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

May 29, 2014

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

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING RELOCATION OF PRESSURE AND TEMPERATURE LIMIT CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (TAC NO. MF0345)

Dear Mr. Costanzo:

The Commission has issued the enclosed Amendment No.145 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP2). The amendment consists of changes to the Technical Specifications(TSs) in response to your application transmitted by letter dated November 21, 2012, as supplemented by letters dated March 25, July 31, September 6, November 4, December 13, 2013, and February 25, 2014.

Further, as a part of its application for the license transfer and conforming amendment of the Renewed Facility Operating License for Nine Mile Point Units and 2, in the letter dated March 28, 2014, Exelon Generation Company, LLC has stated that:

> "Prior to the license transfers, CENG made docketed submittals to the NRC that requested specific licensing actions, such as license amendment requests, relief requests, exemption requests, etc. Furthermore, in the application for the license transfers, Exelon stated that upon transfer of the licenses, Exelon would assume all current regulatory commitments made for these units. Accordingly, Exelon hereby adopts and endorses those docketed requests currently before the NRC for review and approval. Exelon requests that the NRC continue to process those pending actions on the schedules previously requested by CENG."

The amendment revised NMP2 Technical Specification (TS) Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits," by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). In addition, a new definition for the PTLR was added to TS Section 1.1, "Definitions," and a new section addressing administrative requirements for the PTLR was added to TS Section 5.0, "Administrative Controls."

Relocation of the P-T limit curves to the PTLR is consistent with the guidance provided in NRC approved General Electric Hitachi Nuclear Engineering (GEH) Licensing Topical Report, NEDC-33178P-A, Revision 1, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves." This topical report uses the guidelines provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The proposed TS

changes are consistent with the guidance provided in GL 96-03 as supplemented by Technical Specification Task Force (TSTF) traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS PTLR."

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,

BKVaidya

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

Docket No. 50-410

Enclosures:
1. Amendment No. 145 to NPF-69
2. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 145
Renewed License No. NPF-69

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

   A. The application for amendment by Nine Mile Point Nuclear Station, LLC (the licensee) date November 21, 2012, as supplemented by letters dated March 25, July 31, September 6, November 4, December 13, 2013, and February 25, 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 I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 145, 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 no later than July 18, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin Beasley

Benjamin 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: May 29, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 145

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 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 1.1-5	TS 1.1-5
TS 3.4.11-1 Through TS 3.4.11-5	TS 3.4.11-1 Through TS 3.4.11-5
TS 3.4.11-6 Through TS 3.4.11-10	
TS 5.6-4	TS 5.6-4

(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. 145 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).

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* 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

LEAKAGE (continued)	2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. Unidentified LEAKAGE All LEAKAGE into the drywell that is not identified LEAKAGE; and c. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

(continued)

1.1 Definitions (continued)

OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3988 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. 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.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation loop temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1	Restore parameter(s) to within limits.	30 minutes
	AND		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and Associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed if this Condition is entered. ----------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	---------- NOTE ---------- Only required to be performed during RCS heatup and cooldown operations, and RCS system leakage and hydrostatic testing. ----------	
	Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in the PTLR; b. RCS heatup and cooldown rates are within limits specified in the PTLR; and c. RCS temperature change during system leakage and hydrostatic testing is maintained within limits specified in the PTLR.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.11.3	---------- NOTE ---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	---------- NOTE ---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.5	---------- NOTE ---------- Only required to be met in single loop operation with THERMAL POWER ≤ 30% RTP or the operating jet pump loop flow ≤ 50% rated jet pump loop flow. ---------- Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is within limits specified in the PTLR.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in jet pump loop flow
SR 3.4.11.6	---------- NOTE ---------- Only required to be met in single loop operation when the idle recirculation loop is not isolated from the RPV, and with THERMAL POWER ≤ 30% RTP or the operating jet pump loop flow ≤ 50% rated jet pump loop flow. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is within limits specified in the PTLR.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in jet pump loop flow
SR 3.4.11.7	---------- NOTE ---------- Only required to be performed when tensioning the reactor vessel head bolting studs. ---------- Verify reactor vessel flange and head flange temperatures are within limits specified in the PTLR.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.8	---------- NOTE ---------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within limits specified in the PTLR.	30 minutes
SR 3.4.11.9	---------- NOTE ---------- Not required to be performed until 12 hours after RCS temperature ≤ 90°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within limits specified in the PTLR.	12 hours

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and system leakage and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Condition for Operation 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

   2. Surveillance Requirements 3.4.11.1 through 3.4.11.9.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. NEDC-33178P-A, Revision 1, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," dated June 2009. The licensee will calculate the fluence for determining the adjusted reference temperature using either; (1) values determined using an NRC-approved, RG 1.190-adherent method, or (2) a fluence estimate, which the licensee has verified as conservative, using an NRC-approved, RG 1.190-adherent method.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-69

EXELON GENERATION COMPANY, LLC

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated November 21, 2012 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML123380348 (Ref. 1)), as supplemented by letters dated March 25 (Ref. 2), July 31 (Ref. 3), September 6 (Ref. 4), November 4 (Ref. 5), December 13, 2013 (Ref. 6), and February 25, 2014 (Ref. 7), (ADAMS Accession Nos. ML130910038, ML13214A396, ML13254A156, ML13311A044, ML13353A277, and ML14063A474, respectively), Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Technical Specifications (TSs).

Further, as a part of its application for the license transfer and conforming amendment of the Renewed Facility Operating License for Nine Mile Point Units and 2, in the letter dated March 28, 2014 (ADAMS Accession No. ML14087A274), Exelon Generation has stated that:

> "Prior to the license transfers, CENG made docketed submittals to the NRC that requested specific licensing actions, such as license amendment requests, relief requests, exemption requests, etc. Furthermore, in the application for the license transfers, Exelon stated that upon transfer of the licenses, Exelon would assume all current regulatory commitments made for these units. Accordingly, Exelon hereby adopts and endorses those docketed requests currently before the NRC for review and approval. Exelon requests that the NRC continue to process those pending actions on the schedules previously requested by CENG."

The supplements dated March 25, July 31, September 6, November 4, December 13, 2013, and February 25, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's initial proposed no significant hazards consideration determination noticed in the *Federal Register* (FR) on March 12, 2013 (78 FR 15749).

The proposed amendment would revise the NMP2 TSs as necessary to relocate the pressure and temperature (P-T) limit curves and associated references currently contained in TS 3.4.11, "RCS [reactor coolant system] Pressure and Temperature (P/T) Limits," to an administratively controlled document known as a Pressure and Temperature Limits Report (PTLR). The request is submitted consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," (Ref. 8), as supplemented by TS Task Force (TSTF) traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS (Improved Standard Technical Specifications) 5.6.6, RCS PTLR," (Ref. 9). Specifically, the request would modify TS Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits," by replacing the existing reactor vessel heatup ,cooldown and leak test P-T limit curves with references to the PTLR. The request would also add a new definition for the PTLR to TS Section 1.0, "Definitions," and a new section addressing administrative requirements for the PTLR would be added to TS Section 6.0, "Administrative Controls," that would: (1) identify the individual TSs that address reactor coolant system pressure-temperature limits, (2) reference an NRC-approved methodology used to generate the PTLR, and (3) require that the PTLR and any revision or supplement be submitted to the NRC.

The proposed amendment also would implement new P-T limits for NMP2 that are valid up to a peak internal diameter (ID) fluence of 9.60×10^{17} n/cm^2, which corresponds to a projected 32 Effective Full Power Years (EFPY) of core operation. The revised P-T limits were developed using the methodology of NRC-approved General Electric Hitachi Nuclear Engineering (GEH) Licensing Topical Report, NEDC-33178P-A, Revision 1, " GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," (the GEH Methodology, Ref. 10) transmitted to boiling-water reactor (BWR) licensees via Reference 11, and would replace the current P-T limits which were valid to an estimated 22 EFPY. The non-proprietary version of the GEH methodology is Reference 12.

2.0 REGULATORY EVALUATION

In Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR), the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation, (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

In general, there are two classes of changes to TSs: (1) changes needed to reflect modifications to the design basis, as TSs are derived from the design basis, and (2) changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. In determining the acceptability of such changes, the NRC staff interprets the requirements in 10 CFR 50.36 using as a model the accumulation of generically approved guidance in the ISTS. For this review, the NRC staff used NUREG-1433, Revision 3, "Standard Technical Specifications, General Electric Plants BWR/4 (Ref. 13)."

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the acceptability of a facility's proposed PTLR based on the following NRC regulations and guidance:

(1) Appendix G to 10 CFR Part 50; Appendix G to 10 CFR Part 50 requires that facility P-T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2007 Edition through 2008 addenda. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

(2) Appendix H to 10 CFR Part 50; Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs.

(3) Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

(4) Generic Letter (GL) 92-01, Rev. 1; including GL 92-01, Rev. 1, Supplement 1; GL 92-01, Rev. 1 requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

(5) Standard Review Plan (SRP) Section 5.3.2; SRP Section 5.3.2 provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G methodology.

(6) GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The first requirement listed in the table, "Requirements for Methodology and PTLR," that appears in Attachment 1 to GL 96-03, is that the PTLR methodology shall describe how the neutron fluence is calculated. The methodology must describe transport calculation methods including computer codes and formulas used to calculate neutron fluence, and provide references.

(7) Section 3.2.4 of this safety evaluation (SE) provides the NRC staff's evaluation of the methodology of calculating the neutron fluence, including the regulatory requirements with respect to:

- NRC-Approved Methodology for use of General Electric-Hitachi (GEH) methods documented in NEDO-33178-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (ML092370487), including the condition set forth in the NRC staff safety evaluation that approved the GEH document,

- Acceptable Fluence Calculations, the guidance provided in RG 1.190 regarding an acceptable fluence calculation,

- GL 96-03 requirements discussed above.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

In its license amendment request (LAR), as supplemented, the licensee's proposed amendment would modify TS Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits," by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). In addition, a new definition for the PTLR would be added to TS Section 1.1, "Definitions," and a new section addressing administrative requirements for the PTLR would be added to TS Section 5.0, "Administrative Controls."

In its submissions, as supplemented, the licensee also noted that relocation of the P-T limit curves to the PTLR is consistent with the guidance provided in the GEH methodology. This topical report uses the guidelines provided in NRC GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The licensee stated that the proposed TS changes are consistent with the guidance provided in GL 96-03 as supplemented by TSTF traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," which allows the licensee to relocate their P-T curves and associated numerical limits (such as heatup and cooldown rates) from the plant TS to a PTLR, which is a licensee-controlled document. In order for the licensee to implement the PTLR, the analytical methods used to develop the P-T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant TS.

The proposed PTLR contains new P-T limit curves for NMP2 that are valid for a peak ID fluence of 9.60E+17 n/cm^2, which corresponds to a projected 32 EFPY of core operation.

Further, In its submissions, as supplemented, the licensee noted that the purpose of the GEH methodology is to provide BWRs with an NRC-approved report that can be referenced in plant TS to establish BWR fracture mechanics methods for generating P-T curves/limits, and other associated numerical limits, thereby allowing BWR plants to adopt the PTLR option. The licensee stated that the NMP2 P-T curves have been developed in accordance with the methodology and template in the GEH methodology, as documented in the PTLR provided in Attachment 3 of the licensee's submittal dated January 21, 2012. The licensee provided a revision to the PTLR as Attachment 2 to its November 4, 2013 letter. The licensee stated that the GEH methodology does not include development or licensing of vessel fluence methods, and that the methodology used to calculate RPV neutron fluence values utilized in the development of the NMP2 P-T limit curves in the PTLR is in accordance with RG 1.190 methods. The licensee further stated that NMP2 maintains an NRC approved RPV neutron fluence calculation methodology documented by MPM Technologies, Inc. in MPM-402781, Revision 1, September 2003 (Ref. 14), which has been approved by the NRC with

the issuance of NMPNS, Unit 1 (NMP1) License Amendment No. 183, and that the fluence values are in compliance with the recommendations of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Ref. 15).

3.2 NRC Staff Evaluation

3.2.1 PTLR Implementation

GL 96-03 requires that the licensee evaluate seven technical criteria to demonstrate the acceptability of its PTLR. The NRC staff examined the proposed PTLR and determined that it was developed from the Template PTLR found in the GE methodology report (Reference 10) and meets the seven technical criteria:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences.

The NRC staff's evaluation of the methodology of calculating the neutron fluence is contained in Section 3.2.4 of this SE. As detailed in Section 3.2.4, the NRC staff finds that the licensee's fluence methodology is acceptable for use, either as input to determine the PT limits, or as a verification that the PT limits to be used are based on a conservative fluence estimate. The licensee's revised proposed citation to TS 5.6.7.b will ensure that fluence values used in future revisions to the PTLR are acceptable, in light of the licensee's departure from the fluence requirements set forth in NEDO-33178-A and its approving NRC staff safety evaluation. Therefore, this criterion is met.

(2) The PTLR methodology describes the surveillance program.

The NMP2 PTLR indicated that NMP2 participates in the approved Boiling Water Reactor Vessel and Internals Program (BWRVIP) Integrated Surveillance Program (ISP), which meets the requirements of 10 CFR Part 50, Appendix H. Hence, the second criterion is met.

(3) The PTLR methodology describes how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics.

This is not applicable to NMP2 because it is a BWR.

(4) The PTLR methodology describes the method for calculating the adjusted reference temperature (ART) values using RG 1.99, Revision 2. The NMP2 PTLR indicated that RG 1.99, Revision 2, provides the methods for determining the ARTs for the beltline materials, with their chemistry factors determined by surveillance data information from the BWRVIP ISP. Hence, the fourth criterion is met.

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on ASME Code, Section XI, Appendix G, and the SRP.

The PTLR states that the P-T limits were calculated in accordance with the GEH methodology. This description is sufficient as the NRC staff documented in its SE that the methodology meets the fifth criterion. The NRC staff's SE that approved the PTLR methodology is part of Reference 10. Also, as documented in Section 3.2.3 of this SE, the NRC staff found the NMP2 PTLR appropriately implements the GEH methodology. Hence, the fifth criterion is met.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature.

As stated earlier, referencing the GEH methodology is sufficient because the PTLR methodology (Reference 10) contains detailed information regarding the minimum temperature requirements for boltup temperature and hydrotest temperature. The NRC staff documented in its SE that the GEH methodology meets the sixth criterion. The NRC staff's SE that approved the PTLR methodology is part of Reference 10. Also, as documented in Section 3.2.3 of this SE, the NRC staff found the NMP2 P-T limits meet the minimum temperature requirements of 10 CFR 50, Appendix G. Hence, the sixth criterion is met.

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

The NRC staff reviewed and approved the use of the ISP described in Appendix I of the GEH methodology report as documented in its SE. In addition, Appendix A of the PTLR describes the specifics of the ISP as it relates to NMP2. The footnotes to the table entitled "NMP2 Adjusted Reference Temperature values for 32 EFPY" explain how the ART values were determined from ISP data, if applicable, for the materials represented in the ISP. Also, proprietary Attachment 6 to the PTLR describes specifics of the ISP representative materials for NMP2, and how these materials were used in the ART calculation for NMP2. Hence, the seventh criterion is met.

Based on the above, the NRC staff concludes that implementation of the NMP2 PTLR is acceptable.

3.2.2 Beltline Materials Adjusted Reference Temperature

In Section 5.0 of the PTLR, the licensee stated that for NMP2, the plate heat C3147-1 is the limiting material for the beltline region, and that the limiting ART for the beltline LPCI N6 and Water Level Instrumentation N12 nozzle forgings and welds are also considered in the development of the beltline P-T curves. The table entitled "NMP2 Adjusted Reference Temperatures for 32 EFPY" in PTLR Appendix B provides details of the ART determination for all the NMP2 beltline and extended beltline materials.

The NRC staff performed independent calculations of the ART at 32 EFPY for each RPV beltline material listed in Appendix B of the PTLR. The NRC staff verified that the copper and nickel content and initial RT_{NDT} are consistent with both the Reactor Vessel Integrity Database (RVID) and the license renewal application for NMP2, which is more recent. Neither the RVID

nor the License Renewal Application (LRA) contain data for the low pressure coolant injection (LPCI) or water level instrumentation (WLI) nozzles, so values for the nozzle could not be checked. The NRC staff independently determined the chemistry factor (CF) and calculated the 32 EFPY ART values using the method of RG 1.99, Rev. 2. The CFs and resulting one-quarter RPV thickness (1/4t) ART values were found to be in agreement with the licensee's calculated values. The PTLR indicates that the plant-specific copper and nickel values for the WLI nozzles were not available, so the licensee determined the copper and nickel values based on a bounding estimate for forgings fabricated from SA-508 Class 1 materials, and that the values used for the NMP2 WLI nozzle were based on the mean plus one standard deviation of the data. In the Division of Engineering (DE), Vessels and Internals Integrity Branch (EVIB) request for additional information (RAI) 6 Part (c), the NRC staff requested justification for the use of copper and nickel values based on mean plus one standard deviation of the data rather than mean plus two standard deviations. The November 4, 2013, response to EVIB RAI 6 Part (c) cited two previous PTLR applications where nozzle copper and nickel values were determined based on mean plus one standard deviation of the relevant data. The response also cited RG 1.99, Rev. 2, which states that "conservative estimates (mean plus one standard deviation) based on generic data may be used if justification is provided. The NRC staff finds the licensee's response to EVIB RAI 6 Part (c) acceptable, since the method of estimating the WLI nozzle RT_{NDT} is consistent with NRC guidance.

Consideration of Surveillance Program Data

NMP2 is part of the ISP which combines all the surveillance programs for U.S. BWRs into a single integrated program. Under the ISP, the limiting plate and weld (target materials) in the NMP2 RPV are represented by similar materials under irradiation in other BWR plants. The representative plate material for NMP2 contained in a capsule at another plant is not of the same material heat as the target plate at NMP2. Therefore, per Footnote 5 to table "NMP2 Adjusted Reference Temperatures for 32 EFPY" in PTLR Appendix B, the ART of limiting NMP2 Plate C-3147-1 was determined using Position 1.1 of RG 1.99 (tables) and was not based on the CF of the ISP plate. Per Footnote 6 of the same table, the representative weld heat for NMP2, Heat 5P6214B, is not the same as the target weld at NMP2; however, this weld heat does exist in the NMP2 RPV beltline; therefore, the surveillance data was considered and RG 1.99 Revision 2, Regulatory Position 2.1 was used to adjust the CF. An adjusted CF of 53.7 °F was determined for weld Heat 5P6214B based on the ISP data. The ART based on this CF of -4 °F is provided in the line item for Heat 5P6214B under "Integrated Surveillance Program," in the same table. However, the table shows the CFs of 27 °F and 22.8 °F based on the plant-specific vessel weld chemistry (rather than the surveillance chemistry) were used to determine the ART of the two NMP2 lower shell axial welds that include this heat, which results in lower ART values. However, if the surveillance CF was used to determine the ART for Heat 5P6214B, this weld would still have a lower ART than the limiting weld and plate materials, therefore there would be no effect on the P-T limits.

3.2.3 P-T Limit Confirmatory Calculations

The GEH methodology requires the separate P-T curves be generated for the upper vessel region (including the feedwater nozzle), the beltline region, and the bottom head. The NRC staff therefore performed confirmatory calculations for each of these regions to ensure that the proposed P-T limits for NMP2 are at least as conservative as those determined using the

methodology of the ASME Code, Section XI, Appendix G for the pressure test (Curve A), core non-critical heatup and cooldown (Curve B), and core critical heatup and cooldown (Curve C).

For the beltline, the NRC staff independently generated the P-T curve using the methodology of the ASME Code, Appendix G, 2010 Edition through 2011 Addenda, based on the limiting beltline material ART of 51 °F. For a full penetration nozzle in the beltline region, such as the NMP2 LPCI nozzle, the GEH methodology specifies the P-T limits for the nozzle are calculated using the same method specified for the feedwater nozzle in the GEH methodology except that the effect of fluence on the ART must be considered. For the LPCI nozzle P-T limits, the NRC staff performed confirmatory calculations duplicating the method of calculating the P-T limits of the GEH methodology, and also using an independent method. For Curve B, the NRC staff's independent calculation for the LPCI nozzle considers only the cooldown transient, and conservatively assumed no temperature differential between the coolant and the 1/4t location. A stress concentration factor for the nozzle corner of 2.0 was used, which is typical for BWR full-penetration nozzles, and a thermal stress intensity K_{It} for the nozzle of 50 ksi$\sqrt{\text{in}}$ was used at all temperatures which is consistent with the average thermal stress calculated using the GEH methodology.

In addition to the beltline region, the methodology of the GEH methodology also requires that P-T limit curves be generated for the RPV bottom head and upper vessel region (for which the feedwater nozzle is controlling due to the stress concentration factor at the nozzle corner), and any nozzles located in the RPV beltline region. For the bottom head and WLI nozzle, the NRC staff performed confirmatory calculations of the P-T limits using the methodology of the GEH methodology since the ASME Code provides no guidance for determining P-T limits for these configurations.

The licensee clarified in its September 6, 2013, response to EVIB-RAI-5, that the NMP2 WLI nozzle is a partial penetration design nozzle and that the method of Appendix J of the GEH methodology is used to calculate the P-T limits for the WLI nozzle. The licensee also provided the calculation of the minimum temperature for Curve A (pressure test) and Curve B for the WLI nozzle, including the calculation of the R factor. As described in Section 4.3.2.1.4 of the GEH methodology, the R factor is used to adjust for nonlinear effects in the plastic region when the total stress exceed the material yield stress. Although the use of the R factor is part of the GEH methodology for full penetration nozzles, it is not clear that the GEH methodology allows the R factor to be used with the partial penetration nozzle methodology described in Appendix J of the GEH methodology. However, the NRC staff determined that additional conservative assumptions were made with regard to the determination of thermal stress values, which resulted in a conservative applied stress intensity for the WLI nozzle, compared to the requirements of the ASME Code, Section XI, Appendix G. Since the resulting applied stress intensity determined for WLI nozzle should still be very conservative even with the application of the R factor, the NRC staff finds the use of the R adjustment acceptable.

In its November 4, 2013, response to EVIB-RAI-6, the licensee clarified the source of the ART of the WLI nozzle forging of 39 °F listed in Appendix B of the PTLR, and how the ART used is consistent with the methodology specified in Appendix J of the GEH methodology. The NRC

staff finds the response to EVIB RAI 6, parts (a) and (b), to be acceptable, since the highest (most conservative) ART of the nozzle and adjacent shell material was used to generate the WLI nozzle P-T curve.

The NRC staff generated full P-T curves for the WLI nozzle for Curves A, B, and C. For both the LPCI and WLI nozzles, the PTLR did not show the specific P-T curves for the nozzles in Figures 1 and 2. However, the NRC staff's calculations confirmed the licensee's calculated temperature for the full operating or pressure test pressure of 1050 psig provided in the responses to EVIB-RAI-2 and EVIB-RAI-5. For the bottom head curve, the NRC staff's calculated P-T limits which are in agreement with the licensee's P-T limits for the bottom head provided in Figures 1 and 2 and Table 1 of the PTLR.

The NRC staff's calculations determined that the P-T limits for the LPCI nozzle, WLI nozzle, and bottom head are less limiting than NMP2's P-T limits, which are largely determined by the "flange notch" resulting from the minimum temperature requirements of 10 CFR 50, Appendix G.

The NRC staff's calculations verified, as stated in the NMP2 response to GGNS [Grand Gulf Nuclear Station, Unit 1] RAI-7 in Attachment 4 to the PTLR that for Curve C: (1) the upper vessel is bounding at pressures between 110 and 312.5 psig; (2) For pressures above 1250 psig, the beltline is bounding; and (3) For the remaining pressure ranges from 0 to 110 psig and between 312.5 and 1250 psig, the 10 CFR 50 Appendix G requirement are bounding. Most importantly, NMP2's proposed P-T limits bound all P-T curves generated by the NRC staff's confirmatory calculations.

The NRC staff also verified that the proposed P-T limits meet the minimum temperature requirements of 10 CFR 50, Appendix G, which contains additional requirements determined from the most limiting material (highest RT_{NDT}) in closure head flange and vessel flange regions. For NMP2, the closure flange region material with the highest RT_{NDT} is the vessel flange with an RT_{NDT} of 10 °F. The 10 CFR 50, Appendix G limits require that the pressure not exceed 20 percent of the preservice hydrostatic test (PSHT) pressure unless the temperature is greater than or equal to the RT_{NDT} of the limiting (highest RT_{NDT}) flange material plus (1) 90 °F for hydrostatic and leak tests, and (2) 120 °F for normal heatup and cooldown with the core not critical. The corresponding NMP2 values are 100 °F and 130 °F, which meet the 10 CFR 50, Appendix G requirements. For core critical operation, the pressure may not exceed 20 percent of the PSHT pressure unless the temperature is equal to the larger of the RT_{NDT} of the limiting flange material plus 160 °F or the minimum temperature for the inservice hydrostatic test. The corresponding NMP2 value is 170 °F, which is controlled by flange RT_{NDT} + 160 °F rather than the inservice hydrostatic test temperature. In addition, for core critical operation at greater than 20 percent of the PSHT temperature, 40 °F must be added to the ASME Code, Section XI, Appendix G limits. For BWRs operating with the core critical at less than or equal to 20 percent PSHT pressure, 10 CFR 50, Appendix G requires a minimum temperature equal to the limiting flange RT_{NDT} + 60 °F. For operation with the core not critical and pressure less than or equal to 20 percent of PSHT pressure, the minimum temperature must be not less than the limiting RT_{NDT} of the closure flange region. For NMP2, the minimum temperature of the P-T curves (boltup temperature) is 70 °F, thus meets both the minimum temperature for criticality at pressures less than 20 percent of PSHT pressure, and the minimum boltup temperature requirement. Therefore, the NMP2 P-T limits met all the 10 CFR 50, Appendix G minimum temperature requirements.

The NRC staff finds, based on its confirmatory calculations, that NMP2's proposed P-T limits for both all normal operating and test conditions meet the minimum temperature requirements of 10 CFR 50, Appendix G, and are as conservative or more conservative than P-T limits generated using the methods of the ASME Code, Section XI, Appendix G, and are therefore acceptable. In addition, the NRC staff finds that PTLR appropriately implements the GEH methodology.

3.2.4 Fluence Methodology

3.2.4.1 Regulatory Aspects

(a) Generic Letter 96-03 and Neutron Fluence

The generic framework for this TS change is set forth in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Ref. 8). The first requirement listed in the table, "Requirements for Methodology and PTLR," that appears in Attachment 1 to GL 96-03, is that the PTLR methodology shall describe how the neutron fluence is calculated. The methodology must describe transport calculation methods including computer codes and formulas used to calculate neutron fluence, and provide references.

(b) NRC-Approved Methodology

The licensee proposed to implement General Electric-Hitachi (GEH) methods documented in the GEH methodology. The NRC safety evaluation approving this methodology states as follows, regarding the fluence methods:

> ...this LTR [licensing topical report] does not describe the transport calculation methods including computer codes and formulas used to calculate neutron fluence. However, Section 4.2.1.2 of the LTR indicates that the neutron fluence will be determined using an approved methodology consistent with RG 1.190. Further, this section indicates the neutron fluence is defined in Section 4.2.1.2 of Attachment 1 and Appendix B of the PTLR. Section 4.2.1.2 of Attachment 1 requires the licensee to identify the report used to calculate the neutron fluence and to document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology. Therefore, this will be a plant-specific action item to be addressed by licensees. Since the LTR methodology indicates that the neutron fluence calculation methodology must comply with RG 1.190 and have been approved by the NRC, this criterion has been satisfied.

The considerations discussed above were set forth as the following condition[1] of the P-T limits methodology (i.e., the GEH methodology):

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[1] The term "condition" is used because the GEH methodology lists this "plant-specific action item" in Section 4.0, "Limitations and Conditions," of the approving safety evaluation. The term is interchangeable with "action item," which is subsequently used in this SE.

> The licensee must identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.

Based on the above, the NRC staff evaluates a plant-specific request to implement the GEH methodology using the guidance set forth in RG 1.190 to determine whether the calculational method used to perform the plant-specific fluence calculations are acceptable. As stated above, this determination should reflect the fact that the methodology has received NRC staff approval and adheres to the guidance contained in RG 1.190.

(c) Acceptable Fluence Calculations

RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDCs) contained in Appendix A to Title 10, "Energy" of the Code of Federal Regulations (10 CFR), Part 50. In consideration of the guidance set forth in RG 1.190, GDCs 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The guidance provided in RG 1.190 indicates that the following elements comprise an acceptable fluence calculation:

1. Determination of the geometrical and material input data,
2. Determination of the core neutron source,
3. Propagation of the neutron fluence from core to vessel, and into cavity, and
4. Qualification of the calculational procedure.

The NRC's review is performed to establish that elements 1 – 4, above, of the calculational method adhere to the regulatory position set forth in RG 1.190.

3.2.4.2 Technical Aspects

This safety evaluation concludes that the licensee is proposing to implement a PTLR in an unprecedented way, for two reasons. First, the licensee is using plant-specific methods to calculate reactor vessel neutron fluence. Second, the licensee is using its plant-specific fluence

methods as a means to confirm that the PTLR is based on a conservative fluence estimate, rather than the explicitly calculated fluence value. Therefore, it is necessary to review the applicable, precedential applications to explain how the NMP2 proposal differs.

After summarizing the necessary background information, this SE reviews the technical information provided by the licensee, provides the NRC staff evaluation, and explains the NRC staff's basis for concluding that the proposal is acceptable.

(a) Background and Regulatory Precedent

Since NMP2 is a boiling-water reactor (BWR), this background is limited to the PTLR and fluence methods that are commonly used at BWRs. A separate set of methods is available for use at PWRs, and is beyond the scope of this evaluation. In addition, this SE is not intended to provide an exhaustive list of fluence methods. Therefore, the review is limited to those methods, which have been approved by the NRC staff, and which have been referenced for use in PTLR-implementing amendments for BWRs.

(b) BWR P-T Limits Methodologies

At the time this SE was written, there were two NRC-approved methodologies, which BWR licensees could reference in license amendment requests seeking to implement a PTLR consistent with GL 96-03. The first is furnished by Structural Integrity Associates, as documented in SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors" (Ref. 16). The second was developed by General Electric-Hitachi Nuclear Energy (GEH), as documented in the GEH methodology. As both of these methods are NRC-approved, the reports each include NRC safety evaluations documenting their acceptability. The SEs approving these two methodologies are similar in a noteworthy respect: they both include a plant-specific action item to be addressed by licensees. This item relates to the fluence calculation.

As written, GL 96-03 requires an acceptable PTLR methodology to describe transport calculation methods including computer codes and formulas used to calculate neutron fluence. Neither of the BWR PTLR methodologies, however, satisfies this requirement in a stand-alone sense. Rather, each gives the implementing licensee the flexibility to select its own fluence methodology, provided that the selected methodology is NRC-approved and adherent to RG 1.190. The plant-specific action item requires each requesting licensee to identify the fluence methodology used to determine the fluence for input to the P-T limits and confirm that the methodology is NRC-approved and RG 1.190-adherent.

The next section summarizes the acceptable fluence methodologies that have been applied for use in PTLR implementation requests.

(c) Acceptable Fluence Methods

At the time this SE was written, BWR licensees had successfully implemented PTLRs, using one of the two methods described in Section 3.2.4.2(b), in concert with three different fluence methodologies. Oyster Creek was the first implementing licensee, referencing SIR-05-044-A in combination with the RAMA [Radiation Analysis Modeling Application] fluence methodology, and a second licensee followed this precedent. The GEH methodology has been implemented via license amendments for four facilities, with reference to the GEH fluence methods. One plant (NMP1) implemented SIR-05-044-A with reference to plant-specific methods, which had been previously approved in a plant- specific license amendment.

NMP2 proposes to use similar, plant-specific neutron fluence methods as NMP1, but in a slightly different fashion that is addressed in Section 3.2.4 of this evaluation. Since the GEH methodology (PTLR methodology described in NEDO-33178P-A) requires documentation that the "neutron fluence calculation will be performed using an approved neutron fluence methodology" (Section 3.2.4.1(b), above). It is necessary to provide a clear description of this methodology.

To aid in the description of the NMPNS plant-specific methods and determine what comprises the methodology, it is necessary to identify those documents, which address the 4 items, listed in Section 3.2.4.1(c), that comprise an acceptable fluence methodology. To elucidate this concept, the following subsections summarize the two generically approved fluence methodologies, as well as the methods approved for use at NMPNS.

(d) RAMA Fluence Methodology

The RAMA methodology is documented in Boiling Water Reactor Vessel Internals Project (BWRVIP) licensing topical report BWRVIP-114NP-A, "RAMA Fluence Methodology Theory Manual," and its companion reports[2] (Ref. 17). The series of four BWRVIP topical reports, and a separate, proprietary report documenting the results of a fluence evaluation for a Hope Creek flux wire dosimeter comprise the RAMA methodology. A single NRC staff safety evaluation considers these reports altogether and concludes that they are acceptable for referencing in licensing actions.

Owing to the above, the RAMA methodology may be considered as somewhat modular. It is comprised of methods described in five reports, because each report addresses certain elements (i.e., Items 1-4 identified in Section 3.4.2.1 of this SE) of the fluence methodology. BWRVIP-114NP-A and BWRVIP-121NP-A address Items 1-3 of the overall methodology, and BWRVIP-115NP-A, BWRVIP-117NP-A, and the Hope Creek flux wire dosimeter comparison address Item 4 of the methodology. The BWRVIP reports include appendices with NRC staff requests for additional information and responses to those requests.

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[2] The companion reports are BWRVIP-115NP-A, "RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems," BWRVIP-117NP-A, "RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," and BWRVIP-121NP-A, "RAMA Fluence Methodology Procedures Manual" (References 16, 17, and 18 respectively).

The first two entries in Table 1 provide references for the two precedential PTLR implementation amendments that used the RAMA methodology.

(e) GEH Fluence Methodology

The GEH fluence methodology is documented in GEH licensing topical report NEDO-32983-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations" (Ref. 21). The GEH methodology is different from the RAMA methodology in that a single, comprehensive report describes the computational framework *in toto*. Thus, the approved copy of the topical report describing the methodology includes an NRC staff safety evaluation, a main report that largely addresses the 4 elements of an acceptable fluence methodology, and appendices that include responses to NRC staff requests for additional information that were generated during the NRC review, and a large body of supporting data.

The final three entries in Table 1 provide references for the three precedential PTLR implementation amendments that used the GEH methodology. The following table summarized precedential applications, in which the licensees provided information in the associated license amendment request, indicating that the GEH fluence methodology had been used to determine the fluence inputs for the P-T limits:

Table 1. Summary of Precedent PTLR Amendments

Plant	Docket	PTLR Method	Fluence Method	Reference
Oyster Creek	50-219	SIR-05-044-A	RAMA	20
James A. FitzPatrick	50-333	SIR-05-044-A	RAMA	21
Grand Gulf	50-416	GEH	NEDO-32983-A	22
Monticello	50-263	GEH	NEDO-32983-A	23
Peach Bottom 2 Peach Bottom 3	50-277 50-278	GEH	NEDO-32983-A	24

(f) NMPNS Plant-Specific Methods

The licensee for NMPNS uses neither RAMA nor GEH methods to calculate fluence. Rather, NMPNS calculates reactor vessel neutron fluence in accordance with plant-specific methods. The calculations are performed by MPM Technologies, Inc. NMP1 has received NRC approval to implement a PTLR using the MPM methods in concert with SIR-05-044-A (Ref. 27).

Whereas the other methods used to determine reactor vessel neutron fluence are documented in generic licensing topical reports (i.e., a methodology) and have an NRC safety evaluation determining that the methods are acceptable for referencing in licensing actions, the MPM Technologies methods in use at NMP1 and NMP2 are described in plant-specific reports, as supplemented by responses to NRC staff requests for additional information. The basis for their acceptability is documented in NRC staff safety evaluations determining their acceptability for an intended application (e.g., for input to pressure-temperature limits for a given exposure period or as a basis to obtain relief from the inspection of circumferential shell welds).

Examples of safety evaluations for each unit are listed below:

- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 183 to Facility Operating License No. DPR-63," (Ref. 28).

- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 240 to Facility Operating License No. NPF-69," (Ref. 29).

The NRC staff distinguishes these methods from the RAMA or GEH methods in that the previously discussed methods have a distinct methodology described by one or a series of topical reports. For RAMA, it is those topical reports listed in the title of the approving safety evaluation, and for the GEH method, it is NEDO-32983-A. Regardless of the vendor, the approved methodology consists of an approving safety evaluation, a description of the system of methods sufficient to establish that the 4 elements of an acceptable fluence methodology identified in Section 3.4.2.1 of this SE are addressed, and responses to the NRC staff RAIs issued during the methodology review.

For NMPNS, a similar body of information encompasses the following documents, in addition to the safety evaluations listed above:

- MPM-402781, "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Transport Calculations," Revision 1, September 2003, (Ref. 14).
- Letter NMP1L 1749, "Request for Additional Information (RAI) – Amendment Application Re: Pressure-Temperature Limit Curves," including Attachment 1, "Response to Request for Additional Information," and Attachment 2, MPM-703782, "Response to NRC Request for Additional Information: Nine Mile Point Unit 1 P-T Limit Curves" (Ref. 30).

The inclusion of the NMP1L letter and its attachments in the overall methodology is significant in two respects. First, the NRC staff required the additional documentation before it could issue a safety evaluation concluding that the plant-specific methodology was acceptable for use at NMPNS. Second, the additional information in the response to the NRC request provides additional insight regarding the NRC review approaches, and specific changes to the methods and qualification required throughout the NRC staff review. Thus, despite that MPM-402781 had been revised to be consistent with the RAI response, the RAI response also contains important information about the methods approved for use at NMPNS.

In the present review, the licensee provided additional information about the current application of fluence calculations. The more recent fluence calculations include the effects of a power uprate, and the benchmarking data were reviewed to support a conclusion that an update to the uncertainty database was not necessary. The original methods are summarized in Section 3.2 of this SE, with additional detail describing how the present calculations differ from those previously reviewed.

In its RAI response dated November 4, 2013 (Ref. 5), the licensee provided an additional calculation, which provided updated information concerning the methodology used to calculate fluence at NMP2. This document supplements those identified above as further documentation of the methodology used to calculate the fluence:

- MPM-913991, "Fluence Extrapolation in Support of NMP2 P-T Curve Update," September, 2013 (Ref. 31).

This distinction, and the inclusion of updated reports in the licensee's methodology, is significant. The licensee does not have a specific, stand-alone, NRC-approved, generic methodology to reference in licensing actions, and the licensee may have applied the 10 CFR 50.59 criteria to revise its methodology without prior review and approval.

Since the licensee references a prior NRC approval for its method of evaluation, the NRC staff reviewed those methods already approved and reviewed the documentation submitted describing the present calculations to identify differences in the calculational framework and, where identified, determine their acceptability. Unless the NRC staff specifically identifies and discusses a difference, the NRC staff was not aware that the difference existed and did not consider it in its review. Any such differences may not be considered explicitly reviewed and approved by the NRC.

(g) Review of Fluence Methods

(1) Determination of the Geometric and Material Input Data

The NRC staff reviewed the licensee determination of the geometric and material input data using Regulatory Position 1.1, "Input Data," of RG 1.190. The licensee describes the problem materials and geometry in various locations in MPM-402781, Revision 1. Chapter 2 indicates that "the calculation uses a model of the reactor geometry that includes the significant structures and geometrical details necessary to define the neutron environment at locations of interest." The chapter also states that "all the data used in the models are documented and verified," and Chapter 5 provides further detail concerning the NMP2-specific problem model. The information in Chapter 5 discusses the octant problem geometry for the r-θ model, and explains how the model accounts for axial variations in moderator density. Temperature and pressure assumptions are also provided.

RG 1.190, Regulatory Position 1.1.1, recommends that the model be based on verified geometric data, and that water density be modeled based on full-power operating temperatures and pressures. The information provided by the licensee in MPM-402781, Revision 1, confirms that the problem materials and geometry are consistent with the Regulatory Position; hence, the NRC staff finds this element acceptable.

(2) Cross-Sections

In order to conform to the recommendation in RG 1.190, Regulatory Position 1.1.2, neutron cross sections should be based on the latest version of the Evaluated Nuclear Data File (ENDF). MPM-402781, Revision 1, Chapter 2 states that the calculations are based on the BUGLE-96

wide group cross section library, which is based on ENDF/B-VI. Although ENDF/B-VII has since been issued, the NRC staff still accepts the use of ENDF/B-VI based cross sections. RG 1.190 specifically mentions that ENDF/B-VI was the latest version of ENDF at the time of the RG's issuance, and also states, in Regulatory Position 1.1.2.2, that the BUGLE-96 may be used for light water reactor applications. Based on its review, the NRC staff determined that the licensee's cross section approximations are consistent with RG 1.190 and hence acceptable for NMP2 fluence calculations insofar as they support the development of PTLRs.

(3) Core Neutron Source

The core neutron source representation is reviewed using the guidance in Regulatory Position 1.2, "Core Neutron Source," of RG 1.190. The determination of the neutron source should include the temporal, spatial, and energy dependence together with the absolute source normalization. The spatial dependence should be based on depletion calculations that simulate core operation or measured data. A pin-wise source distribution should be used for peripheral bundles. The energy dependence should reflect the burnup-dependent power distribution, the power level, and the fuel management. The RG recommends that the source approximation account for the state-point dependence; however, averaging over the operating power distribution is acceptable. Source projections may be based on best-estimate approximations; however, fluence calculations based on these projections must be updated if changes in operating strategy compromise the accuracy of such projections. The burnup dependence of the fission energy spectrum should also be reflected in the source calculation. The RG accepts the use of octant-symmetric problem geometry, provided that jet-pump positioning is not quadrant-specific. For an r-θ model, a θ-mesh of at least 40 angular intervals over an octant must be applied.

The licensee describes the core neutron source calculation in Chapter 5.2 of MPM-402781, Revision 1. The octant-specific power distribution is represented, together with pin-wise power distributions for the peripheral bundles. The r-θ model used 48 azimuthal mesh points, and the licensee stated that only minor variations from octant symmetry were observed during an inspection of the fuel loading patterns. The NRC staff determined that these aspects of the neutron source calculation are acceptable, because they adhere to the guidance summarized above. Specifically, the θ-mesh exceeds the 40 angular intervals recommended, the licensee uses pin-wise source distributions for the peripheral assemblies, and information provided by the licensee justified the use of an octant-symmetric model.

The licensee used ORIGEN 2.1 to calculate the effects of burnup on the neutron source. Appendix A to MPM-402781 discusses the use of ORIGEN 2.1 further, as ORIGEN is an atypical data source for transport calculations. The appendix compares ORIGEN calculations to CASMO-4/SIMULATE-based core follow output for NMP2 Cycle 9. The comparison showed that the ORIGEN calculations produced a calculated fluence rate that was uniformly higher by 1.3 percent, and that the ORIGEN result differs in a conservative direction. The NRC staff previously reviewed the use of ORIGEN and determined that it was acceptable for NMP2, as described in the NRC staff safety evaluation dated January 27, 2004, stating as follows:

The licensee used the ORIGEN 2.1 Code to calculate the effects of burnup on the neutron source ("ORIGEN 2.1, Isotope Generation and Depletion Code Matrix Exponential Method," Radiation Safety Computational Center, Oak Ridge, TN,

May 1999). This is an unusual practice because burnup information is available from the cycle-reload calculations. However, the licensee stated [in MPM-703782, Attachment 2 to licensee's July 31, 2003, letter (ML032250595)] that the CASMO-SIMULATE data were not available. In the process of benchmarking the methodology, the licensee presented information that demonstrated that the ORIGEN and the refueling data for the fission source are nearly identical. The ORIGEN code, among others, calculates the fissionable isotope fractions and the average number of neutrons per fission, v, and the average energy per fission, κ. The neutron source is then derived from v/κ; the value of this ratio changes with burnup. Use of the ORIGEN code for source calculation is acceptable because it was benchmarked to cycle-specific data.

Based on the NRC staff's previous finding, and the information that the license provided in MPM-703782, which demonstrates that the use of ORIGEN results in slightly higher fluence rates than those obtained from cycle-specific core follow data, the NRC staff determined that the use of ORIGEN is acceptable. An appendix was added to MPM-402781, Revision 1, summarizing the ORIGEN–CASMO-4/SIMULATE comparison.

The licensee provided further information concerning the source representation in response to SRXB – RAI 1, Attachment 1, to the licensee's letter dated July 31, 2013 (Ref. 3). The licensee stated that "the fluence monitoring program includes updates of the neutron transport calculations based on actual exposure history at intervals of approximately every two refueling cycles, or based on a change in fuel type, core operating strategy, or other significant changes such as extended power uprate (EPU)." The NRC staff notes that this element is different from that described in MPM-402781, Revision 1, which states on Page 5, "No fluence projections are made in this benchmarking effort. Fluence projections for plant analyses use best-estimate fuel loadings." The NRC staff determined that the RAI response supplants the information in the benchmarking report, since the determination of P-T limits for future plant operation requires a fluence projection. The statement in the RAI response confirms that the licensee's calculations consider the temporal and energy dependence of the core neutron source, and that best-estimate projections used in the fluence calculations are confirmed periodically. As described in the first paragraph of this section, these approaches are consistent with Regulatory Position 1.2 of RG 1.190, and therefore are acceptable.

The NRC staff notes that MPM-913991 (Ref. 31), submitted with the licensee's November 4, 2013, letter, describes the fluence extrapolation consistent with the July 31, 2013, RAI response. Chapter 1 of MPM-913991 provides the following:

The present fluence evaluation uses a detailed cycle-by-cycle evaluation of fluence throughout the reactor system for the first 10 cycles of operation. The fluence for cycles 11 through 13 is estimated using the final cycle 10 results... The effect of changing the depleted uranium fuel blanket height from 12 inches to 6 inches starting with fuel inserted in cycle 11 is also accounted for in the evaluation. Starting with cycle 14, NMP2 is operating at an uprated power level 15% higher. A calculation of a typical cycle for this extended power uprate was documented... and this is used for evaluation of the cycle 14 fluence as well as in the extrapolation to 32 and 54 EFPY of operation.

Since this description indicates that the licensee's extrapolation accounts for past, present, and future operation, the NRC staff determined that they were acceptable.

The NRC staff determined that the core neutron source approximation used for the NMP2 fluence calculation is acceptable. The NRC staff's conclusion is based on the review of information contained in: (1) MPM-402781, Revision 1; (2) MPM-703782; (3) NRC staff safety evaluation dated January 27, 2004; (4) licensee letter dated July 31, 2013, as described in the preceding paragraphs, and (5) MPM-913991.

(4) Fluence Calculation

The NRC staff reviewed the fluence calculations performed using NMPNS plant-specific methods to determine whether they adhere to Regulatory Position 1.3, "Fluence Calculation," of RG 1.190. Regarding discrete ordinates transport calculations, the regulatory position states that "an S_8 fully symmetric angular quadrature must be used as a minimum for determining the fluence at the vessel." Regulatory Position 1.3.4 provides two acceptable ways to synthesize the 3-dimensional fluence from calculations of lower dimension.

Chapter 2.0 of MPM-402781, Revision 1, states that the fluence calculations employ the Discrete Ordinates Radial Transport (DORT) code, and use S_8 angular quadrature. The chapter also provides the equation used to perform the flux synthesis. The NRC staff determined that the DORT calculations in S_8 angular quadrature are acceptable because it is consistent with Regulatory Position 1.3.1 of RG 1.190; the synthesis approach is acceptable because it is consistent with Equation 4 of RG 1.190.

(5) Qualification of Calculation Methodology

The NMP2 fluence methodology is extensively qualified. The qualification includes the following comparisons:

- The Pool Critical Assembly benchmark
- The BWR calculational benchmark described in NUREG/CR-6115
- Capsule dosimetry and shroud boat analysis from NMPNS Units 1 and 2

The above items are described in Chapters 3-5 of MPM-402781, Revision 1. The NRC staff review of the benchmarking is described in NMP1L 1749, and also in Section 3.3, "Neutron Transport Calculations," of the NRC safety evaluation issued to NMP1, dated October 27, 2003. The NRC staff determined that, since the benchmarking results agreed to the experimental and measured data within 20-percent, the results were "well within the guidance contained in RG 1.190," and are therefore acceptable.

In response to SRXB – RAI 2 (Ref. 3), the licensee stated, "The MPM-402781 neutron transport methods have been applied for evaluation of the River Bend 183 degree capsule. The capsule data reported in BWRVIP-113NP, and the River Bend and Grand Gulf Cycle 1 dosimeter wires, have been used to confirm that the bias and uncertainties remain qualified for use at NMP2." The NRC staff reviewed BWRVIP-113NP (Ref. 32), and determined that the average measured to calculated (M/C) ratios for the iron and copper samples in the capsule agreed within the RG 1.190-recommended 20 percent, which is acceptable. The NRC staff did not evaluate the Grand Gulf Cycle 1 dosimeter wire results because these results have not been submitted for

NRC staff review. Based on the River Bend dosimetry, however, there is no information to indicate any change to the previously determined bias and uncertainty associated with the NMP2 fluence methodology. Therefore, the NRC staff agrees with the licensee's statement, in response to SRXB – RAI 2, that "an update of the bias and uncertainty is not required at this time." The additional information provided by the licensee adheres to Regulatory Position 1.4.2.1, "Operating Reactor Measurements," of RG 1.190, which states that, "[a]s capsule and cavity measurements become available, they should be incorporated into the operating reactor measurements database and the calculational biases should be updated, as necessary." The licensee has incorporated new capsule measurements and made an acceptable determination that such an update is not necessary.

Chapter 2 of MPM-402781, Revision 1, states that "[a]n extensive evaluation of all contributors to the uncertainty in the calculated fluence was made for the NMP1 and NMP2 calculations." The licensee also estimates that the analytic uncertainty is about 15 percent, which is within the 20 percent allowance recommended by RG 1.190. The NRC staff notes that the uncertainty analysis does not appear in MPM-402781 or other NMP2 related documentation; however, a similar analytic uncertainty analysis is discussed in detail in BWRVIP-113NP, and the results of the BWRVIP-113NP uncertainty analysis are consistent with those described in MPM-402781.

In addition, the uncertainty analysis contained in BWRVIP-113NP addresses the major sources of uncertainty identified in Regulatory Position 1.4.1, "Analytic Uncertainty Analysis," of RG 1.190. While the maximum vessel fluence uncertainty for River Bend Station is higher, BWRVIP-113NP also notes that the largest contributor to the River Bend fluence uncertainty is the vessel radius uncertainty, a plant-specific value. Since the estimated uncertainty is within the 20 percent allowance recommended by RG 1.190, and since its results are consistent with the more detailed analysis discussed in BWRVIP-113NP, the NRC staff determined that the uncertainty estimate is acceptable.

Based on the following considerations: (1) the previous NRC staff review of the plant-specific qualification established that the NMP2 fluence methodology was acceptably qualified, (2) the licensee reviewed additional, updated benchmarking information pertinent to the NMP2 fluence methodology and determined that an update to the uncertainty and biases was not necessary, and (3) the analytic uncertainty is less than the 20 percent allowance discussed in RG 1.190, the NRC staff determined that the NMP2 fluence methodology is acceptably qualified. For its determination regarding the acceptability of the methodology qualification, the NRC staff determination is based, in part, on information contained in BWRVIP-113NP.

(6) Plant-Specific Application of Conservative Fluence Values

The licensee's November 21, 2012, license amendment request letter stated the following:

> The NRC safety evaluation dated April 27, 2009, included a limitation and condition requiring the licensee to identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology. NMP2

> maintains an NRC approved RG 1.190 fluence monitoring program... and reviews actual fluence on a routine basis. The fluence projections have been confirmed to be conservative using the NMP2 fluence methods...

This passage refers both to MPM-402781 and to the NRC staff safety evaluation approving its use for NMP1. The draft PTLR provided as Attachment 3 to the licensee's November 21, 2012, request letter contains similar information. As discussed under "Review of Fluence Methods," the methods are also acceptable for use at NMP2, in part because they had been previously approved at NMP2, and because the licensee provided updated information showing that their use remains acceptable for the present license amendment request.

Despite the apparent acceptability of the fluence methods, the NRC staff was unable to determine how the statement provided by the licensee addressed the plant-specific action item required for application of the GEH methodology. In the previously discussed precedential applications, each requesting licensee provided a specific statement referring directly to the method used to perform the fluence calculation. An example appears in Reference 33, which includes the following statement:

> ...the PTLR incorporates a fluence calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, which has been approved by the NRC... and is in compliance with Regulatory Guide 1.190.

The NRC staff did not identify any approved precedents that referred to maintenance of an NRC-approved fluence monitoring program, as the NMP2 request letter did. To evaluate this apparently unprecedented approach, the NRC staff requested that the licensee provide more specific information concerning how the fluence values, on which the P-T limits were based, were calculated.

The licensee's response to SRXB – RAI 1 (Ref. 3) indicated that the methods reviewed above were not actually used as input to the P-T limits:

> ...For EPU, the EPU equilibrium core neutron transport calculation were performed and compared to the Cycle 10 (pre-uprate) core using two separate NRC-approved methods: (1) the GE-Hitachi methods described in NEDC-32983P-A, January 2006; and (2) the MPM Technologies methods described in MPM-402781, Revision 1... the P/T limit curves were developed based on the more conservative GEH fluence rate."

The response also stated that, "NMPNS considers the MPM neutron transport calculation methods to be the more accurate representation for NMP2..." In consideration of both this RAI response, and of the information submitted in the LAR, the NRC staff was unable to determine how, exactly, the transport calculations used, as input to the PTLR, had been performed. In the original letter, the licensee's language suggested that the MPM methods had been used; however, in the RAI response, the licensee indicated that GE-Hitachi methods had been used.

In a subsequent RAI response dated November 4, 2013 (Ref. 5), the licensee provided a response to SRXB – RAI 4, a revised PTLR, and MPM-913991, describing the EPU, 32, and 54 EFPY fluence extrapolations. In the November 4, 2013, RAI response, the licensee appeared to revert to its previous position that PTLR fluence values were based on MPM calculations:

The PTLR and the SRXB – RAI 1 response[3] state the fluence methods remain the NMP2 plant-specific methods described in references 6.2 and 6.3 of the PTLR. References 6.2 and 6.3 of the PTLR refer to the approved MPM method for calculating fluence. The derivation of the P-T limit curves used the NMP2 plant specific RG 1.190 fluence calculation results from the end of cycle 10. The P-T limits were defined using conservative extrapolation methods to account for EPU conditions and to provide margin to account for typical variations in neutron flux related to future variation in core and fuel design. Additional information is provided in the PTLR reference 6.4 (MPM-913991) documenting the NMP2 plant specific RG 1.190 fluence and the current best estimate projections based on these methods.

Although this information indicates that the fluence values in the PTLR were based on MPM calculations, the NRC staff reviewed the results from MPM-913991 and the PTLR, and determined that they did not agree, as shown in Table 2.

Table 2. Comparison of PTLR and MPM-Calculated Fluence Values

Location	32 EFPY Comparison		54 EFPY Comparison	
	MPM (n/cm^2)	PTLR (n/cm^2)	MPM (n/cm^2)	PTLR (n/cm^2)
Lower Shell, Shell 1 to Shell 2 Girth Weld and Lower Axial Shell Welds	7.20E+17[1]	9.36E+17[2]	1.39E+18[1]	1.58E+18[2]
Lower-Intermediate Shell and Axial Welds	7.42E+17[3]	9.60E+17[2]	3 1.38E+18[3]	1.62E+18[3]
N6 Nozzle	2.51E+17[4]	3.16E+17[2]	4.57E+17[4]	5.34E+17[2]
N12 Nozzle	1.68E+17[4]	2.16E+17[2]	3.01E+17[4]	3.65E+17[2]
Notes:	1. From MPM-913991, Table 2-6, Circumferential Weld Between Lower and			
	2. From PTLR, Appendix B.			
	3. From MPM-913991, Table 2-6, Welds BE and BF.			
	4. From MPM-913991, Table 2-9.			

Since the table clearly shows that the values calculated by MPM and the values on which the PT limits are based are different, the NRC staff determined that the licensee did not provide

[3] The passage from licensee response to SRXB – RAI 1 is quoted directly above: "...the P/T limit curves were developed based on the more conservative GEH fluence rate." NRC staff does not agree that the response stated that the PTLR was based on MPM calculations.

information to satisfy the plant-specific action item identified in the SE approving NEDO-33178-A, because the licensee did not provide clear information identifying which NRC-approved method had been used to determine the PTLR fluence values, and the information provided by the licensee also showed that the values were not those determined using MPM methods.

Even so, the NRC staff found, as discussed in the section of this SE entitled "Review of Fluence Methods," that the fluence values calculated using the MPM methods are acceptable for use at NMP2. In addition, Table 2 shows that the PTLR fluence values are higher than the MPM-calculated values. This confirms, as stated by the licensee, that the "P-T limits were defined using conservative extrapolation methods." T he use of the conservative methods is acceptable because a comparison to acceptable values, contained in MPM-913991, confirms their conservatism. The use of the more conservative values introduces additional conservatism in the P-T limits assessment, which the NRC staff determined is acceptable.

The licensee's use of a conservative extrapolation, which does not specify how the extrapolation was performed, and subsequent verification of the conservatism using a RG 1.190-adherent method, is a plant-specific change to the methodology described in the GEH methodology, which the licensee has proposed to reference in its TS. Therefore, the NRC did not agree with the licensee response to SRXB – RAI 4 part 6, which stated, "This approach is fully consistent with the GEH methodology... which states that the input for fluence will be based on an approved RG 1.190 fluence method." The licensee's input is not based on such a method; rather, it is verified using such a method.

The NRC staff has determined that the NMP2 disposition for fluence is different from every BWR precedent, in that it introduces an "NRC-approved RG 1.190 fluence monitoring program and reviews actual fluence on a routine basis," and proposes that, "The fluence projections [in the PTLR] have been confirmed to be conservative..." (Page 2 of Enclosure to LAR). BWR precedential amendments generally state which methodology was used to determine the fluence value on which the ART calculations were based explicitly, without mention of a fluence monitoring program or verification effort.

In its December 13, 2013, supplemental letter (Ref. 6), the licensee addressed the plant-specific nature of its request by proposing a revised citation for TS 5.6.7.b, which will read as follows:

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. NEDC-33178P-A, Revision 1, Revision 1, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," dated June 2009. The licensee will calculate the fluence for determining the adjusted reference temperature using either; (1) values determined using an NRC-approved, RG 1.190-adherent method, or (2) a fluence estimate, which the licensee has verified as conservative, using an NRC-approved, RG 1.190-adherent method.

This TS wording specifies two conditions, one of which must be satisfied, for the licensee's PTLR. The P-T limits contained therein will either be consistent with the GEH methodology, in that a single, NRC-approved, RG 1.190-adherent methodology will be used to determine the fluence values, or the licensee may use any fluence estimate for the PTLR, provided a single, NRC-approved, RG 1.190-adherent methodology is used to establish that the estimate is conservative, as was shown in Table 2.

The NRC staff notes that the licensee's proposed wording for TS 5.6.7.b is consistent with the SE approving the GEH methodology in that it refers to the fluence methodology in a singular sense, and it mentions adherence to NRC regulatory guidance. Although this condition appears to afford the licensee the flexibility to use different NRC-approved fluence methodology than that described in the section entitled "Review of Fluence Methods" of this SE, the NRC staff would not consider a combination of two or more previously approved fluence methods to satisfy these conditions, particularly because the language is singular, and because RG 1.190 currently provides no guidance for qualifying fluence estimates determined using a combination of stand-alone methods.

The NRC staff determined that the revised citation is acceptable, because it will ensure that the fluence projections used in the PTLR are fully consistent with the GEH methodology, or, if not, that an NRC-approved method has been used to confirm that the fluence estimate is conservative.

3.2.4.3 Conclusion – Fluence Methodology

Based on the considerations discussed in the preceding sections, the NRC staff determined that the licensee's fluence methodology is acceptable for use, either as input to P-T limits, or as a verification that the P-T limits are based on a conservative fluence estimate. The licensee's revised proposed citation to TS 5.6.7.b will ensure that fluence values used in future revisions to the PTLR are acceptable, in light of the licensee's departure from the fluence requirements set forth in the GEH methodology and its approving safety evaluation.

3.2.5 Compliance with TSTF-419

Industry/TSTF Standard Technical Specification Change Traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," September 15, 2001 (Ref. 9), as modified by changes proposed in Reference 34 that were approved by the NRC staff in Reference 35, requires that references to NRC-approved topical reports for the PTLR methodology must include the revision number and date of the topical report. These changes have been incorporated into NUREG-1433, Rev. 4, Vol. 1 "Standard Technical Specifications, General Electric BWR/4 Plants," Section 5.6.4 (Ref. 13). The proposed markup of TS 5.6.7b in the licensee's December 13, 2013 letter references Revision 1 of the GEH Methodology dated June, 2009. Therefore, the requirements of TSTF-419 as modified by Reference 34 and 35 are met. Therefore, if the licensee wishes to revise the PTLR to use a different revision of the topical report, it must submit a license amendment request.

3.2.6 CONCLUSIONS – TECHNICAL EVALUATION

The proposed NMP2 PTLR meets the seven criteria of GL 96-03, thus is approved for implementation as part of the NMP2 licensing basis. As long as the P-T limit methodology remains the same, this implementation of a PTLR allows the licensee to revise P-T limits for any licensed period of operation under the 10 CFR 50.59 process. Based on its review of the licensee's proposed P-T limits for 32 EFPY, the staff concludes that the proposed P-T limits meet the requirements of 10 CFR 50, Appendix G, and are therefore acceptable, and that the licensee appropriately applied the GEH methodology.

In addition, the NRC staff determined that the licensee's fluence methodology is acceptable for use, either as input to P-T limits, or as a verification that the P-T limits are based on a conservative fluence estimate. The licensee's proposed revision of TS 5.6.7.b will ensure that fluence values used in future revisions to the PTLR are acceptable, in light of the licensee's departure from the fluence requirements set forth in the GEH methodology and its approving safety evaluation.

The proposed P-T limits are acceptable up to a maximum RPV ID fluence of 9.60×10^{17} n/cm^2.

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 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 (March 12, 2013, 78 FR 15749). 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 NRC 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) 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.

7.0 REFERENCES

1. Nine Mile Point, Unit 2, License Amendment Request Pursuant to 10 CFR 50.90: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report, November 21, 2012, Letter, ADAMS Accession No. ML123380336, Package, ADAMS Accession No. ML123380348.

2. Nine Mile Point, Unit 2, Supplement to Nine Mile Point Nuclear Station License Amendment Request to Relocate the Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report, March 25, 2014, Letter, ADAMS Accession No. ML13091A0038.

3. Nine Mile Point, Unit 2, License Amendment Request to Relocate the Pressure & Temperature Limit Curves to the Pressure & Temperature Limits Report - Supplemental Information in Response to NRC Request for Additional Information, July 31, 2013, Letter, ADAMS Accession No. ML13214A388.

4. Nine Mile Point, Unit 2, Amendment Request to Relocate the Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report - Supplemental Information in Response to NRC Request for Additional Information, September 6, 2013, Letter, ADAMS Accession No. ML13254A155.

5. Nine Mile Point Nuclear Station License Amendment Request to Relocate the Pressure & Temperature Limit Curves to the Pressure and Temperature Limits Report - Supplemental Information in Response to NRC Request for Additional Information (TAC No. MF0345), November 4, 2013, Letter, ADAMS Accession No. ML13311A053.

6. Nine Mile Point, Unit 2, License Amendment Request to Relocate the Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report - Supplemental Information in Response to NRC Request for Additional Information and Clarification of RAI EVIB-6, Letter, December 13, 2013, ADAMS Accession No. ML13353A305.

7. Nine Mile Point, Unit 2, License Amendment Request to Relocate the Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report – Supplement to Revise Implementation Date, Letter, February 25, 2014, ADAMS Accession No. ML14063A474.

8. U.S. Nuclear Regulatory Commission, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," Generic Letter 96-03, January 31, 1996, ADAMS Accession No. ML031110004.

9. Industry/TSTF Standard Technical Specification Change Traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," September 15, 2001, ADAMS Accession No. ML012690234.

10. NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Enclosure 1 to MFN 09-506, June 30, 2009, ADAMS Accession No. ML092370488.

11. Submittal of GE BWROG Topical Report NEDC-33178P-A, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves", July 29, 2009, ADAMS Accession No. ML092370486.

12. NEDO-33178-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Enclosure 2 to MFN 09-506, June 30, 2009, ADAMS Accession No. ML092370487.

13. NUREG-1433 Vol. 1, Rev. 4, "Standard Technical Specifications: General Electric BWR/4 Plants- Specifications, Revision 4.0, Volume 1, Specifications," April 30, 2012, ADAMS Accession No. ML12104A192.

14. Manahan, M. P,, "Benchmarking of Nine Mile Point Unit 1 and 2 Neutron Transport Calculations," MPM-402781, September 2003, ADAMS Accession No. ML032681023.

15. U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," RG 1.190, March 31, 2001, ADAMS Accession No. ML010890301.

16. Report No. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 30, 2007, ADAMS Accession No. ML072340283.

17. BWRVIP-114NP-A, "BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual," Final Report, June 30, 2009, ADAMS Accession No. ML092650376.

18. BWRVIP-115NP-A: BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems, Final Report, December 31, 2009, ADAMS Accession No. ML100540367.

19. BWRVIP-117NP-A: BWR Vessel and Internals Project - RAMA Fluence Methodology Plant Application-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5, Final Report, December 31, 2009, ADAMS Accession No. ML100480185.

20. BWRVIP-121NP-A: BWR Vessel and Internals Project, RAMA Fluence Methodology Procedures Manual, 1019052NP, June 30, 2009, ADAMS Accession No. ML100110356.

21. NEDO-32983-A, Rev. 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," January 31, 2006, ADAMS Accession No. ML072480121.

22. Oyster Creek Nuclear Generating Station - Issuance of Amendment 269 Re: Relocation of Pressure and Temperature Curves to the Pressure and Temperature

Limits Report (TAC MD8253), September 30, 2008, ADAMS Accession No. ML082390685.

23. James A. FitzPatrick, Issuance of Amendment Re: Relocation of Pressure and Temperature Curves to the Pressure And Temperature Limits Report Consistent With TSTF-419-A, October 3, 2008, ADAMS Accession No. ML082630365.

24. Grand Gulf Nuclear Station - Proprietary Safety Evaluation, Issuance of Amendment No. 191, Extended Power Uprate to Increase the Maximum Reactor Core Power Operating Limit from 3898 to 4408 megawatts thermal (TAC No. ME4679), July 18, 2012, ADAMS Accession No. ML121210003.

25. Monticello Nuclear Generating Plant - Issuance of License Amendment No. 172 to Revise and Relocate Pressure Temperature Curves to a Pressure-Temperature Limits Report (TAC ME7930), February 27, 2013, ADAMS Accession No. ML13025A155.

26. Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report (TAC Nos. ME8535 and ME8536), April 1, 2013, ADAMS Accession No. ML13079A219.

27. Nine Mile Point, Unit 1, Issuance of Amendment re: Relocation of Pressure and Temperature Curves to the Pressure Temperature Limits Report, January 21, 2010, ADAMS Accession No. ML093370002.

28. Nine Mile Point, Unit No. 1, License Amendment, Issuance of Amendment Re. Pressure Temperature Limit Curves (TAC No. MB6687), October 27, 2003, ADAMS Accession No. ML032760696.

29. Nine Mile, Unit No. 2, License Amendment, Issuance of Amendment Re: Pressure-Temperature Limit Curves, January 27, 2004, ADAMS Accession No. ML040220584.

30. Nine Mile Point Unit 1, Response to Request for Additional Information Re: Pressure-Temperature Limit Curves, July 21,2003, ADAMS Accession No. ML032250595.

31. Fluence Extrapolation in Support of NMP2 P-T Cure Update, MPM-913991, September 30, 2013, Attachment 3, September 30, 2013, ADAMS Accession No. ML13311A055.

32. EPRI - Submittal of "BWRVIP-113NP: BWR Vessel and Internals Project, River Bend 183,. Degree Surveillance Capsule Report," September 13, 2010, ADAMS Accession No. ML102580248.

33. Peach Bottom Atomic Power Station, Units 2 and 3, License Amendment Request - Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report, April 27, 2012, ADAMS Accession No. ML121230354.

34. Letter from Thomas B. Blount to TSTF; Subject: "Implementation Of Travelers TSTF-363, Revision 0, "Revise Topical Report References In Its 5.6.5, COLR [Core

Operating Limits Report]," TSTF-408, Revision 1, "Relocation Of LTOP [Low-Temperature Overpressure Protection] Enable Temperature And PORV [Power-Operated Relief Valve] Lift Setting To The PTLR [Pressure-Temperature Limits Report]," And TSTF-419, Revision 0, "Revise PTLR Definition And References In ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR," dated March 7, 2011, ADAMS Accession No. ML110660285.

35. Letter from William D. Beckner to Anthony R. Pietrangelo regarding disposition made on three travelers containing proposed changes to the Standard Technical Specification (STS) NUREGs, March 21, 2002, ADAMS Accession No. ML020800488.

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