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 Attachments 12, 13, 14, 15 and 16.

Thomas A. Lynch
Plant General Manager

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Constellation Energy
Nine Mile Point Nuclear Station

December 23, 2009

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

Response to Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 2 – Re: License Amendment Request for Extended Power Uprate Operation (TAC No. ME1476)

REFERENCES:

(a) Letter from K. J. Polson (NMPNS) to Document Control Desk (NRC), dated May 27, 2009, License Amendment Request (LAR) Pursuant to 10 CFR 50.90: Extended Power Uprate

(b) Email from R. Guzman (NRC) to J. J. Dosa (NMPNS), dated September 17, 2009, Item (2) Disposition of Condition 9.20 in the PUSAR Appendix A

(c) Letter from R. V. Guzman (NRC) to S. L. Belcher (NMPNS), dated November 13, 2009, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 2 – Re: The License Amendment Request for Extended Power Uprate Operation (TAC No. ME1476)

(d) Letter from R. V. Guzman (NRC) to S. L. Belcher (NMPNS), dated November 18, 2009, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 2 – Re: License Amendment Request for Extended Power Uprate Operation (TAC No. ME1476)

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 Attachments 12, 13, 14, 15 and 16.

A001
NRR

(e) Email from R. Guzman (NRC) to T. H. Darling (NMPNS), dated December 1, 2009, Request for Reference 77: GE Nuclear Energy, "Nine Mile Point-2 SAFER/GESTR Loss-of-Coolant Accident Analysis for GE14 Fuel," GE-NE-0000-0024-6517-R0, March 2004.

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits revised and supplemental information in support of a previously submitted request for amendment to Nine Mile Point Unit 2 (NMP2) Renewed Operating License (OL) NPF-69. The request, dated May 27, 2009 (Reference a), proposed an amendment to increase the power level authorized by OL Section 2.C.(1), Maximum Power Level, from 3467 megawatts-thermal (MWt) to 3988 MWt. By email dated September 17, 2009 (Reference b), letter dated November 13, 2009 (Reference c), letter dated November 18, 2009 (Reference d), and email dated December 1, 2009 (Reference e) the NRC staff determined that additional information was needed to support its review. The information in Attachments 1 and 12 and the other Attachments referenced therein is provided to address the above-referenced RAIs.

Attachments 12, 13, 14, 15 and 16 are considered to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of Continuum Dynamics, Inc. (CDI) and GE Hitachi Nuclear Energy (GEH), NMPNS hereby makes application to withhold these attachments from public disclosure in accordance with 10 CFR 2.390(b)(1). Affidavits executed by CDI and GEH detailing the reasons for the requests to withhold the proprietary information are provided in Attachments 9, 10, and 11. Attachments 1, 3, 4, 5 and 6 provide non-proprietary versions of the reports in Attachments 12, 13, 14, 15 and 16.

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

Very truly yours,

Thyrell

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF OSWEGO :

I, Thomas A. Lynch, being duly sworn, state that I am the Nine Mile Point Plant General Manager, and that I am duly authorized to execute and file this response on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

TLynch

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Onondaga, this 23rd day of December, 2009.

WITNESS my Hand and Notarial Seal:

Dennis E. V
Notary Public

My Commission Expires:

3/17/2012
Date

DENNIS E. VANDEPUTTE
Notary Public, State of New York
No. 01VA6183401
Qualified in Onondaga County
Certificate Filed in Oswego County
Commission Expires 3/17/2012

TAL/JJD

Attachments:

1. Response to Request for Additional Information Regarding License Amendment Request for Extended Power Uprate Operation (Non-Proprietary)
2. Steam Dryer Evaluation Executive Summary - Revised (LAR Attachment 13)
3. CDI Technical Note No. 09-17NP (Non-Proprietary), Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Nine Mile Point Unit 2, Rev. 0
4. CDI Report 08-08NP (Non-Proprietary), Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz, Rev. 3 (LAR Attachment 13.8)
5. CDI Report 09-26NP (Non-Proprietary), Stress Assessment of Nine Mile Point Unit 2 Steam Dryer at CLTP and EPU Conditions, Rev. 1 (LAR Attachment 13.7)
6. GE-NE-0000-0024-6517-R0 (Non-Proprietary), SAFER/GESTER Loss-of-Coolant Accident Analysis for GE14 Fuel
7. Replacement Pages for NEDO-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachment 3)

8. Replacement Pages for NEDO-33351P, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachment 11)
9. Affidavit Justifying Withholding Proprietary Information in GE-Hitachi Nuclear Energy Americas LLC Document GE-NE-0000-0024-6517-R0 SAFER/GESTER Loss-of-Coolant Accident Analysis for GE14 Fuel
10. Affidavit Justifying Withholding Proprietary Information in GE-Hitachi Nuclear Energy Americas LLC Document RAI Responses B2, E1, F1 through F9, and SNPB-1
11. Affidavit Justifying Withholding Proprietary Information in RAI Response A2, CDI Technical Note No. 09-17P, Rev.0, CDI Report 08-08P, Rev. 3, and CDI Report 08-24P, Rev. 2
12. Response to Request for Additional Information Regarding License Amendment Request for Extended Power Uprate Operation (Proprietary)
13. CDI Technical Note No. 09-17P (Proprietary), Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Nine Mile Point Unit 2, Rev. 0 (New LAR Attachment 13.10)
14. CDI Report 08-08P (Proprietary), Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz, Rev. 3 (LAR Attachment 13.2)
15. CDI Report 09-26P (Proprietary), Stress Assessment of Nine Mile Point Unit 2 Steam Dryer at CLTP and EPU Conditions, Rev. 1 (LAR Attachment 13.1)
16. GE-NE-0000-0024-6517-R0 (Proprietary), SAFER/GESTER Loss-of-Coolant Accident Analysis for GE14 Fuel

cc: NRC Regional Administrator, Region I
NRC Resident Inspector
NRC Project Manager
A. L. Peterson, NYSERDA (w/o Attachments 12, 13, 14, 15 and 16)

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE OPERATION (NON-PROPRIETARY)

Certain information, considered proprietary by GEH and CDI, has been deleted from this Attachment. The deletions are identified by double square brackets.

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE OPERATION (NON-PROPRIETARY)

By letter dated May 27, 2009, as supplemented on August 28, 2009, Nine Mile Point Nuclear Station, LLC (NMPNS), submitted for Nuclear Regulatory Commission (NRC) staff review and approval, a proposed license amendment requesting an increase in the maximum steady-state power level from 3467 megawatts thermal (MWt) to 3988 MWt for Nine Mile Point Unit 2 (NMP2). This attachment provides supplemental information in response to the request for additional information (RAI) documented by NRC emails and letters dated September 17, 2009, November 13, 2009, November 18, 2009, and December 1, 2009. The NRC request is repeated (in italics), followed by the NMPNS response.

Mechanical & Civil Engineering – Steam Dryer Evaluation

RAI A1

In the executive summary report (Attachment 13.0) and in the executive summary and references of the CDI Report 09-26P, "Stress Assessment of Nine Mile Point Unit 2 Steam Dryer at CLTP and EPU Conditions," the licensee refers to the BWRVIP-194 Report, "BWR Vessel and Internals Project, Methodologies for Demonstrating Steam Dryer Integrity for Power Uprate." The licensee also states that the steam dryer integrity analysis has followed the guidelines outlined in this report. The licensee is requested to omit references to the BWRVIP-194 topical report, "Methodologies for Demonstrating Steam Dryer Integrity for Power Uprate," in its application, as it has neither been reviewed nor approved by the NRC staff. The licensee is requested to supplement the necessary information as stand-alone information in the EPU application rather than referencing the topical report that is not yet endorsed by the NRC.

In Attachments 13 and 13.1, the licensee references BWRVIP topical reports, BWRVIP-181, "Steam Dryer Repair Design Criteria," and BWRVIP-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," that have not been reviewed nor approved by the staff. These topical reports are currently being reviewed by the staff. Reference to such unapproved documents is not acceptable. The licensee is requested to supplement the necessary information as stand-alone information in the EPU application rather than referencing the topical reports that are not yet endorsed by the NRC.

NMPNS Response RAI A1

To address this question, NMPNS has removed references to BWRVIP-194, "BWR Vessel and Internals Project, Methodologies for Demonstrating Steam Dryer Integrity for Power Uprate," from Attachments 13, 13.1, 13.2, 13.7 and 13.8 of the license amendment request (LAR). The following revisions to the LAR attachments are provided:

- Attachment 2 contains replacement pages for the "Steam Dryer Evaluation" (LAR Attachment 13).
- Attachment 15 replaces CDI Report No. 09-26P, Revision 0, "Stress Assessment of Nine Mile Point Unit 2 Steam Dryer at CLTP and EPU Conditions" (LAR Attachment 13.1).
- Attachment 14 replaces CDI Report No. 08-08P, Revision 2, "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz" (LAR Attachment 13.2).

- Attachment 5 replaces CDI Report No. 09-26NP, Revision 0, "Stress Assessment of Nine Mile Point Unit 2 Steam Dryer at CLTP and EPU Conditions" (LAR Attachment 13.7).
- Attachment 4 replaces CDI Report No. 08-08NP, Revision 2, "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz" (LAR Attachment 13.8).

BWRVIP topical reports BWRVIP-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," (draft Safety Evaluation issued November 23, 2009 - TAC No. MD9427) and BWRVIP-181, "Steam Dryer Repair Design Criteria," (draft Safety Evaluation issued October 22, 2009 - TAC No. MD8325), have been reviewed by the staff. NMPNS believes it is appropriate to reference these two BWRVIP reports and has verified that any changes to the topical reports resulting from the NRC's reviews do not impact the LAR.

RAI A2

Table 4 in SIA Calculation NMP-26Q-302R0 (Attachment 13.4) presents the frequencies of the peaks which were removed from the strain gage power spectra. Some of these peaks are identified to be related to the pump vane pass frequencies, but others are referred to as "non-identified sources." The licensee is requested to include in its submittal, the rationale of filtering these spectral peaks for which the sources have not been identified.

NMPNS Response RAI A2

The "Non-identified source" frequency peaks (from Table 4, reproduced below) are purely electrical in nature and do not contribute to the strain configuration in any level. The "non-identified source" frequency peaks were identified by collecting data without excitation power to the Wheatstone bridge. The frequency content in this data set is used to identify electrical interferences from other sources not specifically related to the vane pass frequencies; thus, these peaks can be removed from the input of the stress analysis.

Table 4: Applied Notch Filters

[[

(3)]]

These peaks are captured by Electrical Interference Check (EIC):

From Section 2.1 of NMP-26Q-302R0:

Additionally, at every power level above NOP/NOT a measurement was repeated without Wheatstone bridge excitation voltage. In this configuration the cables of the strain gages serve as antennae capturing only the electric noise normally present in the signal. The purpose of this measurement is to identify the noise characteristic of the system. This technique is called Electrical Interference Check (EIC).

SIA used the expression "Non-identified source" because it is not known what equipment is generating the Electro-Magnetic Interference (EMI) that is being inducted into the strain gage signal cables.

Example: Justification for removal of the 88.25 Hz peak at the 100% Power level:

There is a dominant peak in the EIC spectra of MSL-D-Upper location (Figures A2.1 and A2.2). The same peak can be observed in the 100% DATA file of MSL-D-Upper (Figures A2.3 and A2.4). Therefore, this peak in the frequency spectra is a result of electrical interferences, but the source of this electrical interference is unknown.

[[

(3)]]

Figure A2.1: 100% Power EIC File 20080419094134.dta MSL-D-Upper

[[

(3)]]

Figure A2.2: 100% Power EIC File 20080419094134.dta MSL-D-Upper (zoomed plot)

[[

(3)]]

Figure A2.3: 100% Power DATA File 20080419094734.dta MSL-D-Upper

[[

(3)]]

Figure A2.4: 100% Power DATA File 20080419094734.dta MSL-D-Upper (zoomed plot)

RAI A3

In CDI Report No. 08-08P, "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point, Unit 2 Steam Dryer to 250 Hz.," the Electrical Interference Check (EIC) signals are filtered out from the CLTP signals to estimate the dryer load by means of the Acoustic Circuit Model (ACM), Rev. 4. However, the EIC signals are not provided in CDI Report No. 08-08P. The licensee is requested to revise this report to include:

(a) *CLTP signals at all locations on the MSLs, before subtracting the EIC signals,*

(b) *The EIC signals at all locations for CLTP condition, and*

(c) *TP signals after subtracting the EIC signals.*

The licensee is requested to provide this information in the form of overlapping power spectral density (PSD) plots for each location, separately, to facilitate comparisons between the various signals.

NMPNS Response RAI A3

See Attachment 14, CDI Report 08-08P (Proprietary), "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz," Rev. 3, which contains the requested plots.

RAI A4

Attachment 13.2 (CDI Report 08-08P Rev.1) provides sample level 1 and level 2 limit curves for MSL-A upper and lower locations. The licensee is requested to:

(a) Describe what is meant by sample limit curves.

(b) Clarify if the sample limit curves are NMP2 specific limit curves.

(c) Provide the limit curves for all four NMP2 MSL lines.

NMPNS Response RAI A4

The "sample limit curves" are, in fact, the limit curves for main steam line A.

See Attachment 13, CDI Technical Note No. 09-17P, "Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Nine Mile Point Unit 2," Rev. 0, which provides the limit curves for all four NMP2 main steam lines.

RAI A5

Appendix-A of CDI Report 09-26P describes the methodology used for submodeling the four locations with the stress reduction factors (SRFs) ranging from 0.62 to 0.88. Item 6 of Table 7 of CDI Report 09-26P lists an SRF of 0.79, but does not describe the submodeling methodology used for the location of the bottom of the hood/hood support weld at junctions of the base plate. The licensee is requested to address and clarify whether the submodeling methodology used for this location is the same as in Appendix-A, or discuss whether a non-standard submodeling approach was used. Also, the licensee is requested to provide a discussion of whether the forces and displacements at the cut boundary of the global shell model were imported to the solid submodel. The NRC staff does not endorse non-standard and unconventional submodeling approaches that use non-unique and arbitrary loading or displacements along with some arbitrary boundary conditions, to establish the applicable SRF value.

NMPNS Response RAI A5

Submodel 6 in Table 7 of CDI Report No. 09-26P corresponds to the common junction between the bottom of a hood, hood stiffener, and base plate. It was originally developed by Structural Integrity Associates (SIA) for Browns Ferry as part of their EPU uprate program. The stress reduction factor of 0.79 is reused for the Nine Mile Point steam dryer because of the identical local geometry and similar loading conditions pertaining to both the Browns Ferry and Nine Mile Point units at this location. The submodeling methodology (referred to hereafter as submodel 1 or SM1) used to analyze this location was extensively validated in Reference A5.1. Its application to the junction of concern here is documented in Reference A5.2.

To justify this reuse, it is important to note that in any submodeling method, the stress reduction factor obtained by the method is independent of the applied load or stress magnitude. Only the local geometry, structural properties, and the manner in which the forces and moments are distributed over the submodel boundaries are significant in establishing the stress reduction factor. This means that if a stress reduction factor is obtained at a given location for an imposed load force, F(x,y,z), then this factor will not change when the force is changed to α F(x,y,z) for some arbitrary scaling factor α. Therefore, even though the stress amplitudes in the Browns Ferry and Nine Mile Point steam dryer units are different, the same submodel can be applied provided that:

(a) The load condition in the submodel accurately reflects the load condition in the global model, and
(b) The submodel accurately reproduces the stress field in the vicinity of the junction.

The load condition giving rise to the maximum alternating stress is dominated by a strong in-plane force in the hood stiffener (note that since the alternating stress is being considered, this force is alternately tensile and compressive). This is established in Figure A5.1, which compares the peak membrane (Pm) and membrane+bending (Pm+Pb) stress intensities in the region of concern. Over the hood stiffener component these contours are seen to be: (i) similar in magnitude and (ii) strongest as one approaches the bottom edge. The implication of (i) is that the bending stresses are small. This is confirmed quantitatively in Figure A5.2, which reports the peak and the alternating stress intensities along the bottom edge at the limiting frequency shift (-5%). The fact that the stress intensities at the bottom, middle, and top surfaces of the bottom hood stiffener edge are virtually identical confirms that bending stresses are negligible and membrane stresses dominate in this member all the way to the junction. From observation (ii), the in-plane stresses are highest along the bottom edge of the hood stiffener. Moreover, since the free edge cannot support shear or normal (i.e., to the edge) stresses, it follows that this stress is a longitudinal stress directed parallel to the free edge. These considerations naturally motivate an applied load condition in the submodel consisting of a point force applied to the cut edge of the hood stiffener as indicated in Figure 4-1 on page 8 of Reference A5.2.

The stress distribution in the hood and base plate involves both membrane and bending components as can be inferred from Figure A5.1 and as would be expected intuitively given the dominant loading. Generally, the stresses in the base plate and hood near the junction act to resist the loads in the stiffener. Because of the hood/stiffener/base plate arrangement, other contributions to the local stress at the junction are expected to be small. To see this, note that stresses in the hood induced by hood vibration modes will be small locally because bending about the horizontal axis is constrained by the hood stiffener and bending about the vertical axis is prevented by the base plate. For the base plate, bending vibrations about the x-axis (i.e., the axis pointing from one MSL pair to the other) are prevented by the hood. Bending about the y-axis (pointing along the long length of the base plate) is not resisted, but the associated modal frequencies are expected to be high because of the relatively small width of the base plates. Finally, for the stiffener itself, it has already been established that the bending response is negligible. Since bending-induced stresses are either absent or suppressed near the junction, it follows that the dominant stress state is similar to that resulting from the application of the point force described above.

Finally, in the response to Browns Ferry RAI EMCB 201/162 where the same benchmark problem examined in Reference A5.1 was re-analyzed using the CDI submodeling procedure described in Appendix A of CDI Report No. 09-26P (hereafter referred to as submodel SM2), it was noted that the stress reduction factor obtained using the SIA submodeling technology (SM1) was more conservative than using SM2.

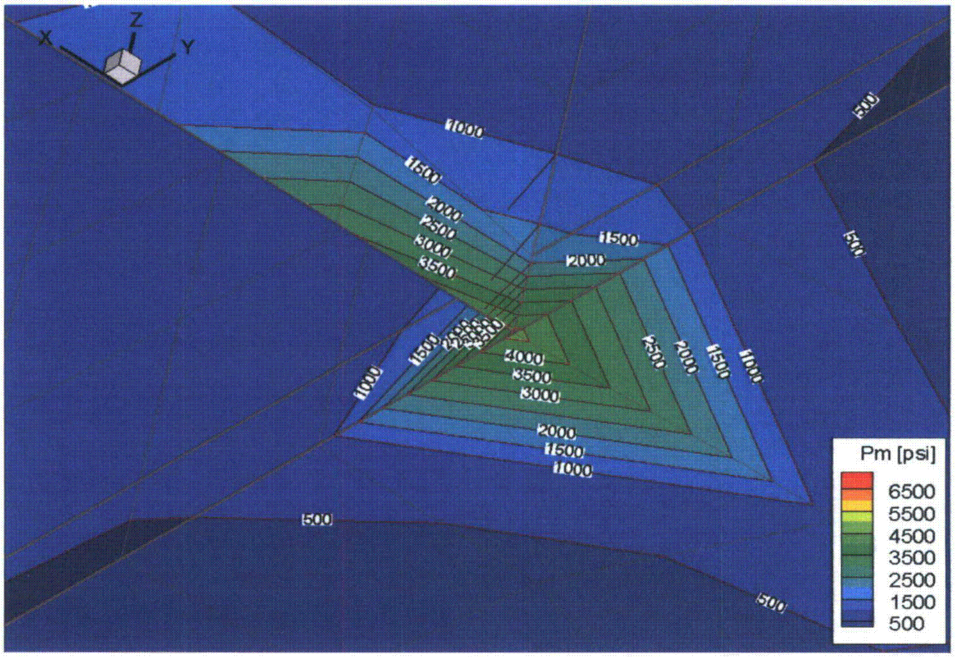
In conclusion, while the model SM1 lacks the formal uniqueness and general applicability of SM2, in this particular instance the approach is adequate because the geometry and known stress distribution allow the applied load to be reliably inferred. Since vibration-induced bending stresses are locally suppressed, the loading is dominated by a strong in-plane membrane stress in the stiffener whose orientation is known on physical grounds (i.e., the boundary conditions at a free edge). In addition, the configuration is conceptually similar to one studied extensively in References A5.1 and A5.3 (in-plane axial force applied to a cantilevered beam) using both the SM1 and SM2 submodels as well as the sub-structuring technology contained in the ANSYS stress analysis. There it was shown (Reference A5.3) that SM1 was more conservative than SM2 with regard to estimating the stress reduction factor.

References

A5.1 Structural Integrity Associates, Inc. (2008), *Comparison Study of Substructure and Submodel Analysis using ANSYS*, Calculation Package, 0006982.304.

A5.2 Structural Integrity Associates, Inc. (2008), *Shell and Solid Sub-Model Finite Element Stress Comparison, Rev. 2*, Calculation Package, 0006982.301.

A5.3 Continuum Dynamics, Inc. (2008), *Response to NRC RAI EMCB 201/162.*



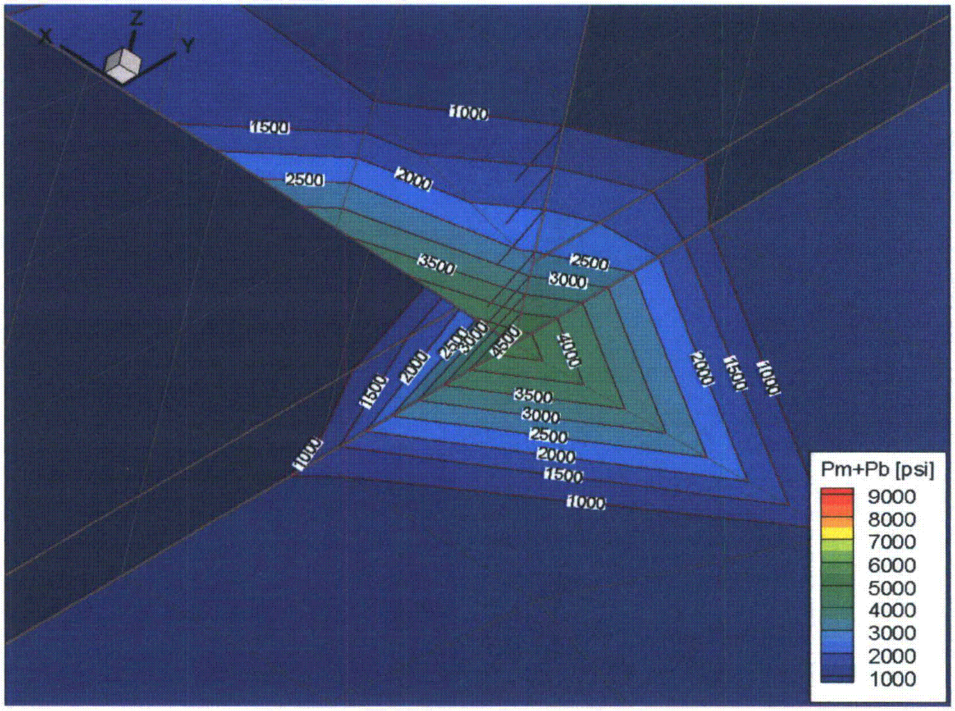


Figure A5.1. Comparison of peak membrane (Pm) and membrane+bending (Pm+Pb) stresses at the hood/stiffener/base plate junction. These contours correspond to the nominal (zero) frequency shift.

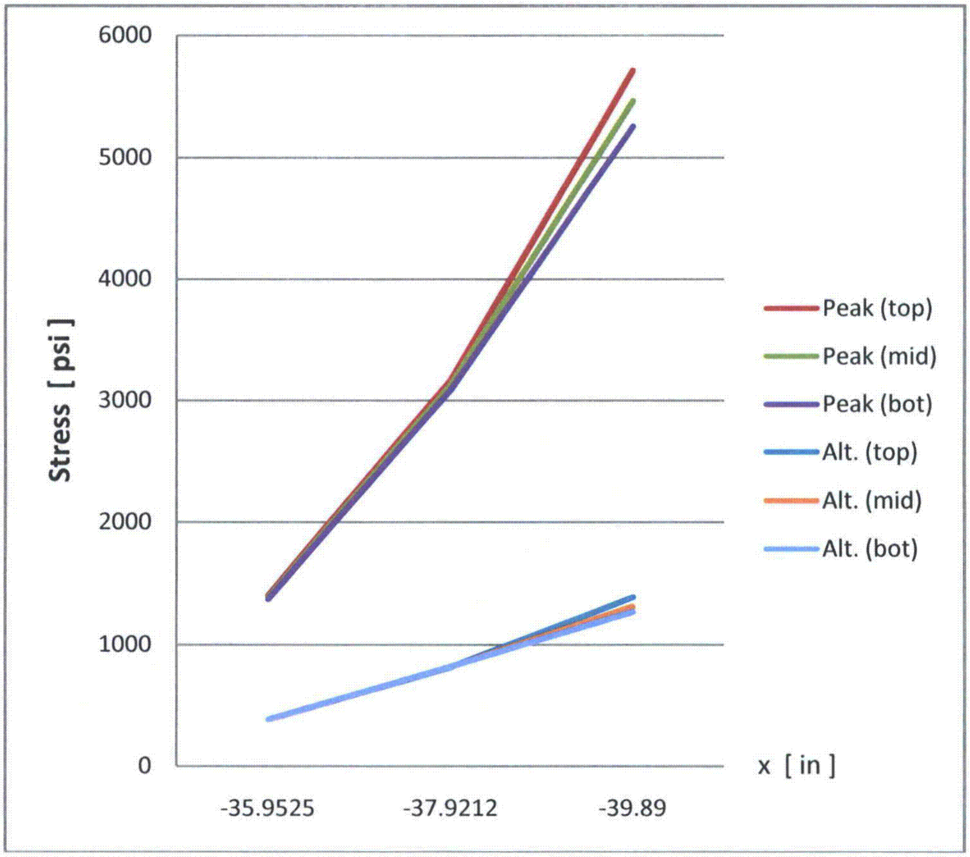


Figure A5.2. Peak and alternating stress intensities along the bottom edge of the hood stiffener as a function of location x. Stress intensities are reported for the top, middle, and bottom surfaces of the edge. The hood/stiffener/base plate junction lies at x = -39.89 in. Results correspond to the limiting frequency shift (-5%).

Piping & NDE

RAI B1

Table 2.1.4 of NEDC-33351 provides a summary of Category D and E dissimilar metal welds.

RAI B1a)

Please provide information on Category B and C welds, if any, and provide information concerning the stress improvement process and the size of the cracks determined by the subsequent examination.

NMPNS Response RAI B1a)

NMP2 has no welds in Category B or C.

RAI B1b)

Please describe the disposition of Category E welds listed in the table, whether they have been reinforced by weld overlay or mitigated by stress improvement treatment, and the size of the defects determined by the subsequent examination.

NMPNS Response RAI B1b)

NMP2 has two Category E welds:

1. Weld ID 2RPV-KC-32 (safe end-to-safe end extension)
   The mechanical stress improvement process (MSIP) was performed for this High Pressure Core Spray nozzle N16 safe-end extension weld in 1990. The results of the 2006 performance demonstration initiative (PDI) qualified automated ultrasonic testing determined that this planar flaw was 2.30" long by 0.167" in depth (20% through-wall). Full structural weld overlay repair will be performed if the flaw depth reaches 41% through-wall.

2. Weld ID 2RPV-KB-20 (nozzle-to-safe end)
   The original flaw in this Feedwater nozzle N4D weld was 5.0" long by 0.84" through-wall. Following application of a full structural weld overlay repair, the remaining ligament from the outside diameter of the overlay to the tip of the flaw is 0.98". The weld overlay was performed in accordance with American Society of Mechanical Engineers (ASME) Code Case N-504-2.

RAI B1c)

For all welds other than Category A welds, describe the augmented inspection programs and discuss their adequacy in light of the EPU.

NMPNS Response RAI B1c)

NMPNS has implemented an augmented intergranular stress corrosion cracking (IGSCC) inspection program in accordance with Generic Letter 88-01, NUREG-0313, and as modified by BWRVIP-75 for IGSCC Category D weld examination frequency using normal water chemistry. NMPNS has implemented ASME Section XI, Appendix VIII for the performance demonstration for ultrasonic examination systems administrated through the Electric Power Research Institute (EPRI) PDI program. Appendix VIII provides the requirements for the performance demonstration for ultrasonic examination procedures, equipment, and personnel to detect and size flaws.

Provided below is a summary by IGSCC Category of the current augmented program. EPU does not change the temperature, pressure or chemistry of these process fluids. The assumptions of the GL 88-01 program remain bounding for the radiolytic conditions for EPU. Refer to the response to RAI B2 below for a discussion of hydrogen water chemistry (HWC).

IGSCC Category B and C Weldments

NMP2 has no Category B or C welds.

IGSCC Category D Weldments - Non-resistant materials; no stress improvement

IGSCC Category D welds are examined at a frequency of 100% of the population (47 welds) every six years. NMP2 is on a two-year refueling cycle, so the frequency is 100% every three refueling outages.

IGSCC Category E Weldments - All welds included in this category are weld overlays

IGSCC Category E welds are inspected using a 25% sample (2 welds) every ten years. Fifty percent (1 of the 2 welds) will be completed within the first 6 years of the interval.

IGSCC Category F and G Weldments

NMP2 has no Category F or G welds.

RAI B2

Oxygen content in the coolant is expected to increase due to increased radiolysis of water resulting from EPU. Since hydrogen water chemistry (HWC) is being employed, describe how the electrochemical potential measurements will be made to ensure that the hydrogen injection rate is adequate to maintain the effectiveness of HWC at the most limiting locations.

NMPNS Response RAI B2

NMPNS does not use HWC alone for IGSCC mitigation. NMPNS uses the On-line NobleChem™ (OLNC) process of noble metal injection along with HWC injection for IGSCC mitigation of piping and reactor internals.

Original guidelines for an effective IGSCC mitigation program with HWC or noble metal chemical addition (NMCA) were issued in BWRVIP-62 in December 1998. Chapter 6 of BWRVIP-190, BWR Water Chemistry Guidelines, 2008 Revision, includes updates to the chemistry control parameters for IGSCC mitigation. OLNC, being relatively new, is introduced in Appendix C of BWRVIP-190, so the chemistry control parameter tables in Chapter 6 do not explicitly address OLNC. However, the H_2:O_2 molar ratio guidelines for NMCA plants are equally applicable to an OLNC plant.

NMPNS does not monitor electrochemical potential (ECP), so the primary parameters monitored for mitigation are catalyst loading (with the Mitigation Monitoring system (MMS)) and the measured H_2:O_2 Molar Ratio (by means of reactor water chemical analysis).

At EPU conditions, the 100% power hydrogen injection rate will be increased to the EPU predicted value of 17.6 scfm (presently 15 scfm). This will mitigate the increased oxygen generation due to the increased radiolysis at the power levels of EPU. The predicted injection rate is preliminary; therefore, testing/monitoring under EPU conditions will be performed to obtain the final injection rate for mitigation at EPU.

After EPU implementation, NMPNS will continue to use our established IGSCC mitigation monitoring program to measure the H_2:O_2 Molar Ratio. The HWC hydrogen injection rate will be changed as needed after EPU implementation to assure that a molar ratio of three or more is maintained.

Chemical Engineering

RAI C1

Table 2.1-5 of NEDC-33351P, "Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (PUSAR)," shows several entries where the predicted flow accelerated corrosion wear rate due to the power uprate using CHECWORKS™ decreases. For many of these entries, temperature and velocity increase, and oxygen is unchanged or decreases between current and EPU conditions. Examples include, "Cond Htr 5 to Header," and "FW Pmp to Balance Ln." Please discuss the reasons for the predicted decrease in flow-accelerated corrosion.

NMPNS Response RAI C1

The rate of Flow Accelerated Corrosion (FAC) is a complex process that is interdependent on many variables including temperature, velocity, oxygen concentration, steam quality, etc. Each variable impacts the overall wear rate for a component differently. The algorithm in the CHECWORKS™ code has the ability to determine the overall impact on wear rate based on changes in each variable. The rate of FAC is related to the interaction of the parameters; thus, the primary reason for the predicted decrease under EPU conditions is associated with the change in operating temperature.

The influence of temperature is represented by a bell curve. Flow accelerated corrosion rates increase as temperature increases up to approximately 300°F and then decrease as the temperature continues to increase beyond 300°F. The slopes of the bell curve are quite steep, which results in a relatively large decrease in wear rate based on a relatively small increase in temperature.

The influence of velocity on the rate of flow accelerated corrosion is fairly linear. The slope of the velocity curve is relatively flat indicating that larger changes in velocity will have a lesser impact on rate of FAC degradation verses temperature.

Evaluation of the entries in Table 2.1-5 of the PUSAR with negative changes in the predicted FAC wear rate indicates that the increase in temperature resulted in a larger overall reduction in the predicted wear rate than the corresponding increase from velocity. This results in a net reduction in the predicted wear rate.

Fire Protection

RAI D1

Attachment 11 to NEDC-33351P Revision 0, Section 2.5.1.4, "Fire Protection," states that "...Any changes in physical plant configuration or combustible loading as a result of modifications to implement the extended power uprate (EPU) will be evaluated in accordance with plant modification and fire protection programs..." Clarify whether this request involves plant modifications or physical changes to the fire protection program. If any, the staff requests the licensee to identify proposed modifications and discuss impact of these modifications on plant's compliance with fire protection program licensing basis, Title 10 of the Code of Federal Regulations (10 CFR) 50.48, or applicable portions of 10 CFR 50, Appendix R.

NMPNS Response RAI D1

None of the plant modifications listed in Attachment 6, Modifications to Support EPU, represent physical changes to plant fire protection equipment or systems to support EPU conditions. However, this request does involve a modification to the fire protection program. The plant fire protection program licensing basis will be modified as described in Section 2.5.1.4 to change the acceptance criteria for reactor vessel fuel cladding integrity in response to a postulated 10 CFR 50 Appendix R fire event at EPU conditions. Currently, Updated Safety Analysis Report (USAR) vessel water level performance criteria for Appendix R safe shutdown requires water level to remain above top of active fuel (TAF). The criteria will be changed from vessel water level remaining above TAF to assuring that peak clad temperature (PCT) remains below 1500°F in accordance with GE BWROG report, "BWROG Position on the Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths," which has been accepted by the NRC in a letter to the BWROG dated December 12, 2000 (Accession No. ML003776828).

RAI D2

The NRC staff notes that Attachment 11 to NEDC-33351P Revision 0, Section 2.5.1.4, "Fire Protection," states that "...The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for EPU conditions. The operator actions required to maintain the consequences of a fire are defined..." The NRC staff requests the licensee to verify that additional heat in the plant environment from the EPU will not (1) interfere with required operator manual actions being performed at their designated time, or (2) require any new operator actions to maintain hot shutdown and then place the reactor in a cold shutdown condition.

NMPNS Response RAI D2

The effect of EPU process temperature and electrical heat load changes were evaluated for impact on normal area temperatures. Areas of the plant where operator manual actions are being performed for safe shutdown following a fire were reviewed to determine if additional heat due to EPU conditions could adversely impact those defined operator actions. Areas requiring operator entry include various locations in the electric tunnels and in the control, turbine, reactor, and normal switchgear buildings. EPU conditions only impact the areas of the reactor building exposed to Residual Heat Removal (RHR) system process piping. The RHR process piping temperature increase is small and the maximum process piping temperature is bounded by the heat loss analysis assumption of 212°F in the suppression pool; therefore, EPU does not impact operator actions.

There are no new post-fire operator actions required due to EPU.

RAI D3

The NRC staff notes that Attachment 11 to NEDC-33351P Revision 0, Section 2.5.1.4, "Fire Protection," states that "...The results show that the peak fuel cladding temperature, reactor pressure and containment pressures and temperatures are below the acceptance limits and demonstrate that there is sufficient time for the operator to perform the necessary actions to achieve and maintain cold shutdown conditions..." The NRC staff requests the licensee to discuss the operator action response time, including any assumptions that may have been made in determining that the operator manual actions are feasible and reliable and can be accomplished to achieve and maintain hot and then cold shutdown condition.

NMPNS Response RAI D3

Refer to the response to RAI D4, which addresses operator action response time to achieve and maintain hot shutdown, including any assumptions made in determining that the operator manual actions are feasible and reliable. The response to this RAI addresses the operator actions needed to achieve and maintain cold shutdown.

After the plant is stabilized with adequate core cooling assured by using Reactor Core Isolation Cooling (RCIC) or pseudo Low Pressure Coolant Injection (LPCI), operator action is needed to bring the plant to cold shutdown using either normal shutdown cooling or alternate shutdown cooling. The Appendix R analysis makes the assumption that shutdown cooling is established at greater than 120 minutes from initiation of the fire. This assumption remains unchanged due to EPU. The actions needed to bring the plant to cold shutdown using the shutdown cooling mode of the RHR system are similar to those required under normal plant conditions from the control room. However, some of the actions require local operation instead of remote operation from the control room. These actions include:

If normal shutdown cooling is used:

- Local operation of Reactor Recirculation pump breakers. These actions are performed inside the north and south auxiliary bays of the reactor building on elevation 240, and the east and west normal switchgear building on elevation 261.
- Local power operation of Reactor Recirculation pump discharge valve 2RCS*MOV18B(A). This action is performed in the reactor building on elevation 261.
- Local power operation of the Low Pressure Coolant Injection (LPCI) injection valve 2RHS*MOV24A(B). This action is performed in the Division 1(2) switchgear rooms.
- Local manual verification that 2RHS*MOV24A(B) is closed. This action is performed on reactor building elevation 289.

If alternate shutdown cooling is used:

- Local manual operation of 2RHS*MOV24A(B). This action is performed on reactor building elevation 289.
- Local monitoring of SRV tail pipe temperatures. This action is performed in the control building, elevation 261 west cable chase.

While there are manual operator actions at various locations in the plant, these actions are feasible and reliable in terms of accessibility during an Appendix R fire event. The actions do not add significant operator action response time to reach cold shutdown from the hot shutdown condition. Analysis of Alternate Shutdown Cooling under EPU conditions concluded the system is capable of bringing the reactor from hot shutdown to cold shutdown conditions within approximately 50 hours which represents an approximately 16 hour increase in the time needed to reach cold shutdown and is within the Appendix R 72-hour cold shutdown requirement. The additional time is due to the increased decay heat load associated with EPU conditions. Since there are no changes to the operator actions for achieving cold shutdown, there is no difference in expected operator action response time.

<u>RAI D4</u>

The NRC staff notes that Attachment 11 to NEDC-33351P Revision 0, Section 2.5.1.4.1, "10 CFR 50 Appendix R Fire Event," states that "...The results of Appendix R evaluation for current license thermal power CLTP and EPU provided in Table 2.5-1 and Figures 2.5-1 through 2.5-4 demonstrate that the fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions..." The NRC staff requests the licensee to provide actual time for the operator to perform the necessary actions, including the anticipated "time margin" between when the actions are completed and when any thermal-hydraulic constraints are likely to be reached.

<u>NMPNS Response RAI D4</u>

In preparation for submittal of the LAR for extended power uprate, NMPNS evaluated operator actions and response times needed to mitigate an Appendix R fire event. The performance objective is to achieve hot shutdown and then to achieve and maintain a cold shutdown condition. This evaluation determined that no additional operator actions are required to meet the performance objective. For CLTP conditions, operators have demonstrated that the actions for the control room evacuation to achieve hot shutdown can be performed in 9 minutes, which is the time needed to initiate a reactor vessel blowdown from the Remote Shutdown panel and enable injection by operation of the Low Pressure Coolant Injection (LPCI) system. The operator action time assumed under EPU conditions is 10 minutes, which provides a 3.4 minute margin to the calculated time to reach the Minimum Steam Cooling Water Level of -39 inches actual reactor water level. Fuel clad temperature remains well below 1500°F with reactor water level at the Minimum Steam Cooling Water Level.

In the analysis for EPU, the basis for operator action time was changed from core submergence to steam cooling as the acceptance criteria for core cooling. Core submergence is defined as water level at TAF

(-14 inches actual water level). Minimum Steam Cooling Water Level is defined as -39 inches actual water level. The change in acceptance criteria explains why the allowable operator action time is increased from 9 minutes to 13.4 minutes with no change in operator actions at EPU conditions.

In June 2006, the NRC independently observed a demonstration, by licensed operators, of a simulated transfer of plant control from the main control room to alternate safe shutdown panels, and a simulated plant shutdown to hot standby conditions from the remote shutdown panel. The team primarily focused on the portion of the procedures associated with achieving stable hot shutdown conditions within the time frames assumed in the safe shutdown thermal hydraulic analysis. The NRC team evaluated the approximate time to perform critical steps, such as establishing makeup flow to the reactor vessel, to assess the ability of operators to maintain plant parameters within the required limits. As documented in Nine Mile Point Nuclear Station Units 1 and 2 NRC Triennial Fire Protection Inspection Reports 05000220/2006006 and 05000410/2006006, dated July 6, 2006, no findings of significance were identified by the NRC during this inspection.

In June 2009, the NRC independently confirmed that NMP2 operators are able to meet the assumed action times to maintain effective reactivity control, reactor coolant makeup, reactor decay heat removal, process monitoring instrumentation, and support systems functions during a shutdown from outside the control room with and without the availability of offsite power. This is documented in Nine Mile Point Nuclear Station Units 1 and 2 NRC Triennial Fire Protection Inspection Reports 05000220/2009006 and 05000410/2009006 and Exercise of Enforcement Discretion, dated August 3, 2009.

The following is an excerpt from the 2009 inspection report:

> The [NRC] team verified that the training program for licensed and non-licensed operators included alternative shutdown capability. The team also verified that personnel required for safe shutdown using the normal or alternative shutdown systems and procedures are trained and available onsite at all times, and were exclusive of those assigned as fire brigade members.
>
> The [NRC] team reviewed the adequacy of procedures utilized for post-fire safe shutdown and performed an independent walk through of procedure steps to ensure the implementation and human factors adequacy of the procedures. The team also verified that the operators could be reasonable expected to perform specific actions within the time required to maintain plant parameters within specified limits. Time critical actions, which were verified, included the restoration of alternating current (AC) electrical power, establishing the remote shutdown and local shutdown panels, establishing reactor coolant makeup, and establishing decay heat removal.

While the referenced NRC inspection reports are based on current licensed power conditions, the assessment remains valid since there are no changes to required operator actions and the operator action time is longer at EPU conditions.

RAI D5

The results of the Appendix R evaluation for CLTP and EPU are provided in Table 2.5-1 and Figures 2.5-1 through 2.5-4. The NRC staff notes in Table 2.5-1 that at EPU condition, there is an increase in the suppression pool bulk temperature of 198.1°F, 9.5°F above the current suppression pool bulk temperature of 188.6°F. Do the NMPNS Unit 2 safe shutdown instructions credit any operator manual action in the secondary containment? If any, discuss how this operator manual action can be accomplished within the available time at higher suppression pool bulk temperature (e.g., manually opening the main steam relief valves). In addition, if a low-pressure coolant injection (LPCI) pump is used for safe-shutdown for NMP2, how does the licensee ensure adequate net positive suction head (NPSH) available to the LPCI pump throughout the Appendix R event?

NMPNS Response RAI D5

The NMP2 safe shutdown instructions do credit operator manual actions in the secondary containment (i.e., reactor building). The effect of EPU process temperature and electrical heat load changes, including the increase in suppression pool bulk temperature, were evaluated for impact on secondary containment area temperatures. The electrical heat load is the dominant heat load and is unchanged by EPU. The heat load from the process pipe temperatures represents 17% of the total for the RHR pump rooms. The EPU decay heat increases the maximum suppression pool temperature by 9.5°F; however the design analysis bounds the 9.5°F increase because it assumed a design temperature of 212°F in the suppression pool. Therefore, operator manual actions in the secondary containment credited for safe shutdown are not impacted by the increase in suppression pool bulk temperature.

Low pressure coolant injection (pseudo LPCI) is used for safe-shutdown for NMP2. Adequate net positive suction head (NPSH) is ensured for the increase in suppression pool bulk temperature under EPU conditions since the NPSH calculations use a suppression pool bulk temperature of 212°F, at atmospheric pressure, which bounds the EPU suppression pool bulk temperature.

RAI D6

Some plants credit aspects of their fire protection system for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If the NMPNS, Unit 2, credits its fire protection system in this way, the EPU LAR should identify the specific situations and discuss to what extent, if any, the EPU affects these "non-fire-protection" aspects of the plant fire protection system. If NMP2 does not take such credit, the NRC staff requests that the licensee verify this as well.

NMPNS Response RAI D6

NMPNS does not credit the fire protection system to support the design basis for non-fire protection functions at NMP2.

Instrumentation & Controls

RAI E1

Regarding the setpoints below, provide documentation of the methodology used for establishing the limiting setpoint (or NSP) and the limiting acceptable values for the As-Found and As-Left setpoints as measured in periodic surveillance testing. Indicate the related Analytical Limits and other limiting design values (and the sources of these values).

- *Average Power Range Monitor Flow Biased Simulated Thermal Power- Upscale*
- *Main Steam Line Flow - High*

NMPNS Response RAI E1

- Average Power Range Monitor (APRM) Flow Biased Simulated Thermal Power- Upscale

Adjustments to the APRM Flow-Biased Scram Analytic Limit (AL) lines [[

]]

The Allowable Value (AV) and Nominal Trip Setpoint (NTSP) are calculated [[
]] This methodology is applicable and has been accepted in NEDC-33004P-A, "Licensing Topical Report Constant Pressure Power Uprate," Revision 4, Class III, July 2003.

Instrument reference accuracy is used for the As-Found and As-Left tolerances. The nominal trip setpoints and As-Left tolerances are incorporated into periodic surveillance procedures and are selected to ensure that the actual setpoints do not exceed the Allowable Value (AV) between successive channel calibrations.

Design values for the AV and AL are given in Section 2.2, Technical Specification Changes, and Section 3.2.1, Setpoint Calculation Methodology, of the LAR Enclosure titled "Evaluation of the Proposed Change."

- Main Steam Line Flow - High

The Main Steam Line Flow – High setpoint AL was retained at 140%; however, by being set at 140% of the EPU condition rated MSL flow rate, there will be a change in the differential pressure settings.

The AV is 184.4 psid and the NTSP is 183.0 psid. While the AL for EPU conditions remained the same at 140% of the rated steam flow, the AV and NTSP both increase in units of psid due to the change in the higher absolute mass flow rate for EPU at the AL. The AV and NTSP are re-calculated using approved GEH setpoint methodology (i.e., NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," Class 3, September 1996). A sample calculation demonstrating the application of this methodology is provided in Section 2.4.2 of the PUSAR (LAR Attachment 11).

Instrument reference accuracy is used for the As-Found and As-Left tolerances. The nominal trip setpoints and As-Left tolerances are incorporated into periodic surveillance procedures and are selected to ensure that the actual setpoints do not exceed the Allowable Value (AV) between successive channel calibrations.

Design values for the AV and AL are given in Section 2.2, Technical Specification Changes, and Section 3.2.1, Setpoint Calculation Methodology, of the LAR Enclosure titled "Evaluation of the Proposed Change."

<u>RAI E2</u>

For non-SL-related setpoint, "Main Steam Line Flow – High," describe the measures to be taken to ensure that the associated instrument channel is capable of performing its specified safety functions in accordance with applicable design requirements and associated analyses. Include in your discussion information on the controls you employ to ensure that the as left trip setting after completion of periodic surveillance is consistent with your setpoint methodology. Also, discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded." If the controls are located in a document other than the Technical Specifications (e.g., plant test procedure), describe how it is ensured that the controls will be implemented.

<u>NMPNS Response RAI E2</u>

Refer to Section 3.2, Technical Specification Instrument Setpoint Changes, of the LAR Enclosure titled "Evaluation of the Proposed Change."

Nominal trip setpoints are specified in the setpoint calculations and incorporated into applicable Surveillance Test Procedures. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the AV between successive channel calibrations.

The AV and NTSP were re-calculated using NRC-approved GEH methodology in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," and NEDC-32889P, "General Electric Methodology for Instrumentation Technical Specification and Setpoint Analysis." A sample calculation demonstrating the application of this methodology is provided in Section 2.4.2 of the PUSAR (LAR Attachment 11). The EPU AV for this function is provided in Section 2.2 of this TS enclosure. There is a plant specific program which verifies that this instrument channel functions as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

The Surveillance Test Program establishes the administrative controls for testing. As-Left trip settings are controlled under procedures based on the Surveillance Test Program. As-Found settings found outside specified as-found tolerances are evaluated for functionality through the NMPNS 10CFR50, Appendix B , Criterion XVI, Corrective Action Program. A channel is inoperable if its actual trip setpoint is not within its required AV. Operability determinations are integral to the corrective action program. When the condition described in a condition report represents an operability concern, an operability determination is completed. Return of the degraded or non-conforming component to a fully-qualified status is addressed under the corrective action program. Instrument reference accuracy is used for the As-Found and As-Left tolerances. An As-Left setting is procedurally required to be within the As-Left tolerance prior to returning the channel to service. If the As-Found setting is outside the required As-Found tolerance, the device is reset to within the As-Left tolerance.

The initial LAR submittal includes changes to the applicable Technical Specification (TS) Bases that are consistent with the NRC staff's position on complying with 10 CFR 50.36, as discussed in Regulatory Issue Summary (RIS) 2006-17 and further clarified by TSTF-493, Revision 4, for non-SL-Related LSSS functions. Specifically, the following changes to TS Bases Section B 3.3.6.1 for the Main Steam Line High Flow MSIV Isolation function are shown in Attachment 2 of the initial LAR submittal:

> "There is a plant specific program that verifies that this instrument channel functions as required by verifying the As-Found and As-Left settings are consistent with those established by the setpoint methodology."

Containment & Ventilation

RAI F1

NEDC-33351P Rev. 0 Section 2.2.1 states that the EPU has no effect on the mass and energy released from a high energy line break (HELB) in a steam line. Provide clarification, or reference, previously submitted documentation that justifies why EPU increased steam flow rates will not increase the mass or energy release from a HELB in a steam line. If a flow restricting nozzle or orifice is the justification, provide verification that a break cannot occur between the flow restrictor and the reactor vessel.

NMPNS Response RAI F1

NEDC-33351P, Rev. 0, Section 2.2.1, provides documentation of the confirmation performed for the NMP2 EPU project that the CLTR generic disposition for releases from a high energy line break of a line containing steam is applicable for the NMP2 constant pressure power uprate project. The CLTR generic disposition concludes that releases from a HELB of a line containing steam are unaffected since the initial conditions in the line are based on steam conditions in the vessel, downstream of the steam dryer (saturated steam at a pressure equal to the dome pressure) and the dome pressure does not change for the constant pressure power uprate.

Additional information related to the impact of increased core power on (1) the post-break vessel depressurization rate, and (2) the steam line break liquid carryover fraction, is contained in the NRC Safety Evaluation Report (SER) for the CLTR and supports the conclusion that plant specific evaluations are not required to evaluate the impact of a constant pressure power uprate on the releases from a HELB of a line containing steam.

Credit for the restricting orifice is not part of the justification of the CLTR generic disposition for HELBs from steam line breaks.

RAI F2

NEDC-33351P Rev. 0 Section 2.2.1 states that the results of the NMP2 evaluation of HELBs are provided in Table 2.2-1. Table 2.2-1 is for liquid line breaks. Please provide the table that provides the results for HELBs from steam line breaks.

NMPNS Response RAI F2

Plant specific evaluations were not performed for HELBs from steam line breaks. The confirmation of the generic CLTR disposition for HELBs from steam line breaks is documented in NEDC-33351P, Rev. 0, Section 2.2.1.

RAI F3

Where is the location of the break used to calculate the design basis loss-of-coolant accident (DBLOCA) peak values provided in Table 2.6-1?

NMPNS Response RAI F3

The break postulated by the design basis loss-of-coolant accident (DBLOCA) peak values in Table 2.6-1 is a Recirculation Suction Line Break (RSLB). It is modeled as a double-ended break located at the nozzle.

RAI F4

The sub-compartment pressurization evaluation for the drywell head is based on a postulated break in the reactor core isolation cooling (RCIC) head spray line. Provide a discussion or reference a previously docketed discussion that documents a break in the RCIC head spray line is the limiting break for the drywell head sub-compartment pressurization.

NMPNS Response RAI F4

The NMP2 USAR, Section 6.2.1.2, provides the details of the design basis drywell head subcompartment pressurization evaluation, including the pipe breaks analyzed. The RCIC break as identified would be a high energy steam line break, and the generic evaluations provided in the CLTR (NEDC-33004P-A, Section 10.1) would apply.

RAI F5

Section 4.5, page 2-258, is the radioiodine inventory indicated the same as the iodine loading on the standby gas treatment system (SGTS) charcoal adsorber?

NMPNS Response RAI F5

[[..]] This is not the same as the iodine loading on the SGTS charcoal adsorber.

RAI F6

Provide the SGTS charcoal adsorber loading in terms of milligrams radioiodine per grams of activated carbon.

NMPNS Response RAI F6

The actual [[..]] in terms of radioiodine per grams of activated carbon. This value can be found in NEDC-33351P, Revision 0, Section 2.5.2.1, as shown in the table column titled [[..]] can be found in Footnote 1 of Table 2.5-2.

RAI F7

Section 4.5 provides a maximum component temperature with the higher iodine inventory. Is the component in discussion the SGTS HEPA, the charcoal adsorber, or both? Is the temperature based on local air temperature, radioiodine decay heat, or a combination of both?

NMPNS Response RAI F7

Two cases are addressed in the cited paragraph. The maximum component temperature is calculated as a comparator for all susceptible SGTS filter train components (which, if failed, could adversely affect train operation) under normal system flow conditions. Second, the maximum component temperature for the most susceptible component (the charcoal adsorbers) is determined under conditions of minimum cooling flow. The total heating used for these calculations is based on the inlet air temperature plus the sum of the decay heat experienced by the HEPA filters and the decay heat experienced by the charcoal adsorbers.

RAI F8

Table 2.5-2 provides SGTS radioiodine removal capacity parameters. It is not clear if the parameters are for all trains or for one train. Is the mass of activated carbon for each filtration train? How many charcoal adsorber modules are installed in each SGTS filter train? Is the maximum charcoal adsorber temperature indicated based on radioiodine decay heat at minimum specified airflow (or no airflow)?

NMPNS Response RAI F8

The values provided in Table 2.5-2 are all for a single SGTS train. The GEH analysis is performed based on one SGTS train being in operation throughout the post-LOCA evaluation period (days) for decay heat and iodine loading analysis.

The mass of activated carbon is for one single SGTS train. The charcoal is bulk loaded. There are no individual adsorber modules.

The maximum charcoal adsorber temperature is based on radioiodine decay heat at the minimum specified airflow.

RAI F9

The reactor power listed in Table 2.5-2 does not match the proposed power uprate to 4067 MWt. Provide an updated Table 2.5-2, "SGTS Iodine Removal Capacity Parameters" based on the requested 4067 MWt reactor power.

NMPNS Response RAI F9

The parameters presented in Table 2.5-2 of NEDC-33351P, Revision 0 are input values, with one column for the generic CLTR analysis and the other column for the NMP2-specific analysis. The listed NMP2 reactor power value of 3988 MWt has not been multiplied by the 1.02 uncertainty factor; however, the NMP2 alternative source term analyses (performed per NRC Regulatory Guide 1.183) did use a reactor power value of 4067 MWt.

RAI F10

Section 2.5.2.1, page 2-178 and Table 2.5-2 discuss the charcoal adsorber temperature with minimum airflow. The discussion on page 2-178 provides the adsorber temperature with "a failed fan with minimum cooling flow". Provide a discussion or reference a previously docketed discussion that provides assurance minimum cooling will be maintained with a failed fan. If the alternate SGTS train provides minimum airflow, discuss:

RAI F10 a)

If damper manipulation is required to provide minimum air flow (manual, automatic, or both);

NMPNS Response RAI F10 a)

The alternate SGTS train provides minimum airflow for decay heat removal in the train with a failed fan. Manual operator action is required to open the decay heat removal inlet valve. This action is accomplished using a hand control switch in the Control Room.

RAI F10 b)

Control Room indications that minimum cooling airflow is required and maintained;

NMPNS Response RAI F10 b)

Control room indication of a SGTS charcoal adsorber high temperature condition is provided by an SGTS Train A(B) System Trouble alarm. There is a separate alarm for each division. Computer points from a temperature element provide input to this alarm at 200°F and 300°F.

The success of actions to establish and maintain minimum required cooling air flow may be confirmed by monitoring charcoal adsorber temperature alarm response. No flow monitoring is required.

RAI F10 c)

If any manual actions to assure minimum air flow are addressed in the emergency operating procedures;

NMPNS Response RAI F10 c)

Actions to identify the need for and to establish minimum air flow are contained in the respective Alarm Response Procedure (ARP) and completed using the off-normal section of the system Operating Procedure (OP).

RAI F10 d)

The impact of minimum cooling flow on the operating SGTS train;

NMPNS Response RAI F10 d)

SGTS normal and minimum cooling flows are unchanged by EPU. Therefore the SGTS system function for reactor building drawdown is unaffected by EPU. Table 2.5-2 reflects that the current normal air flow and minimum cooling flow both exceed the Generic Input Criteria and thus are adequate to accommodate normal operation of the inservice train and increased decay heat removal from the inactive train without any changes to the current SGTS design or operation.

RAI F11

What will be the maximum temperature maintained in the drywell during extended power operation? How does this compare with the initial temperature assumed for the drywell for containment accident analysis?

NMPNS Response RAI F11

The long-term containment accident analyses, directed primarily at the suppression pool temperature response, assumed the maximum allowable average drywell temperature per TS 3.6.1.5 of 150°F initial drywell temperature.

The peak containment pressure is defined by the short term response where the maximum pressure is defined by a lower drywell initial condition temperature assumption. The original USAR licensing basis analysis that established the peak calculated accident pressure (Pa = 39.75) assumed a nominal temperature of 135°F. The current typical 100% power drywell average temperature is approximately 110°F and the evaluation of the impact of EPU on the operating temperature is calculated to be less than a 1°F increase. Therefore, the small increase in normal operating temperature has no impact on the peak containment analysis pressure. It is noted that the licensing basis containment peak calculated pressure assumes nominal operating drywell conditions, and the lower bound initial conditions are evaluated relative to the margin to the structural design allowable of 45 psig.

Component Performance & Testing

RAI G1

The first paragraph of Section 2.2.4 states that NMP2 evaluated the lessons learned from the motor-operated valve (MOV) program and applied those lessons learned to other safety-related power-operated valves. Please provide specific examples of the lessons learned from the MOV program that were applied to other safety-related power-operated valves.

NMPNS Response RAI G1

A Constellation fleet procedure has been created for the air operated valve (AOV) program that reflects the lessons learned from the MOV program. Elements that have been successful for the MOV program that are included in the procedure for safety-related AOVs include:

- Documentation of the design basis operating requirements for the valve and adequacy of the actuator to meet those requirements.
- Establishment and control of set-up criteria.
- Periodic testing.

RAI G2

*The last paragraph of Section 2.2.4.2 discusses Generic Letter 95-07 and states that MOVs were modified to provide mitigation of pressure-locking occurrences. Please discuss if a thrust-prediction methodology is used to demonstrate that valves 2ICS*MOV122, RCIC Steam Exhaust to Suppression Pool, and 1ICS*MOV128, RCIC Steam Supply Inboard Isolation, are capable of opening during pressure-locking conditions. Please explain if the increase in suppression pool temperature due to EPU could effect the pressure-locking calculation for 2ICS*MOV122 if a thrust-prediction methodology is used to demonstrate that this valve is capable of operating during pressure-locking conditions.*

NMPNS Response RAI G2

An analysis using a thrust prediction methodology was performed for both 2ICS*MOV122 and 2ICS*MOV128. The increase in suppression pool temperature that results from EPU will not impact the analysis results. This is because the valves direct steam from the reactor vessel to the RCIC turbine or suppression pool and are not directly exposed to suppression pool water. Therefore, the valve temperatures are impacted only by the pressure and temperature conditions of the inlet steam which are not changing as a result of EPU.

RAI G3

The Technical Evaluation in Section 2.8.4.5 states that there is 31.6 psi margin between the maximum reactor upper plenum and the standby liquid control system (SLCS) pump relief valve setpoint. This 31.6 psi margin includes a SLCS pump relief valve setpoint tolerance of 3% but it appears that the margin does not include an overall combined accuracy of the instrumentation used to perform the SLCS pump relief valve setpoint test. Please explain how the overall combined accuracy of the instrumentation used to perform SLCS pump relief valve setpoint tests was accounted for when calculating the margin between the maximum reactor upper plenum and the SLCS pump relief valve setpoint.

NMPNS Response RAI G3

The SLCS pump relief valve setpoint maintains a 31.6 psi margin to relief valve lift in addition to the 3% set pressure tolerance (42 psi). Therefore, the combined margin is 73.6 psi.

The overall combined accuracy of the test instrument is not accounted for in the margin calculation. However, the instrument used for the set pressure determination meets the ASME Operation and Maintenance (OM) Code, 2004 Edition, which is adopted in the NMP2 Inservice Testing program. The gauge accuracy used in the set pressure determination is ±0.5% compared to the code requirement of ±1%.

The combined margin of 73.6 psi is recognized to provide minimally sufficient margin for SLCS pump relief valves, at both CLTP and EPU conditions. As such, the piping design pressure is being rerated and the relief valve set pressure increased to provide additional margin.

RAI G4

The Technical Evaluation in Section 2.8.4.5 states that the SLCS pump relief valves are periodically tested. Please verify that as-found setpoint test history for the SLCS pump relief valves demonstrates that the SLCS pump relief valves were consistently within the 3% tolerance.

NMPNS Response RAI G4

A review of "as found" test data for the SLCS pump relief valves going back to 1995 shows that the "as found" settings have been within the 3% tolerance.

RAI from NRC Letter Dated November 18, 2009 (SNPB-1, Non Proprietary)

It has come to the NRC staff's attention that there is an error in the linear heat generation rate (LHGR) uncertainty analysis provided in the approved interim methods licensing topical report (IMLTR), NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains." The LHGR uncertainty analysis includes the local power range monitor (LPRM) update uncertainty of [[______]] percent. However, the basis for this value is the bundle power, whereas the LHGR is monitored on a nodal level with uncertainties that take into account the peak pin power uncertainty.

Appendix B of NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limiting MCPR Evaluations," provides a revised LPRM update uncertainty for the LHGR evaluation of [[____]] percent. Appendix B of NEDC-32694P-A provides a calculation of the LHGR uncertainties and calculates this value as [[___]] percent. When certain parameters are updated to account for [[______________________]] the LHGR uncertainty is [[____]] percent.

When this update uncertainty is corrected in the IMLTR LHGR uncertainty calculation (see Table 2-11 from the IMLTR), the resultant LHGR uncertainty is [[______]] percent. This value remains below the value assumed in the thermal-mechanical (T-M) analysis.

However, the value of the LPRM update uncertainty is a function of the exposure interval between LPRM calibrations. As the exposure interval increases, the uncertainty associated with the nodal power attributed to the update uncertainty component is expected to increase. The proposed NMP2 LPRM calibration interval is defined as 1000 effective full-power hours. Since the LAR requests an increase in the licensed thermal power, the calibration interval in terms of accumulated exposure would increase. The increased interval (in terms of exposure) may exceed the interval assumed in the development of the [[____]] percent generic value.

1. *Please quantify the LPRM exposure interval proposed for NMP2 in the units of mega-watt-days per metric ton.*

2. *Please determine the LPRM update uncertainty for the nodal power consistent with, or conservatively larger than, the exposure interval between LPRM calibrations.*

3. *Please combine the update uncertainty with the other component uncertainties in Table 2-11 of the IMLTR and compare this value to the value assumed in the T-M analysis.*

4. *If the LHGR uncertainty exceeds the value assumed in the T-M analysis, please justify*

NMPNS Response RAI SNBP-1 (Non Proprietary)

1. The LPRM exposure interval proposed for NMP2 is 1000 EFPH, which translates to [[..............]] MWd/MT at the original licensed thermal power and [[..............]] MWd/MT at the extended power uprate power.

2. Appendix B of NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limiting MCPR Evaluations," provides a revised LPRM update uncertainty for the LHGR evaluation of [[............]] percent. This uncertainty continues to apply for the LPRM exposure interval proposed for NMP2 EPU and has been reaffirmed by a comprehensive study using core-monitoring information from several recent plants/cycles. This modern database covers the spectrum of BWR 4, 5, and 6 plants, lattice types D, C, and S, number of fuel assemblies from 368 to 800, and both gamma and neutron traversing in-core probes (TIPs) for periods bounding the NMP2 EPU exposure interval.

3. The combined uncertainty is [[..........]] percent and is less than the [[..........]] assumed in the thermal-mechanical (T-M) analysis.

4. Not applicable. The LHGR uncertainty does not exceed the value assumed in the T-M analysis.

PUSAR Appendix A Update NRC Acceptance Review Question (from NRC Email Dated September 18, 2009)

Disposition of Condition 9.20 in the PUSAR Appendix A We agree that the condition is not applicable to the NMP2 EPU LAR; however, we'd like to get clarification on the stated basis in the Appendix A table (Fuels)

NMPNS Response

The following replacement pages are provided in Attachment 7 for NEDO-33351, Rev. 0, and in Attachment 8 for NEDO-33351P, Rev. 0:

Page 1-7	Table 1-1 Computer Codes Used For EPU
Page 1-9	Table 1-1 Computer Codes Used For EPU (2^{nd} notes page)
Page A-10	Appendix-A Limitations from Safety Evaluation for LTR NEDC-33173P
Page A-12	Appendix-A Limitations from Safety Evaluation for LTR NEDC-33173P

ATTACHMENT 2

STEAM DRYER EVALUATION EXECUTIVE SUMMARY - REVISED (LAR ATTACHMENT 13)

ATTACHMENT 13 – STEAM DRYER EVALUATION

Table of Contents

1.0 Introduction

The Nine Mile Point Unit 2 (NMP2) steam dryer has been evaluated for extended power uprate (EPU) steam flow conditions consistent with the guidance provided in BWRVIP-182 (References 7.10 and 7.11). BWRVIP-182 was created at the request of the NRC to provide guidance that can be followed by boiling water reactor (BWR) utilities applying for a power uprate of greater than 2% of current licensed thermal power (CLTP) in demonstrating the structural integrity of their steam dryer up to the highest planned power level. BWRVIP-182 provides the overarching approach for demonstrating the structural integrity of BWR steam dryers at power uprate conditions and was developed by representatives of Boiling Water Reactor Vessel and Internals Project (BWRVIP) utilities, Electric Power Research Institute (EPRI), General Electric, and Continuum Dynamics Incorporated (CDI). The NRC has issued a draft safety evaluation (SE) (Reference 7.11) stating that the NRC staff has reviewed BWRVIP-182 and additional information the BWRVIP provided in its request for additional information (RAI) responses, and found that the report, as modified and clarified to incorporate the staff's comments, is acceptable for providing guidance on the steam dryer integrity demonstration for power uprate conditions. The NMP2 steam dryer evaluation is in full conformance to these guidelines and the NRC draft SE (Reference 7.11).

BWRVIP-182 requires that, prior to submittal of an application for power uprate, the loading on the steam dryer and associated stresses be defined at power uprate conditions and that appropriate stress margin be demonstrated. The NRC RAI No. 182-4 (c) clarification relative to the required margin in BWRVIP-182 is as follows: "In Section 7, the BWRVIP is requested to clarify the minimum required alternating stress ratio. Considering all end-to-end bias errors and uncertainties (in recent EPU approved license amendment requests such as Hope Creek) as well as stress concentration factors, a minimum stress ratio of 2 shall be maintained in steam dryer components, when fluctuating pressure load prediction on the dryer relies on MSL measurements. The minimum alternating stress ratio is defined as the endurance limit of the material divided by the maximum alternating stress. The stress margin as a percentage is defined as (minimum alternating stress ratio-1) * (100). Specifically, either the alternating stress ratio ≥ 2.0; or stress margin on alternating stress ≥ 100 percent." The NMP2 steam dryer applies this NRC-approved margin as stated in the above RAI.

BWRVIP-182 defines an overall approach for demonstrating steam dryer structural integrity that allows the use of subscale and full scale tests and analytical models. The document requires that the technical basis for and benchmarking of any analytical or testing methodologies utilized in demonstrating steam dryer integrity be documented and submitted to NRC for review, and that specific acceptance criteria and values for key parameters to be used in the evaluation of steam dryers be defined. The BWRVIP-182 guidance is intended to comply with the guidance provided in NRC Regulatory Guide (RG) 1.20, Rev. 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," issued in March 2007.

BWRVP-182 implementation guidance that states: "In accordance with the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, sections 2 through 10 of this report are "needed" and the remaining sections are for information only. The guidance provided herein shall be followed by any BWR utility submitting an application for a power uprate exceeding 2 % of CLTP after the date of publication of this report."

BWRVIP-182 refers to BWRVIP-181, Steam Dryer Repair Design Criteria (Reference 7.9), and states that for structural evaluation of existing and replacement steam dryers for power uprate, the applicant is referred to the load types, load combinations and corresponding allowable stresses defined in Section 7 of BWRVIP-181. The NMP2 steam dryer analysis for EPU was performed consistent with the original steam dryer loads adjusted for the higher EPU normal operating loads and EPU reactor internal pressure difference (RIPD) values including the flow induced vibration (FIV) load added to the design basis load combinations as recommended in Section 7 of BWRVIP-181. The NRC has issued a draft SE (Reference 7.12) stating that implementation of the guidelines in BWRVIP-181, as modified to incorporate the resolution of RAIs as discussed in the draft SE, will provide an acceptable technical basis for the design criteria for use in the structural and design evaluation of steam dryer repairs and replacements, applied loads and load combinations, recommended ASME Code design guidance, fabrication and installation, and inspection of the steam dryers. The NMP2 steam dryer evaluation fully conforms to these guidelines and the NRC draft SE (Reference 7.12).

BWRVIP-182 requires that the technical basis for and benchmarking of any analytical or testing methodologies utilized in demonstrating steam dryer integrity be documented and submitted to the NRC for review and that specific acceptance criteria and values for key parameters to be used in the evaluation of steam dryers are defined. The plant specific analyses included in this attachment meet this requirement.

The NMP2 steam dryer evaluation followed the BWRVIP-182 guidance. The details of the NMP2 specific evaluations are provided in this attachment. Only the summary conclusions related to each step of the evaluation are summarized with reference to the specific details in the included or referenced reports.

2.0 Screening to Assess Potential for Main Steam Line (MSL) Acoustic Excitation at Power Uprate

Nine Mile Point Nuclear Station, LLC (NMPNS) performed the screening evaluation consistent with the analytical techniques described in BWRVIP-182 Section 3. Refined acoustic modeling of MSL standpipes was performed for the main steam safety relief valve (SRV) standpipe. The NMP2 steam lines are configured with no dead leg mounted SRVs or blind flange SRV locations. As required by BWVIP-182, all the MSL piping branch connections greater than two inches in diameter, such as drain lines or Reactor Core Isolation Cooling (RCIC) steam line connections, were screened and determined to be outside the Strouhal Number range for acoustic excitation (0.25 - 0.60) for the full range from 80% CLTP through 120% of EPU power level.

NOTE
Due to the nature of CDI Report 08-13P, the entire document is classified as proprietary. For this reason, a non-proprietary version of this report is not included in this attachment.

The NMP2 SRVs are mounted to a nozzle forging with a contoured entrance radius as depicted in CDI Report 08-13P (Attachment 13.3), Figure 4.5 (Reference 7.3). The inlet radius is included in the refined calculations for the onset velocity. CDI Report 08-13P documents the calculated NMP2 SRV standpipe excitation frequency as 224 Hz at normal operating conditions of pressure and temperature with an onset velocity of 262 ft/sec. The original licensed thermal power (OLTP) steam flow velocity is 143 ft/sec and the EPU steam flow is 177 ft/sec at 120% of OLTP steam flow velocity. This places the onset for SRV standpipe resonance at greater than 45% above the EPU power level, which meets the BWRVIP-182 screening for exclusion of SRV standpipe resonance.

Additional subscale test screening validation was performed. This test models all 4 steam lines from the steam dome to the turbine inlet and includes a scale model of the NMP2 steam dryer. The 1/8th scale testing is documented in CDI Report 08-13P. The testing confirms no single or double vortex onset in the CLTP to EPU steam flow range and confirms the excitation frequency of 219.5 Hz with an onset Mach number of 0.16, which is a 1.9% difference between calculated and the subscale testing. In addition, the plant data from MSL measurement confirm that the double vortex peak possible at 1/2 the Mach number of the single vortex (approximately 90% power) is not present in the data taken through CLTP conditions, which includes data taken at 69%, 88%, 90%, 94%, 97% and 100% CLTP. The conclusion is that the SRV standpipe double vortex is not a significant load for NMP2 at CLTP or EPU conditions.

3.0 Defining MSL Local Pressure Fluctuation Based on In-Plant Tests

NOTE
The non-proprietary version of SAI calculation NMP-26Q-302 is provided as Attachment 13.9. The non-proprietary version of CDI Report No. 08-08P is provided as Attachment 13.8.

The Structural Integrity Associates (SIA) calculation NMP-26Q-302 (Attachment 13.4) documents the strain gauge installation details, the background noise floor evaluation for the data acquisition system (DAS), and the data processing, including the raw data micro strain waterfall plots. The SIA analysis documents the data sets taken, including the electrical interference data sets taken with each power data set. The CDI loads definition report, CDI Report 08-08P (Attachment 13.2) describes the signal conditions applied to the strain gage data.

4.0 Steam Dryer Fluctuating Pressure Loading from In-Plant MSL Pressure Measurements

CDI Report 08-08P (Attachment 13.2) applies the ACM Rev. 4 to the NMP2 steam dryer and main steam line geometry. Strain gauge data obtained from the four main steam lines is used to define pressure levels on the NMP2 full-scale dryer at CLTP.

CDI Report 08-08P (Attachment 13.2), Table 5.1, defines the NMP2 loads including the bias and uncertainty for specified frequency intervals consistent with previously accepted bias and uncertainty for the CDI ACM Rev. 4 model. CDI report 08-08P, Revision 3, Table 3.2, identifies specific exclusion frequencies related to extraneous noise and MSL pipe vibration frequencies. This noise removal is consistent with previous applicants using CDI methods.

CDI Report 08-08P (Attachment 13.2) includes the coherence, Electrical Interference Check (EIC), and discrete frequency filtering, and defines the CLTP Power Spectral Density (PSD ($psid^2/Hz$)) versus frequency (Hz) for each MSL strain gauge location. The report concludes, based on the CLTP strain gauge data using the ACM Rev. 4 model: (a) The steam dryer maximum differential pressure loads, based on the validated Modified Bounding Pressure model, are less than 0.13 psid (CLTP). The maximum load is found on the lower corner of an inner hood opposite main steam line A; (b) Predicted loads on dryer components are largest for components nearest the main steam line inlets and decrease inward into the reactor vessel.

The scaling of the CLTP FIV pressure loading is performed based on velocity squared scaling for the full frequency range, because the NMP2 screening concluded that no resonance is predicted for EPU conditions. The NMP2 velocity squared scaling bump-up factor is 1.39. Application of the velocity squared scaling is consistent with the NRC RAI 182-2. The bump-up for EPU based on velocity squared is performed on the final stress analysis. Because the bump-up is constant throughout the frequency range, a redundant finite element analysis is eliminated.

5.0 Steam Dryer Structural Response and Stress Margin

CDI Report No. 08-26P (Reference 7.1), SIA Report No. 0801273.401 (Reference 7.5), and SIA Report No. 0800528.402 (Reference 7.6) define the steam dryer structural response and stress margin for design basis service levels A, B, C, and D. The FIV load has been added to the design basis loading per the guidance in BWRVIP-181.

The NMP2 specific reports provide the plant specific analysis results and plant specific basis for compliance with the requirements of BWRVIP-182, and is consistent with the recommendation of RG 1.20. In addition, the dryer stress analysis is consistent with NRC RAI-182-4a&b and RAI-182-5a for inclusion of supporting documentation of all known end-to-end biases and uncertainties; and RAI 182-5b for inclusion of documentation for evaluation of existing flaws and their impact on steam dryer operation at EPU.

NOTE
The non-proprietary version of CDI Report No. 08-26P is provided as Attachment 13.7.

The service level A stress analysis is documented in CDI Report No. 08-26P. The NMP2 steam dryer minimum alternating stress has been evaluated with the low flow noise included. The table of the minimum alternating stress ratio for both CLTP and EPU conditions is found in Table 9A and Table 9B of the attached CDI Report 09-26P. These analyses demonstrate that the minimum alternating stress ratio remains above 2.08 at all frequency shifts for the maximum EPU condition of 120% of OLTP. The sub-modeling locations are identified in Table 7 of CDI Report 09-26P.

In order to address a recent NRC concern, NMP2 performed a re-evaluation of dryer stresses utilizing a refinement of biases and uncertainties applied over the frequency interval of 60 to 100 Hertz. This evaluation confirmed that the existing NMP2 stress reports are valid, as the limiting alternating stress ratio remained greater than 2.0. Because the purpose of this evaluation was a validation of results, the existing stress reports have not been revised.

Steam Dryer Reinforcements:

The analysis determined that two components on the steam dryer require reinforcement of selected attachment welds to meet the 100% margin on alternating stress for the normal EPU operating condition. The components are the inner and middle hood end cover welds and the lifting rod upper brace to vane bank weld. The locations are detailed in Appendixes A and B of CDI Report 08-26P (Attachment 13.1). For the closure plates, reinforcement strips are added to stiffen the closure plates. Also, the top 18 inches of the welds connecting the closure plates to the vane banks and to the hoods are reinforced by adding a weld on the inner side of the closure plate. For the lifting rod braces, increasing the weld size from 1/4 in to 1/2 in meets the target stress ratio.

CDI Report 08-26P (Attachment 13.1) applied detailed submodeling to evaluate the reinforcement and establish the limiting alternating stress margin.

Steam Dryer Cracking:

The baseline inspections of the NMP2 steam dryer identified steam dryer cracking consistent with the BWR fleet operating history, as described in Section 2.4 of BWRVIP-139. The indications that require assessment relative to EPU service conditions are the indications located in the upper support ring, the drain channel to skirt vertical weld, and in the tie bar to hood weld heat affected zone (HAZ). Indications in the anti-rotation tack welds associated with the tie rod cam nut washers and the lifting lug have been identified as repair locations prior to EPU service.

SIA Report No. 0801273.401 (Reference 7.5) documents the flaw evaluation and vibration assessment performed consistent with BWRVIP-139 and BWRVIP-182 (including RAIs) for the observed indications at EPU service conditions. The evaluation concludes that these locations do not change the dynamic modal response of the dryer structure and therefore, the FEA model remains applicable to the current condition of the NMP2 steam dryer. The flaw evaluations for the indications in the ring and the indication in the dryer skirt adjacent to a drain channel remain well below the largest R ratio curve potential fatigue crack growth for austenitic stainless steel and therefore, essentially no fatigue crack growth is predicted for EPU operating conditions. Because the indications are characterized as intergranular stress corrosion cracking (IGSCC) they were also evaluated conservatively assuming further IGSCC growth using BWRVIP-14A methods. The IGSCC crack growth assumes an inspection interval of one cycle resulting in an insignificant change in the section thickness and the remaining ligament. The conclusion is that these components remain well within the required code safety factors for all service conditions, including the limiting upset and faulted conditions with the EPU FIV load included.

The tie bar IGSCC cracking is located in the HAZ of several tie bars. This cracking was originally identified during the initial BWRVIP-139 baseline inspection in 2004. The locations have been monitored in the 2006 and 2008 refuel outages with no measured crack growth. This cracking is located in the HAZ of the attachment weld, which is not typical of the industry experience related to fatigue of the tie bar locations. The tie bar attachment location is identified in CDI Report 08-26P (Attachment 13.1), and Table 9b of this report indicates an acceptable stress ratio in the as-welded condition. However, because of the industry experience at this location and the FEA conclusion that it is at one of the higher stress locations in the steam dryer, corrective measures for the IGSCC condition involving an overlay weld will be performed prior to EPU service.

6.0 Power Ascension Monitoring/Data Evaluation

BWRVIP-182, section 9, defines two approaches that can be taken to confirm that steam dryer stresses are within acceptable limits during power ascension: (1) Pre-established limit curves, or (2) Conduct stress analysis during power ascension.

The NMP2 steam dryer monitoring plan is to develop, prior to power ascension, limit curves generated from the CLTP strain gauge data. This approach is a lesson learned from the Hope Creek EPU power ascension where plant noise profiles and refurbished strain gauges impacted the limit curves, requiring the regeneration of the curves. In addition, the NMP2 plan is to implement reanalysis and produce new limit curves whenever it is felt that the Level 2 limit curve is being challenged, and implement real time power ascension stress analysis option at selected hold points. In this context, the limit curves provided in CDI Technical Note No. 09-17P (Attachment 13.10) are considered plant specific sample limit curves.

Based on the CDI Report 08-26P (Attachment 13.1) stress analysis and the CDI Report 08-08P (Attachment 13.2) loads, sample limit curves have been prepared with Level 1 and Level 2 criteria. The sample limit curves are included in CDI Technical Note No. 09-17P (Attachment 13.10). The planned NMP2 action to be taken at Level 1 and Level 2 is as follows: If Level 2 criteria are reached, reactor power ascension is to be suspended until an engineering evaluation concludes that further power ascension is justified. Should Level 1 be reached or exceeded, reactor power is returned to a previously acceptable power level that satisfies Level 2 criteria while an engineering evaluation is undertaken.

7.0 References

7.1 CDI Report No. 08-26P, "Stress Assessments of Nine Mile Point Unit 2 Steam Dryer," Rev. 1

7.2 CDI Report No. 08-08P, "Acoustic and Low Frequency Hydrodynamic Loads at CLTP Power Level on Nine Mile Point Unit 2 Steam Dryer to 250 Hz," Rev. 3

7.3 CDI Report No. 08-13P, "Flow-Induced Vibration in the Main Steam Lines at Nine Mile Point Unit 2 and Resulting Steam Dryer Loads," Rev. 1

7.4 SIA calculation NMP-26Q-302, "Nine Mile Point Unit 2 Main Steam Line Strain Gage Data Reduction," Rev. 0

7.5 SIA Report No. 0801273.401, "Flaw Evaluation and Vibration Assessment of the Nine Mile Point Unit 2 Steam Dryer for Extended Power Uprate Operating Conditions," Rev. 1

7.6 SIA Report No. 0800528.402, "Nine Mile Point Unit 2 Steam Dryer ASME Stress Analysis," Rev. 0

7.7 BWRVIP-14A, "Evaluation of Crack Growth in BWR Stainless Steel RPV Internals," September 2008

7.8 BWRVIP-139, "Steam Dryer Inspection and Flaw Evaluation Guidelines," April 2005

7.9 BWRVIP-181, "Steam Dryer Repair Design Criteria," November 2007

7.10 BWRVIP-182, "Guidance for Demonstration of Steam Dryer Integrity for Power Uprate," (including RAIs and responses), January 2008

7.11 Letter from S. L. Rosenberg (NRC) to R. Libra (BWRVIP) dated November 23, 2009, Draft Safety Evaluation for Boiling Water Reactor Vessel and Internals Project Topical Report 1016166, "BWR Vessel and Internals Project, Guidance for Demonstration of Steam Dryer Integrity for Power Uprate (BWRVIP-182)" (TAC No MD9427)

7.12 Letter from S. L. Rosenberg (NRC) to R. Libra (BWRVIP) dated October 22, 2009, "Draft Safety Evaluation (SE) for Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) Topical Report (TR) BWRVIP-181, "BWR Vessel and Internals Project, Steam Dryer Repair Design Criteria (BWRVIP-181)" (TAC No. MD8325)

7.13 December 2, 2009, BWRVIP letter 2009-338, BWRVIP Letter of Acceptance of the BWRVIP-181 Draft SE

7.14 CDI Technical Note No. 09-17P, "Limit Curve Analysis with ACM Rev. 4 for Power Ascension at Nine Mile Point Unit 2," Rev. 0