the relief valves and safety valves from NMP1 TS 3.6.11. In addition, the proposed changes correct an editorial error in the title of Table 4.6.11. Based on the NRC staff's review of the information provided in the licensee's July 2, 2009, submittal and the topics assessed in this safety evaluation, the NRC staff concludes that the proposed changes are acceptable. Accordingly, future changes to the SRV position indication instrumentation operability and SRs will be performed under regulatory control in accordance with the requirements of 10 CFR 50.59.

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 installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. 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 (74 FR 52826, Oct. 14, 2009). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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 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 Contributor: P. Chung

Date: June 29, 2010

June 29, 2010

Mr. Samuel L. Belcher
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 REGARDING REMOVAL OF POSITION INDICATION FOR RELIEF VALVES AND SAFETY VALVES FROM TECHNICAL SPECIFICATIONS (TAC NO. ME1719)

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Sincerely,

/RA/ by DPickett for

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

Docket No. 50-220

Enclosures:
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ADAMS Accession No.: ML101470493 *SE provided by memo. No substantial changes made. NRR-106 **See Email dated 6/24/10.

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