**This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 4.**

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CENG
a joint venture of Constellation Energy · eDF

NINE MILE POINT
NUCLEAR STATION

July 31, 2013

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**ATTENTION:** Document Control Desk

**SUBJECT:** Nine Mile Point Nuclear Station
Unit No. 2, Docket No. 50-410

Nine Mile Point Nuclear Station 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 (TAC No. MF0345)

**REFERENCE:** (a) Letter from K. Langdon (NMPNS) to Document Control Desk (NRC), dated November 21, 2012, License Amendment Request Pursuant to 10 CFR 50.90: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits supplemental information requested by the NRC in support of a previously submitted request for amendment to Nine Mile Point Unit 2 (NMP2) Renewed Facility Operating License NPF-69. The initial request, dated November 21, 2012 (Reference a), proposed to modify 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). Other associated TS changes would also be incorporated. The supplemental information, provided in Attachment 1, Attachment 2 (non-proprietary), and Attachment 4 (proprietary) to this letter, responds to the request for additional information that was provided in an email from the NRC to NMPNS on June 20, 2013.



**This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 4.**

A001
NRR

Attachment 4 is considered to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GE-Hitachi Nuclear Energy Americas LLC (GEH), NMPNS hereby makes application to withhold this attachment from public disclosure in accordance with 10 CFR 2.390(b)(1). The affidavit from GEH detailing the reasons for the request to withhold the proprietary information is provided in Attachment 3.

This supplemental information does not affect the No Significant Hazards Determination analysis provided by NMPNS in Reference (a). Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this supplemental information, without the proprietary Attachment 4, to the appropriate state representative. This letter contains no new regulatory commitments.

Should you have any questions regarding the information in this submittal, please contact John J. Dosa, Director Licensing, at (315) 349-5219.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 31, 2013.

Very truly yours,

Christopher R. Costanzo

CRC/KJK

Attachments:

1. Response to NRC Request for Additional Information, SRXB – RAI 1, SRXB – RAI 2, and SRXB – RAI 3
2. Response to NRC Request for Additional Information, EVIB – RAI 1 and EVIB – RAI 2 (Non-Proprietary)
3. Affidavit from GE-Hitachi Nuclear Energy Americas LLC (GEH) Justifying Withholding Proprietary Information Contained in Attachment 4
4. Response to NRC Request for Additional Information, EVIB – RAI 1 and EVIB – RAI 2 (Proprietary)

cc: Regional Administrator, NRC
Project Manager, NRC
Resident Inspector, NRC
A. L. Peterson, NYSERDA (without Attachment 4)

**ATTACHMENT 1**

# RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

## SRXB – RAI 1, SRXB – RAI 2, and SRXB – RAI 3

By letter dated November 21, 2012, Nine Mile Point Nuclear Station, LLC (NMPNS) proposed to modify Nine Mile Point Unit 2 (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). Other associated TS changes would also be incorporated. This attachment provides supplemental information in response to the NRC request for additional information that was provided in an email from the NRC to NMPNS on June 20, 2013; specifically, SRXB – RAI 1, SRXB – RAI 2, and SRXB – RAI 3. Each individual NRC question is repeated (in italics), followed by the NMPNS response.

## ***SRXB – RAI 1***

*The fluence calculations discussed in MPM-402781, "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations" were completed prior to the implementation of an extended power uprate (EPU) for NMP Unit 2. Describe how the calculations account for the effects of the power uprate.*

### **Response**

The NMPNS Regulatory Guide (RG) 1.190 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). For EPU, the EPU equilibrium core neutron transport calculations were performed and compared to the Cycle 10 (pre-uprate) core using two separate NRC-approved methods: (1) the GE-Hitachi Nuclear Energy (GEH) methods described in NEDC-32983P-A, January 2006; and (2) the MPM Technologies methods described in MPM-402781, Revision 1. The results of the calculations using these two methods agreed within the RG 1.190 tolerance of ±20% on the relative effects of EPU, with the MPM methods predicting a slightly larger percentage change due to EPU. The GEH methods predicted a higher absolute fluence rate for EPU and for pre-uprate operation, as compared to the MPM-402781 methods; however the GEH methods also includes a +7% bias and higher uncertainty which, when considered, explains why the GEH neutron transport calculation is more conservative. The P/T limit curves were developed based on the more conservative GEH fluence rate. NMPNS considers the MPM neutron transport calculation methods to be the more accurate representation for NMP2 based on the plant specific benchmark, and these methods continue to be used for Integrated Surveillance Program (ISP) program compliance.

## ***SRXB – RAI 2***

*The NRC staff understands that the calculational framework devised by MPM Technologies Inc., has been applied at other plants since its plant-specific approval at Nine Mile Point Nuclear Station. The final sentence of RG 1.190, Regulatory Position 1.4.2.1, "Operating Reactor Measurements," states that, "[a]s capsule and cavity measurements become available, they should be incorporated into the operating reactor measurements database and the calculational biases and uncertainties should be updated, as necessary." Provide information to update the operating reactor measurements database based on the more recently available operating reactor measurements, and confirm that the existing calculational bias and uncertainties established for this application of the MPM analytic methods remain qualified for use at NMPNS.*

**Response**

The MPM-402781 neutron transport calculation methods have been applied for evaluation of the River Bend 183 degree capsule (reference Boiling Water Reactor Vessel and Internals Project (BWRVIP) report BWRVIP-113NP, August 2010). 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 BWRVIP ISP maintains a database of all calculated-to-measured values to ensure the criteria of RG 1.190 are satisfied. The BWRVIP provided data through 2008 as part of the Radiation Analysis Modeling Application (RAMA) benchmark report under BWRVIP-189, July 2008. The NMPNS ISP program includes review of all the ISP reports and the results of the fluence data to determine if the capsule data indicates that a change in the bias and uncertainty is warranted. The BWRVIP and NMPNS review of the ISP capsule data indicates that the RAMA bias and uncertainty remain qualified. NMPNS considers this data applicable and concludes that an update of the bias and uncertainty is not required at this time.

***SRXB – RAI 3***

*Explain whether reactor vessel neutron fluence was calculated using any methods other than the referenced MPM Technologies, Inc. analytic methods described in MPM-402781 Revision 1.*

**Response**

As discussed in the response to SRXB – RAI 1 above, the NMP2 reactor vessel fluence has been calculated using GEH neutron transport calculation methods described in NEDC-32983P-A, January 2006. The GEH methods predict a higher absolute fluence rate for EPU as compared to the MPM Technologies neutron transport calculation methods described in MPM-402781, Revision 1.

# ATTACHMENT 2

# RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

## EVIB – RAI 1 and EVIB – RAI 2

## (Non-Proprietary)

This attachment provides supplemental information in response to the NRC request for additional information that was provided in an email from the NRC to NMPNS on June 20, 2013; specifically, EVIB – RAI 1 and EVIB – RAI 2. Each individual NRC question is repeated (in italics), followed by the NMPNS response.

(13 pages attached)

ENCLOSURE 2

GE-PPO-1GYEF-KG1-705

Responses to EVIB RAIs 1 and 2

Non-proprietary Information- Class I (Public)

**NON-PROPRIETARY NOTICE**

This is a non-proprietary version of the Enclosure 1 of GE-PPO-1GYEF-KG1-705 which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

**EVIB - RAI 1**

*BACKGROUND*

*10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," states, "this appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety...". In addition, 10 CFR Part 50, Appendix G, Paragraph IV.A states that, "the pressure-retaining components of the RCPB that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), supplemented by the additional requirements set forth in [paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]...". Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the entire RCPB, consisting of ferritic RCPB materials in the reactor pressure vessel (RPV) beltline (neutron fluence $\geq 1 \times 10^{17}$ n/cm$^2$, $E > 1$ MeV), as well as ferritic RCPB materials not in the RPV beltline (neutron fluence $< 1 \times 10^{17}$ n/cm$^2$, $E > 1$ MeV).*

*ISSUE*

*P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials, may define curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:*

1. *Some ferritic RCPB components that are not RPV beltline shell materials, such as nozzles, penetrations, and other discontinuities, are complex geometry components that exhibit significantly higher stresses than those for the RV beltline region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature ($RT_{NDT}$) for these components is not as high as that of RPV beltline materials that have simpler geometries.*

2. *Ferritic RCPB components that are not RPV beltline shell materials may have material properties, in particular initial $RT_{NDT}$ values, which may define more restrictive P-T limits that those for the RPV beltline shell materials.*

*REQUEST*

*The methodology used to develop the proposed P-T limits for NMP2 is based on NRC-approved topical report NEDC-33178P-A, which addresses all relevant discontinuities in the RV. However, the topical report methodology does not address the evaluation of ferritic components or piping other than those making up the RPV. If NMP2 has ferritic components or piping in the RCPB that are not part of the RPV, describe how it was confirmed whether the P-T limit curves, and the methodology used to develop these curves, considered these materials, consistent with the requirements of 10 CFR Part 50, Appendix G.*

**NMP2 Response:**

NMPNS used the methods described in Topical Report NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature

Curves" (Reference 1) to develop the pressure-temperature (P-T) limits for Nine Mile Point Unit 2 (NMP2). The Reference 1 report describes methods for compliance with the requirements of 10 CFR 50 Appendix G (Reference 2), ASME Code Section XI, Appendix G (Reference 3), and Welding Research Council (WRC) Bulletin 175 (Reference 4). The NRC has reviewed the methods described in Reference 1 and approved the report by issuance of a Safety Evaluation Report (SER) dated April 27, 2009 (Reference 5).

The NMP2 P-T limits were generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99 (Reference 6) as defined in (Reference 1). The reference temperature for nil-ductility transition ($RT_{NDT}$) values of the reactor vessel beltline and non-beltline plates, welds, forgings, and bolting were determined using the NRC-approved GE/BWROG methodology defined in NEDC-32399-P (Reference 7).

Reference 1 is applicable to much of the BWR fleet. Tables 4-4 and 4-5 of Reference 1 provide a list of all non-beltline vessel RCPB components included in the P-T limit evaluation. In accordance with the definition of materials exposed to "sufficient neutron radiation damage" (Reference 2), the topical report identifies all materials in the vessel beltline region that experience end of license fluence greater than or equal to 1.0e17 n/cm$^2$. This typically extends the beltline region considered to some distance both above and below active fuel. Appendix E of Reference 1 demonstrates the method used to identify these materials, and the Adjusted Reference Temperature (ART) table in Section 4 of Reference 1 includes the evaluation of all materials exposed to fluence greater than or equal to 1.0e17 n/cm$^2$.

The GEH topical report (Reference 1) describes the methodology employed to develop bounding P-T limit curves for three (3) regions of the vessel, shown below. Two (2) P-T curves were developed to represent all vessel non-beltline discontinuities; beltline discontinuities are evaluated in accordance with ASME Code requirements.

- The non-beltline upper vessel curve is based upon the bounding component, the [[ ]], and its associated transients. The bounding transient for the upper vessel is defined in Figure 4-3 of Reference 1. The [[ ]] in the upper vessel curve (see Table 4-4 of Reference 1). The upper vessel curve is [[ ]] of Reference 1; therefore, it is assured that the upper vessel curve bounds the requirements for each of these components.
- The non-beltline bottom head curve is based upon the Control Rod Drive (CRD) penetrations. All components identified in Table 4-5 of Reference 1 are included in the evaluation. The CRD penetration curve is [[ ]] of Reference 1; therefore, it is assured that the bottom head curve bounds the requirements for each of these components. [[ ]], the bounding transients applied for the bottom head curve are [[ ]] conditions as defined in Figure 4-2 of Reference 1.
- The beltline curve considers all materials adjacent to the reactor core plus all materials

above and below this region that will be exposed to fluence greater than or equal to 1.0e17 n/cm$^2$ at end of license.

Each non-beltline discontinuity is considered to be bounded by either or both the upper vessel and bottom head curves. T-$RT_{NDT}$ was determined for each discontinuity using Finite Element Analysis (FEA) methods, conservatively based upon [[ ]] is accepted by the ASME Code. Specific examples showing the method of assuring that the most limiting discontinuity is considered in the development of each curve are provided in Section 4.3.2 of Reference 1. The [[ ]] and CRD penetrations, due to the consideration of loading, transients, and the dimensions of the vessel and nozzles, are more limiting relative to stress than any of the Class 1 ferritic branch piping in the RCPB. It is further noted that most of the RCPB piping is fabricated from non-ferritic materials.

The GEH methodology for P-T curve development (Reference 1) contains a number of [[ ]] considerations. Detailed stress analyses of BWR/6 non-beltline components were performed specifically for the purpose of fracture toughness. The [[ ]] designs. The analyses considered mechanical loading and anticipated thermal transients that [[ ]], and were applied to all vessel discontinuities. In addition, [[ ]] in this evaluation. The results of these transients experienced by and evaluated for the P-T curve evaluation are more severe for the vessel and nozzles than those anticipated for the attached piping and equipment.

ASME Code Section XI, Appendix G, Article G-3000, paragraph G-3100 states that for materials "used for piping, pumps, and valves for which impact tests are required (NB-2311), the tests and acceptance standards of Section III, Division 1 are considered to be adequate to prevent non-ductile failure under the loadings and with the defect sizes encountered under normal, upset, and testing conditions. Level C and Level D Service Limits should be evaluated on an individual case basis (G-2300)." As described in Section 4.3 of Reference 1, [[ ]] in the development of all non-beltline P-T limits.

A Pressure Integrity Specification and a Plant Requirements Specification were used during design and fabrication to ensure that all NMP2 RCPB components would maintain the required capabilities during plant life. Loading and fatigue limits were determined for all vessel attachments, such as nozzles. The pressure integrity specification requires that piping connecting to the vessel does not exceed these loads for all service levels and operating conditions. All materials within the RCPB that were not supplied by GE were required to demonstrate compliance with applicable codes. The specification states that the requirements identified are applicable to the NSSS components such as pressure vessel, heat exchangers, tanks, piping, pumps and valves, and equipment pressure parts such as fittings, flanges, bolts, valve bodies, pump casings, and piping system parts that constitute pressure retaining parts for the process fluid. This specification identifies that these requirements are applicable to Residual Heat Removal, High Pressure Core Spray, Low Pressure Core Spray, and Reactor Core Isolation Cooling systems. It is further noted that there have been no replacements of the ferritic piping or components in the NMP2 RCPB.

This pressure integrity specification also specifies fracture toughness requirements for ferritic materials. For ferritic RCPB materials ordered where the service temperature is less than 250°F when the system is pressurized to more than 20% of the design pressure, impact tests in accordance with the ASME Code were required to demonstrate adequate fracture toughness properties. For RCPB materials having a minimum service temperature of 250°F or more when the system is pressurized to more than 20% of the design pressure, impact testing was not required. Further, impact testing was not required on components or equipment whose rupture could not result in a loss of coolant exceeding the capability of the normal makeup system to maintain adequate core cooling for the duration of reactor shutdown and orderly cooldown.

The General Design Criteria (GDC) in effect at the time that NMP2 was fabricated are discussed in Section 3.1 of the NMP2 USAR (Reference 8). NMP2 complies with the GDC for Nuclear Power Plants, 10 CFR 50, Appendix A except as noted in USAR Section 3.1. The NRC requirement regarding the GDC is that the plant application "provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety". Criteria 14, 30, 31, and 32 specify requirements with respect to the RCPB.

Criterion 14, "Reactor Coolant Pressure Boundary," states:

"The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

Criterion 14 is satisfied as follows:

"The piping and equipment pressure parts within the RCPB extending to and including the outer containment isolation valves are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 [of Reference 8] classifies systems and components within the RCPB as Quality Group A. The design requirements, codes, and standards applied to this quality group ensure a quality product in keeping with the safety functions to be performed.

To minimize the possibility of brittle fracture within the RCPB, fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Criterion 31 describes the methods used to control toughness properties. Materials are impact tested in accordance with applicable portions of the ASME Boiler and Pressure Vessel Code, Section III. Where RCPB piping penetrates the containment, fracture toughness temperature requirements of the RCPB materials apply.

Piping and pressure-retaining portions of components that compose the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. All welding procedures, welders, and welding machine operators used in producing pressure-retaining welds are qualified in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section IX, for the materials to be welded. Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Section 5.2.3 [of Reference 8] contains the detailed material and examination requirements for the piping and components of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance are accomplished as described in Criterion 30. The design, fabrication, erection, and testing of the RCPB assure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14."

Criterion 30, "Quality of Reactor Coolant Pressure Boundary," states:

"Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

Criterion 30 is satisfied as follows:

"By utilizing conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Components that compose the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Sections 5.2.2.6 and 5.3.1.1 and Table 3.2-1 [of Reference 8]. Further, product and process quality planning is provided to ensure conformance with applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is provided in Section 3.1.2.14 [of Reference 8].

Means are provided for detecting reactor coolant leakage. The leak detection system (LDS) consists of sensors and instruments to detect, annunciate, and in some cases, isolate the RCPB from potentially-hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, by increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, increases in drywell pressure or temperature, and changes in reactor water level. Allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, ability to make up coolant system leakage, normally-expected background leakage due to equipment design, and detection capability of various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal AC power with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the LDS provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the LDS are designed to meet the requirements of Criterion 30."

Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," states:

"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."

Criterion 31 is satisfied as follows:

"Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel (RPV), the RPV is designed to meet the requirements of 10 CFR 50. The nil ductility transition temperature (NDTT) is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDTT increases as a function of neutron exposure at integrated neutron exposures greater than 1 x $10^{17}$ nvt with neutrons of energy in excess of 1 MeV.

The balance of the RCPB is designed, maintained, and tested in such a way that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Section 5.2 [of Reference 8] discusses this in further detail. Therefore, the RCPB is in conformance with Criterion 31."

Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," states:

"Components which are part of the RCPB shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel (RPV)."

Criterion 32 is satisfied as follows:

"The RPV design and engineering effort includes provisions for in-service inspection (ISI). Removable plugs in the biological shield wall (BSW) and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system SRVs, recirculation system, and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the RCPB is in accordance with the ASME Boiler and Pressure Vessel Code Section XI. Section 5.2 [of Reference 8] defines the ISI plan, access provisions, and areas of restricted access.

Vessel material surveillance samples are located within the RPV. The program includes specimens of the base metal, weld metal, and heat-affected zone (HAZ) metal.

The plant testing and inspection programs ensure that the requirements of Criterion 32 will be met."

It is noted that NMP2 is participating in the Integrated Surveillance Program (ISP). The NMP2 surveillance capsules are not included in the ISP; however, one capsule was removed and tested before the ISP was implemented. The remaining capsules are still in place in the vessel, and will be available for testing should the need arise.

**EVIB - RAI 2**

*The proposed PTLR states on page 5 that for NMP2, the plate heat C3147-1 is the limiting material for the beltline region, that peak fluence values used in the development of the Pressure-Temperature curves are identified in Appendix B, and that the limiting adjusted reference temperature (ART) for the beltline LPCI N6 and Water Level Instrumentation N12 nozzle forgings and welds are also considered in the development of the beltline PT curves. The adjusted reference temperature (ART) for the N12 forging is 39 °F, compared to the limiting ART for the beltline region, which is 51 °F for beltline plate C3147-1.*

1) *Appendix J to NEDC-33178P-A describes the P-T limit methodology for the N12 nozzle and provides a method for determining the nozzle-specific P-T curve if the instrument nozzle is found to exist in the beltline region of the RPV. Provide a summary of the calculation including the plant-specific inputs that would be needed by the staff to perform a confirmatory calculation using the approved methodology, and the resulting N12 nozzle P-T curve for NMP2.*

2) *Section 4.3.2.2 of NEDC-33178P-A indicates that in the event that a full penetration nozzle is the limiting material for the beltline region, the pressure test and the beltline core not critical P-T curves are calculated in the same manner as the Feedwater Nozzle pressure test and core not critical P-T curves as described in Section 4.3.2.1. Provide a summary of the calculation including the plant-specific inputs that would be needed by the staff to perform a confirmatory calculation using the approved methodology, and the resulting N6 nozzle P-T curve for NMP2.*

**NMP2 Response**

Item 1. Water Level Instrumentation (WLI) Nozzle:

The GEH methodology is that any partial penetration nozzle that is located in the beltline region is evaluated using the [[ ]] methodology as defined in Appendix F of Reference 1. The [[ ]], and therefore uses the initial $RT_{NDT}$ of the forging; for a beltline nozzle, the adjusted reference temperature (ART) is used.

Pressure Test (Curve A)

Detailed calculations are provided below for the core not critical condition. The main differences between calculations for $K_I$ for the pressure test are that the safety factor is 1.5 and that [[
]] as defined in Reference 1.

Core Not Critical (Curve B)

The $K_I$ for pressure and thermal for the [[ ]] WLI nozzle used in the evaluation are defined in Appendix J of Reference 1:

[[

]]

[[

]]

The total $K_I$ is substituted into the following equation in order to solve for T-$RT_{NDT}$.

[[

]]

Considering the 32 EFPY ART of the WLI nozzle of 39°F and the [[

]]

As can be seen in the PTLR that was submitted by NMPNS letter dated November 21, 2012, at 1050 psig the required temperature for the beltline Curve B is 130°F, resulting from the 10 CFR 50 Appendix G rules, and bounds the WLI nozzle requirements.

Item 2, LPCI Nozzle:

The GEH methodology is that any full penetration nozzle that is located in the beltline region is evaluated using the [[ ]] methodology as defined in Appendix F of Reference 1. The [[ ]], and therefore uses the initial $RT_{NDT}$ of the forging; for a beltline nozzle, the adjusted reference temperature (ART) is used.

Pressure Test (Curve A)

Detailed calculations are provided below for the core not critical condition. The main differences between calculations for $K_I$ for the pressure test are that the safety factor is 1.5 and that [[
]] as defined in Reference 1.

Core Not Critical (Curve B)

The following calculations are performed for a pressure of 1050 psig and a temperature of 551°F.

The stresses for the [[ ]] LPCI nozzle used in the evaluation are:

[[

]]

These stresses are used to evaluate the LPCI nozzle; [[ ]] The [[ ]] (Reference 1). For NMP2, the [[ ]]. Therefore, the NMP2 LPCI nozzle primary membrane stress is:

]]

As this value is [[ ]] and has been used for the NMP2 LPCI nozzle evaluation.

The stresses above, [[ ]]. For this calculation, [[ ]]. This [[ ]]. The [[ ]] in WRC Bulletin 175 (Reference 4). The revised stresses are:

[[

]]

The value of $M_m$ is determined as defined in Reference 1.

$M_m$ = 1.85 for $\sqrt{t} \leq 2$

$M_m$ = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

$M_m$ = 3.21 for $\sqrt{t} > 3.464$

For NMP2, [[ ]]. Therefore,

$M_m$ = [[ ]]

The value of $F(a/r_n)$ is obtained from WRC 175 (Reference 4), Figure A5-1 for the NMP2 dimensions.

$a$ = $\frac{1}{4}(t_n^2 + t_v^2)^{1/2}$ = [[ ]]

$t_n$ = [[ ]]

$t_v$ = [[ ]]

$r_i$ = [[ ]]

$r_c$ = [[ ]]

$r_n$ = $r_i + 0.29\, r_c$ = [[ ]]

Therefore, $a/r_n$ = [[ ]], and $F(a/r_n)$ = [[ ]], conservatively.

$K_{IP}$ = $SF * \sigma (\pi a)^{1/2} * F(a/r_n)$
= [[ ]]
= [[ ]]

$K_{IS}$ = $M_m * (\sigma_{sm} + (2/3) * \sigma_{sb})$
= [[ ]]
= [[ ]]

The total $K_I$ is therefore:

$K_I$ = [[ ]]

The total $K_I$ is substituted into the following equation in order to solve for $T\text{-}RT_{NDT}$.

$(T\text{-}RT_{NDT})$ = $\ln [(K_I - 33.2) / 20.734] / 0.02$
= [[ ]]

Considering the 32 EFPY ART of the LPCI nozzle of -4°F and the [[ ]]:

T = [[ ]]

As can be seen in the PTLR that was submitted by NMPNS letter dated November 21, 2012, at 1050 psig, the required temperature for the beltline Curve B is 130°F, resulting from the 10CFR50 Appendix G rules, and bounds the LPCI nozzle requirements.

**References:**

1. GEH Nuclear Energy, NEDC-33178P-A, Revision 1, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves", Report for BWR Owners' Group, Sunol, California, June 2009 (GEH Proprietary).
2. 10CFR50 Appendix G, "Fracture Toughness Requirements", December 1995.
3. ASME Code Section XI Appendix G, "Fracture Toughness Criteria for Protection Against Failure", 2004 Edition.
4. WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials", August 1972.
5. Safety Evaluation Report, Thomas B. Blount (NRC) to Doug Coleman (BWROG), "Final Safety Evaluation for Boiling Water Reactors Owners' Group Licensing Topical Report NEDC-33178P, General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves (TAC No. MD2693)", April 27, 2009.
6. "Radiation Embrittlement of Reactor Vessel Materials", USNRC Regulatory Guide 1.99, Revision 2, May 1988.
7. GE Nuclear Energy, NEDC-32399-P, "Basis for GE $RT_{NDT}$ Estimation Method", Report for BWR Owners' Group, San Jose, CA, September 1994 (GE Proprietary); Letter from B. Sheron to RA Pinelli, "Safety Assessment of Report NEDC-32399-P, Basis for GE $RT_{NDT}$ Estimation Method, September 1994", USNRC, December 16, 1994.
8. NMP2 Updated Safety Analysis Report (USAR), Docket 50-410, Revision 20, October 2012.

# ATTACHMENT 3

# AFFIDAVIT FROM GE-HITACHI NUCLEAR ENERGY AMERICAS LLC (GEH) JUSTIFYING WITHHOLDING PROPRIETARY INFORMATION CONTAINED IN ATTACHMENT 4

# GE-Hitachi Nuclear Energy Americas LLC

# AFFIDAVIT

**I, Linda C. Dolan,** state as follows:

(1) I am the Manager of Regulatory Compliance, of GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, GE-PPO-1GYEF-KG1-705, "GEH Responses to NMP2 PTLR RAIs," dated July 25, 2013. The GEH proprietary information in Enclosure 1, which is entitled "Responses to EVIB RAIs 1 and 2," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. Sec. 552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my

knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains the detailed GEH methodology for pressure-temperature curve analysis for the GEH Boiling Water Reactor (BWR). These methods, techniques, and data along with their application to the design, modification, and analyses associated with the pressure-temperature curves were achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience databases that constitute a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 25th day of July 2013.

Linda C. Dolan
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