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a joint venture of
Constellation Energy
EDF

NINE MILE POINT
NUCLEAR STATION

July 30, 2010

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: The License Amendment Request for Extended Power Uprate Operation (TAC No. ME1476) – Reactor Systems and Health Physics RAI Responses, and Evaluation of Indications in the Steam Dryer Hood Support Attachment

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) E-mail from R. Guzman (NRC) to T. H. Darling (NMPNS), dated June 15, 2010, NMP2 EPU Follow-Up RAIs – Reactor Systems

(c) E-mail from R. Guzman (NRC) to T. H. Darling (NMPNS), dated July 14, 2010, NMP Unit 2 RAI – License Amendment Request for EPU Operation: Health Physics Review (TAC No. ME1476)

(d) Letter from S. Belcher (NMPNS) to Document Control Desk (NRC), dated June 30, 2010, Response to 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) – Steam Dryer and Probabilistic Risk Assessment

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits 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

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increase the power level authorized by OL Section 2.C.(1), Maximum Power Level, from 3467 megawatts-thermal (MWt) to 3988 MWt. By e-mails dated June 15, 2010 and July 14, 2010 (References b and c), the NRC staff provided requests for additional information (RAIs) from the Reactor Systems and Health Physics groups, respectively. The NMPNS responses to those RAIs are provided in Attachments 1 (Non-proprietary) and 9 (Proprietary), with supporting information in Attachments 3 and 10.

In addition, by letter dated June 30, 2010 (Reference d), NMPNS provided responses to NRC RAIs, including a commitment to submit its evaluation and conclusions regarding recently identified indications in the steam dryer hood support attachment by July 30, 2010. Attachment 2 provides a summary of the results of the Structural Integrity Associates, Inc (SIA) evaluation of recently identified indications in the steam dryer hood support attachment (Attachments 4 and 11) and the Continuum Dynamics, Inc. (CDI) design and stress evaluation of NMP2 steam dryer modifications for extended power uprate (EPU) operation (Attachments 5 and 12).

Attachments 9 through 12 are considered to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GE-Hitachi Nuclear Energy Americas LLC (GEH), Global Nuclear Fuel - Americas LLC (GNF-A) and CDI, NMPNS hereby makes application to withhold information from public disclosure in accordance with 10 CFR 2.390(b)(1). Affidavits from GEH, GNF-A, and CDI detailing the reason for the requests to withhold the proprietary information are provided in Attachments 6 through 8, respectively.

No new regulatory commitments are identified in this submittal.

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

Very truly yours,

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

I, Sam Belcher, being duly sworn, state that I am the Vice President-Nine Mile Point, 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.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 30th day of July, 2010.

WITNESS my Hand and Notarial Seal:

Tonya L Jones
Notary Public

My Commission Expires:

11/12/2010
Date

TONYA L. JONES
Notary Public in the State of New York
Oswego County Reg. No. 01JO6083354
My Commission Expires 11/12/2010

Attachments:

1. Responses to Requests for Additional Information Regarding License Amendment Request for Extended Power Uprate Operation (Non-proprietary)

2. Summary of the Structural Integrity Associates, Inc (SIA) Evaluation of Recently Identified Indications in the Steam Dryer Hood Support Attachment and the Continuum Dynamics, Inc. (CDI) Design and Stress Evaluation of NMP2 Steam Dryer Modifications for EPU Operation (Non-proprietary)

3. Global Nuclear Fuel - Americas LLC, MCNP01A Low Enriched UO_2 Pin Lattice in Water Critical Benchmark Evaluations Using ENDF/B-V Nuclear Cross-Section Data, Revision 1 (Non-proprietary)

4. Structural Integrity Associates, Inc, Flaw Evaluation of Indications in the Nine Mile Point Unit 2 Steam Dryer Vertical Support Plates Considering Extended Power Uprate Flow Induced Vibration Loading (Non-proprietary)

5. Continuum Dynamics, Inc., CDI Report No. 10-12NP, Design and Stress Evaluation of Nine Mile Point Unit 2 Steam Dryer Modifications for EPU Operation (Non-proprietary)

6. Affidavit Justifying Withholding Proprietary Information from GE-Hitachi Nuclear Energy Americas LLC

7. Affidavit Justifying Withholding Proprietary Information from Global Nuclear Fuel - Americas LLC

8. Affidavit Justifying Withholding Proprietary Information from Continuum Dynamics, Inc.

9. Responses to Requests for Additional Information Regarding License Amendment Request for Extended Power Uprate Operation (Proprietary)

10. Global Nuclear Fuel - Americas LLC, MCNP01A Low Enriched UO_2 Pin Lattice in Water Critical Benchmark Evaluations Using ENDF/B-V Nuclear Cross-Section Data, Revision 1 (Proprietary)

11. Structural Integrity Associates, Inc, Flaw Evaluation of Indications in the Nine Mile Point Unit 2 Steam Dryer Vertical Support Plates Considering Extended Power Uprate Flow Induced Vibration Loading (Proprietary)

12. Continuum Dynamics, Inc., CDI Report No. 10-12P, Design and Stress Evaluation of Nine Mile Point Unit 2 Steam Dryer Modifications for EPU Operation (Proprietary)

cc: NRC Regional Administrator, Region I
NRC Resident Inspector
NRC Project Manager
A. L. Peterson, NYSERDA (w/o Attachments 9 through 12)

ATTACHMENT 1

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

Certain information, considered proprietary by GE-Hitachi Nuclear Energy Americas LLC has been deleted from this Attachment. The deletions are identified by double square brackets.

Nine Mile Point Nuclear Station, LLC
July 30, 2010

By letter dated May 27, 2009, as supplemented on August 28, 2009, December 23, 2009 February 19, 2010, April 16, 2010, May 7, 2010, June 3, 2010, June 30, 2010, and July 9, 2010, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted for Nuclear Regulatory Commission (NRC) 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).

By e-mails dated June 15, 2010 and July 14, 2010, the NRC provided requests for additional information (RAIs) from the Reactor Systems and Health Physics groups, respectively. The NMPNS responses to those RAIs are provided in this Attachment. The NRC request is repeated (in italics), followed by the NMPNS response.

RAI HP-1

Provide an analysis demonstrating that there will be continued access to vital areas within the plant (consistent with NUREG 0737 item II.B.2) under EPU accident conditions. This analysis should include the full mission dose to each vital area necessary during the course of the accident.

NMPNS Response

RAI HP-1, Table 1 is a summary of the calculated maximum post-accident total travel doses for the worst case activity for the original licensed thermal power (OLTP) and for operation at extended power uprate (EPU) conditions. The EPU dose is 120% of the OLTP dose; this is based on a post accident dose rate increase attributable to the power level increase. This table demonstrates that these doses are less than the 10 CFR 50, Appendix A limits.

RAI HP-1, Table 1
Maximum Total Travel Dose (REM) for Worst Case Activity (Operator Dispatched from the Operations Support Center (OSC) to the North or South Auxiliary Bay)

	OLTP	EPU (OLTP Plus 20%)	10 CFR 50, App. A Limit
Gamma	2.76	3.31	5
Thyroid	1.84	2.21	30
Beta	0.28	0.34	30

RAI HP-1, Table 2 is a summary of the calculated post-accident doses for various tasks in vital access areas for the OLTP and for operation at EPU. The EPU dose is 120% of the OLTP dose; this is based on a post accident dose rate increase attributable to the power level increase. This table demonstrates that these doses are less than the 10 CFR 50, Appendix A limits. The doses for the Main Control Room and Relay and Computer Room and the Technical Support Center (TSC) were calculated for alternative source term at 4067 MWt, which bounds the OLTP and EPU power levels. Therefore, they are applicable for both power levels.

RAI HP-1, Table 2
Doses Associated with Work in Vital Access Areas

		Gamma	Thyroid	Beta	Gamma	Thyroid	Beta
10 CFR 50, Appendix. A Limit		**5**	**30**	**30**	**5**	**30**	**30**
Area	Task(s)	OLTP Dose			EPU Dose (OLTP Plus 20%)		
Main Control Room and Relay and Computer Room	Safe Shutdown(1)	The post-accident doses to personnel in the Main Control Room and Relay and Computer Room are calculated for alternative source term. The doses are less than 5 REM Total Effective Dose Equivalent (TEDE), which satisifies the requirements of 10 CFR 50.67.			The post-accident doses to personnel in the Main Control Room and Relay and Computer Room are calculated for alternative source term. These doses are applicable to EPU. The doses are less than 5 REM TEDE, which satisifies the requirements of 10 CFR 50.67.		
Health Physics / Counting Room	Sample Analysis	0.237(2)	186(2,7)	3.36(2)	0.284(2)	223.2(2,7)	4.03(2)

RAI HP-1, Table 2

Doses Associated with Work in Vital Access Areas

		Gamma	**Thyroid**	**Beta**	**Gamma**	**Thyroid**	**Beta**
10 CFR 50, Appendix. A Limit		**5**	**30**	**30**	**5**	**30**	**30**
Area	Task(s)	OLTP Dose			EPU Dose (OLTP Plus 20%)		
Radwaste Sample (PASS) Room and Unit 1 Chemistry Laboratory	Obtain and Analyze Dilute Reactor Coolant (RC) Samples	1.84[3, 4]	43.5[4,7]	0.859[4]	1.86[3, 4, 5]	52.2[4,7]	1.03[4]
	Obtain and Analyze Atmosphere Samples	2.07 [3,4]	33.9[4,7]	0.670[4]	2.26[3, 4, 5]	40.68[4,7]	0.804[4]
	Obtain and Analyze Gas in RC Samples	0.638[3,4]	85.2[4,7]	1.68[4]	0.634[3, 4, 5]	102. [4,7]	2.02[4]
	Obtain and Analyze Undilute RC Samples	4.41	54.1[4,7]	1.05[4]	4.45	64.9[4,7]	1.26[4]
Online Isotopic Monitors (Turbine Building)	Replace N_2 Supply Dewer	0.362	1.01	0.00113	0.434	1.21	0.00136
Online Isotopic Monitors (Main Stack)	Replace Large N_2 Supply Dewer and Refill Sample Cartridge Feed Hopper	2.44	16.8	0.361	2.93	20.2	0.433
	Manual Sampling	2.73[3]	32.4[7]	0.650	3.28[3]	38.9[7]	0.78
Radwaste Control Room	Turn Off Reactor Building Floor and Equipment Drains Pumps	0.596[4]	13.7[4]	0.210[4]	0.715[4]	16.4[4]	0.252[4]
	Service Emergency Response Facility (ERF) Computer	0.805[4]	26.4[4]	0.404[4]	0.966[4]	31.68[4,7]	0.485[4]
Technical Support Center	Continuous Occupancy To Provide Plant Management And Technical Support To Plant Operations Personnel.	The post-accident doses to personnel in the TSC are calculated for alternative source term for the EPU power. The doses are less than 5 REM TEDE.			The post-accident doses to personnel in the TSC are calculated for alternative source term for the EPU power. These doses are applicable to EPU. The doses are less than 5 REM TEDE.		

1. Continuous 30 Day occupancy
2. Maximum 8-hour shift
3. Whole body gamma doses
4. Includes doses traveling to and from the Operations Support Center
5. Use a multiplier of 1.009 for liquid reactor coolant sampling and analysis doses.
6. The gamma doses received by personnel performing vital post accident functions are all below the 5 REM limit.
7. Some of the thyroid doses received by personnel performing vital post-accident functions exceed the 30 REM limit. Therefore the use of breathing apparatus is required. The use of breathing apparatus will provide approximately a factor of 10 reduction in the thyroid doses.
8. The beta doses received by personnel performing vital post accident functions are all below the 30 REM limit.

RAI HP-2

Table 2.10-2 lists current annual dose at the NMP2 site boundary. Describe the basis for the "current" dose numbers listed. Verify that these doses are for the site boundary location with the maximum dose. Provide a rationale for why the combined dose from NMP Unit 1 and Fitzpatrick plants is less than one third of the NMP2 dose contribution.

NMPNS Response

The "current" dose numbers presented in the top portion of Table 2.10-2 of NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11) are from Table 12.4-1 of the NMP2 Updated Safety Analysis Report (USAR).

The doses presented in the top portion of Table 2.10-2 of NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11) are measured doses (from Nine Mile Point Unit 1 (NMP1) and J. A. Fitzpatrick (JAF) taken in 1985 prior to NMP2 operation) and calculated doses (from NMP2).

The NMP2 doses are calculated at the restricted area boundary at the lake shore. This is considered the location for the maximum dose.

These calculated doses are adjusted for EPU conditions (the waste liners and effluent are based on expected activity and are scaled by 20% and the N-16 contribution (skyshine from the turbine building) is scaled by 30%).

The dose contribution from NMP1 and JAF is smaller than the dose contribution from NMP2 because the NMP1 and JAF dose is a measured value while the NMP2 contribution to the dose is a dose calculation that includes additional conservatisms.

RAI HP-3

Provide a justification for using 2004 reporting data for offsite doses listed in Table 2.10. These values appear low compared to the 2007 & 2008 NMP2 Effluent Reports (both reported greater than 0.4 millirem (mrem) whole body dose vice the 0.02 mrem listed in Table 2.10).

NMPNS Response

The offsite doses from JAF, NMP1 and NMP2 that were reported from 2004 through 2009 to the NRC in the Radioactive Effluent Release Reports for CLTP and the corresponding estimated EPU doses, are summarized below. The EPU doses are estimated by scaling the reported doses (1.2 times for skin and maximum organ and 1.3 for whole body). This is conservative because the contributions from NMP1 and JAF are included, whereas the dose from only NMP2 will increase.

Dose Receptor	JAF, NMP1 and NMP2 Reported CLTP Doses Dose (mrem)							NMP2 EPU Dose, Including JAF and NMP1 Contributions (mrem)
	2004	2005	2006	2007	2008	2009	2004-09 Max	
Whole Body	0.180	1.51	2.01	1.52	0.492	2.76	2.76	3.59
Skin	0.0201	0.0131	0.00938	0.0169	0.00490	0.0202	0.0202	0.0242
Maximum Organ	0.112	0.155	0.0928	0.0932	0.0108	0.146	0.155	0.186

Note: The dose receptor data listed for calendar year 2004 in Table 2.10-2 of the Plant Uprate Safety Analysis Report was not correct. The data values were transposed. This information is corrected in the above table.

The above table supersedes the data regarding reported CLTP doses in Table 2.10-2 of NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11).

RAI HP-4

Page 2-366 of the Safety Analysis Report (SAR) indicates that moisture carryover in the steam is estimated to double. Discuss the potential impact of this increased carryover of non-volatile radionuclides (e.g., soluble iodines, and cesiums; and non-soluble activated corrosion and wear products) on dose rates around the balance-of-plant systems.

NMPNS Response

The moisture carryover used to calculate the main steam radiation sources for EPU is 0.1%. This is a factor of 2 increase in the predicted carryover for CLTP of 0.05%.

For the EPU, radiation sources during normal operation are expected to increase slightly; however, this increase is not significant. The change from 0.05% to 0.1% carryover, including soluble iodines, cesiums, non-soluble activated corrosion and wear products, will not have a significant effect on the doses in the plant because the shielding design is conservative. The shielding design is based on radiation sources that are conservative compared to those calculated for EPU based on 0.1% carryover.

As stated in the EPU SAR, shielding aspects of the plant were conservatively designed using design basis activity. The sum of the expected activated corrosion product activity and the fission product activity for EPU remains a small fraction (<12% for water, <15% for steam) of the original total design basis activity. Thus, the increase in radiation sources does not affect radiation zoning or shielding and plant radiation area procedural controls will compensate for any slight increase in normal radiation sources.

RAI HP-5

Page 2-367 of the SAR indicates that NMP2 has already implemented Hydrogen Water Chemistry (HWC) with noble metal injection. How long has NMP2 been using HWC? Verify that the current site boundary doses listed in Table 2.10 reflect the increased N-16 release from the reactor resulting from the associated hydrogen injection.

NMPNS Response

NMP2 implemented HWC and noble metal injection in calendar year 2000. Noble metal injection was developed to make the HWC more efficient and minimize the N-16 increases while providing Intergranular Stress Corrosion Cracking (IGSCC) mitigation. HWC plus noble metal injection for IGSCC mitigation requires on the order of 0.1 - 0.3 parts per million (ppm) H_2 in feedwater. This is below the level at which Main Steam Line Radiation Monitor (MSLRM) readings start increasing (as compared to no HWC values). Since the required H_2 concentration in the feedwater will not change due to EPU, no increase in N-16 release due to HWC following EPU implementation is expected.

After noble metal injection at NMP2, the plant experiences an expected period of elevated MSLRM readings:

- Noble metal application was initially performed during a plant shutdown and resulted in a less than 50% increase in MSLRM readings for approximately six months after startup.

- Since 2007, NMP2 has performed online noble metal injection, which mitigates the transient N-16 increase and duration previously seen. After the transient ends in approximately one month, MSLRM readings return to baseline levels. These baseline levels are comparable to the levels that existed prior to HWC and noble metal injection.

This transient is not impacted by EPU conditions, because, as discussed above, the hydrogen concentration will not change due to EPU.

The value of N-16 presented in the top portion of Table 2.10-2 of NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11) was calculated prior to the implementation of HWC and noble metal injection (see response to RAI HP-2). As established above, this value is not impacted by HWC and noble metal injection. In addition, the table in the response to RAI HP-3 provides the actual offsite doses for calendar years 2004 – 2009 (the doses include contributions from HWC and noble metal injection).

NMP2-SRXB-RAI-4

Who performed the current depletion and criticality analyses (GE, Holtec, NETCO, etc.)?

Previous NMPNS Response Provided in NMPNS Letter Dated June 3, 2010

GE/GNF performed the depletion and criticality analyses in 2004 as reflected in Section 2.8.6 of NEDC-33351P, Revision 0.

NMPNS Supplemental Information

The previous response refers to the technical approach used to address the impact of EPU on spent fuel pool (SFP) criticality relied upon in NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11). This evaluation concluded that EPU did not adversely impact SFP criticality consistent with the GEH Licensing Topical Report for Constant Pressure Power Uprate. It utilized a GEH analysis that was performed for the transition from GE11 to GE14 fuel in 2004 and is consistent with the EPU cores that are being designed with GE14 fuel. The evaluation concluded that there is no adverse impact on SFP criticality. However, the GEH criticality evaluation is not part of the current licensing basis for the NMP2 spent fuel pool. The Holtec criticality analysis referenced in Section 9.1.2 of the NMP2 Updated Safety Analysis Report (USAR) is the current analysis of record. The Holtec criticality analysis was retained as the analysis of record, because, at the transition to GE 14 fuel, the Holtec criticality analysis already addressed GE14 fuel types (as well as earlier fuel types utilized at NMPNS, which were shown to be bounded by GE14 fuel).

NMPNS will add the GEH criticality analysis utilized in the EPU License Application to Section 9.1.2 of the NMP2 USAR, during the design basis reconciliation for EPU implementation. This will include incorporation of the GEH analysis assumption of a maximum fuel pellet enrichment of 4.9 weight percent (w/o) U-235. This change does not have an impact on the current operation of the SFP, because there is no fuel stored in the NMP2 SFP that has an enrichment greater than 4.9 w/o. NMPNS will also retain the Holtec criticality analysis currently in the USAR as the analysis of record demonstrating that earlier fuel types irradiated under pre-EPU conditions are bounded by GE14 fuel.

Follow-up Reactor Systems RAI-1

Regarding NMPNS response to NMP2-SRXB-RAI-5: Please submit the MCNP01A validation document to support the completion of the NRC staff's safety evaluation and review.

NMPNS Response

The MCNP01A validation documentation evaluated 22 critical experiments. The report ("MCNP: Light Water Reactor Critical Benchmarks," GE Nuclear Energy, NEDO-32028) includes a description of the critical experiments and the corresponding MCNP input files. In preparing the response to this question, GE-Hitachi Nuclear Energy Americas LLC (GEH) / Global Nuclear Fuel - Americas LLC (GNF-A) validation of the response included calculations from a file that could not be retrieved by GEH/GNF-A. However, the MCNP01A validation documentation has been superseded by a more recent and thorough documentation ("MCNP01A Low Enriched UO2 Pin Lattice in Water Critical Benchmark Evaluations Using ENDF/B-V Nuclear Cross-Section Data") that incorporates 190 critical experiments. This later, more rigorous, study is now utilized to support the NMP2 EPU analysis (see the response to follow-up Reactor Systems RAI-2).

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MCNP01A, Low Enriched UO2 Pin Lattice in Water Critical Benchmark Evaluations Using ENDF/B-V Nuclear Cross-Section Data, Revision 1 is provided in Attachments 3 (Non-proprietary) and 10 (Proprietary).

Results of 190 Criticality Experiments								
#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
1	LEU-COMP-THERM-001	1	0.9998	0.0031	[[			
2	LEU-COMP-THERM-001	2	0.9998	0.0031				
3	LEU-COMP-THERM-001	3	0.9998	0.0031				
4	LEU-COMP-THERM-001	4	0.9998	0.0031				
5	LEU-COMP-THERM-001	5	0.9998	0.0031				
6	LEU-COMP-THERM-001	6	0.9998	0.0031				
7	LEU-COMP-THERM-001	7	0.9998	0.0031				
8	LEU-COMP-THERM-001	8	0.9998	0.0031				
9	LEU-COMP-THERM-002	1	0.9997	0.002				
10	LEU-COMP-THERM-002	2	0.9997	0.002				
11	LEU-COMP-THERM-002	3	0.9997	0.002				
12	LEU-COMP-THERM-002	4	0.9997	0.002				
13	LEU-COMP-THERM-002	5	0.9997	0.002				
14	LEU-COMP-THERM-006	1	1	0.002				
15	LEU-COMP-THERM-006	2	1	0.002				
16	LEU-COMP-THERM-006	3	1	0.002				
17	LEU-COMP-THERM-006	4	1	0.002				
18	LEU-COMP-THERM-006	5	1	0.002				
19	LEU-COMP-THERM-006	6	1	0.002				
20	LEU-COMP-THERM-006	7	1	0.002				
21	LEU-COMP-THERM-006	8	1	0.002				
22	LEU-COMP-THERM-006	9	1	0.002				
23	LEU-COMP-THERM-006	10	1	0.002				
24	LEU-COMP-THERM-006	11	1	0.002				
25	LEU-COMP-THERM-006	12	1	0.002				
26	LEU-COMP-THERM-006	13	1	0.002				
27	LEU-COMP-THERM-006	14	1	0.002				
28	LEU-COMP-THERM-006	15	1	0.002				
29	LEU-COMP-THERM-006	16	1	0.002				
30	LEU-COMP-THERM-006	17	1	0.002				

Results of 190 Criticality Experiments

#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
31	LEU-COMP-THERM-006	18	1	0.002				
32	LEU-COMP-THERM-009	1	1	0.0021				
33	LEU-COMP-THERM-009	2	1	0.0021				
34	LEU-COMP-THERM-009	3	1	0.0021				
35	LEU-COMP-THERM-009	4	1	0.0021				
36	LEU-COMP-THERM-009	5	1	0.0021				
37	LEU-COMP-THERM-009	6	1	0.0021				
38	LEU-COMP-THERM-009	7	1	0.0021				
39	LEU-COMP-THERM-009	8	1	0.0021				
40	LEU-COMP-THERM-009	9	1	0.0021				
41	LEU-COMP-THERM-009	24	1	0.0021				
42	LEU-COMP-THERM-009	25	1	0.0021				
43	LEU-COMP-THERM-009	26	1	0.0021				
44	LEU-COMP-THERM-009	27	1	0.0021				
45	LEU-COMP-THERM-016	1	1	0.0031				
46	LEU-COMP-THERM-016	2	1	0.0031				
47	LEU-COMP-THERM-016	3	1	0.0031				
48	LEU-COMP-THERM-016	4	1	0.0031				
49	LEU-COMP-THERM-016	5	1	0.0031				
50	LEU-COMP-THERM-016	6	1	0.0031				
51	LEU-COMP-THERM-016	7	1	0.0031				
52	LEU-COMP-THERM-016	8	1	0.0031				
53	LEU-COMP-THERM-016	9	1	0.0031				
54	LEU-COMP-THERM-016	10	1	0.0031				
55	LEU-COMP-THERM-016	11	1	0.0031				
56	LEU-COMP-THERM-016	12	1	0.0031				
57	LEU-COMP-THERM-016	13	1	0.0031				
58	LEU-COMP-THERM-016	14	1	0.0031				
59	LEU-COMP-THERM-016	18	1	0.0031				
60	LEU-COMP-THERM-016	28	1	0.0031				

Results of 190 Criticality Experiments								
#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
61	LEU-COMP-THERM-016	29	1	0.0031				
62	LEU-COMP-THERM-016	30	1	0.0031				
63	LEU-COMP-THERM-016	31	1	0.0031				
64	LEU-COMP-THERM-016	32	1	0.0031				
65	LEU-COMP-THERM-034	1	1	0.0047				
66	LEU-COMP-THERM-034	2	1	0.0047				
67	LEU-COMP-THERM-034	3	1	0.0039				
68	LEU-COMP-THERM-034	4	1	0.0039				
69	LEU-COMP-THERM-034	5	1	0.0039				
70	LEU-COMP-THERM-034	6	1	0.0039				
71	LEU-COMP-THERM-034	7	1	0.0039				
72	LEU-COMP-THERM-034	8	1	0.0039				
73	LEU-COMP-THERM-034	10	1	0.0048				
74	LEU-COMP-THERM-034	11	1	0.0048				
75	LEU-COMP-THERM-034	12	1	0.0048				
76	LEU-COMP-THERM-034	13	1	0.0048				
77	LEU-COMP-THERM-034	14	1	0.0043				
78	LEU-COMP-THERM-034	15	1	0.0043				
79	LEU-COMP-THERM-039	1	1	0.0014				
80	LEU-COMP-THERM-039	2	1	0.0014				
81	LEU-COMP-THERM-039	3	1	0.0014				
82	LEU-COMP-THERM-039	4	1	0.0014				
83	LEU-COMP-THERM-039	5	1	0.0014				
84	LEU-COMP-THERM-039	6	1	0.0014				
85	LEU-COMP-THERM-039	7	1	0.0014				
86	LEU-COMP-THERM-039	8	1	0.0014				
87	LEU-COMP-THERM-039	9	1	0.0014				
88	LEU-COMP-THERM-039	10	1	0.0014				
89	LEU-COMP-THERM-039	11	1	0.0014				
90	LEU-COMP-THERM-039	12	1	0.0014				

Results of 190 Criticality Experiments

#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
91	LEU-COMP-THERM-039	13	1	0.0014				
92	LEU-COMP-THERM-039	14	1	0.0014				
93	LEU-COMP-THERM-039	15	1	0.0014				
94	LEU-COMP-THERM-039	16	1	0.0014				
95	LEU-COMP-THERM-039	17	1	0.0014				
96	LEU-COMP-THERM-062	1	1	0.0016				
97	LEU-COMP-THERM-062	2	1	0.0016				
98	LEU-COMP-THERM-062	3	1	0.0016				
99	LEU-COMP-THERM-062	4	1	0.0016				
100	LEU-COMP-THERM-062	5	1	0.0016				
101	LEU-COMP-THERM-062	6	1	0.0016				
102	LEU-COMP-THERM-062	7	1	0.0016				
103	LEU-COMP-THERM-062	8	1	0.0016				
104	LEU-COMP-THERM-062	9	1	0.0016				
105	LEU-COMP-THERM-062	10	1	0.0016				
106	LEU-COMP-THERM-062	11	1	0.0016				
107	LEU-COMP-THERM-062	12	1	0.0016				
108	LEU-COMP-THERM-062	13	1	0.0016				
109	LEU-COMP-THERM-062	14	1	0.0016				
110	LEU-COMP-THERM-062	15	1	0.0016				
111	LEU-COMP-THERM-065	1	1	0.0014				
112	LEU-COMP-THERM-065	2	0.9999	0.0014				
113	LEU-COMP-THERM-065	3	0.9996	0.0015				
114	LEU-COMP-THERM-065	4	0.9997	0.0015				
115	LEU-COMP-THERM-065	5	1	0.0014				
116	LEU-COMP-THERM-065	6	0.9998	0.0014				
117	LEU-COMP-THERM-065	7	0.9991	0.0014				
118	LEU-COMP-THERM-065	8	1	0.0016				
119	LEU-COMP-THERM-065	9	1.0001	0.0015				
120	LEU-COMP-THERM-065	10	1.0002	0.0016				

Results of 190 Criticality Experiments

#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
121	LEU-COMP-THERM-065	11	1.0005	0.0016				
122	LEU-COMP-THERM-065	12	1	0.0017				
123	LEU-COMP-THERM-065	13	1.0001	0.0016				
124	LEU-COMP-THERM-065	14	1.0003	0.0016				
125	LEU-COMP-THERM-065	15	0.9994	0.0016				
126	LEU-COMP-THERM-065	16	0.9998	0.0017				
127	LEU-COMP-THERM-065	17	1.0003	0.0016				{3}]]
128	[[							
129								
130								
131								
132								
133								
134								
135								
136								
137								
138								
139								
140								
141								
142								
143								
144								
145								
146								
147								
148								
149								
150								

Results of 190 Criticality Experiments								
#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
151								
152								
153								
154								
155								
156								
157								
158								
159								
160								
161								
162								
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177								
178								
179								
180								

Results of 190 Criticality Experiments

#	Experiment	Expt. #	Benchmark Eigenvalue	Experimental Uncertainty	MCNP01A	MCNP01A Uncertainty	Bias*	Bias Uncertainty
181								
182								
183								
184								
185								
186								
187								
188								
189								
190								{3}]]

*Bias=Benchmark-MCNP01A

The following figure is a histogram (frequency distribution) of the results of all 190 benchmark eigenvalues treated as a single population sample. The analysis shows that the data passes the normality test (P-value=[[{3}]]). The fitted normal curve is shown in red color.

[[

{3}]]

The bias and bias uncertainty from the 190 critical experiments were calculated per NUREG/CR-6698 and are incorporated in the response to follow-up Reactor Systems RAI-2.

Follow-up Reactor Systems RAI-2

Regarding NMPNS response to NMP2-SRXB-RAI-5: Simply stating that "[f]ission products benchmarks were not available and thus were not included in the validation set," is not appropriate. Table 12 of NEDC-33374P, Rev. 3, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants," addresses the validation gaps associated with fission products and actinides. Address these validation gaps for NMP2.

NMPNS Response

To address the validation gaps associated with the extension of MCNP01A validation to include spent fuel bundles, the negative reactivity contribution of fission products to the cold, in-core peak reactivity statepoint of the spent fuel rack design basis lattice was determined. [[
{3}]], is a conservative uncertainty that will be applied to the spent fuel racks studies to cover the isotopic benchmarking validation gap. The table below incorporates the effect of the MCNP01A validation gap as well as the depletion uncertainty and the new MCNP01A bias and bias uncertainty.

The fuel depletion uncertainty and the benchmarking gap uncertainty were included in the roll-up as bias for additional conservatism.

[[

{3}]]

The above table supersedes the data provided in Table 2.8-12 of NEDC-33351, Revision 0, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (LAR Attachments 3 and 11).

Follow-up Reactor Systems RAI-3

Regarding NMPNS response to NMP2-SRXB-RAI-9: Table 12 of NEDC-33374P, Rev. 3, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants," included a rack sliding bias of approximately 1500 pcm. The NRC staff understands that the ESBWR criticality analysis also assumed a no leakage condition. Please provide a discussion explaining why the NMP2 analysis does not need to account for this bias.

NMPNS Response

The spent fuel storage racks at NMP2 have neutron poison (Boral) panels on the periphery. The storage rack model for NMP2 is based on an infinite array of storage cells loaded with the most reactive lattice analyzed, and does not incorporate any radial or axial leakage. Due to the presence of external Boral sheathing in the NMP2 storage racks, the model used bounds the rack-sliding configuration. Thus, no bias was added for this accident condition.

ATTACHMENT 2

SUMMARY OF THE STRUCTURAL INTEGRITY ASSOCIATES, INC (SIA) EVALUATION OF RECENTLY IDENTIFIED INDICATIONS IN THE STEAM DRYER HOOD SUPPORT ATTACHMENT AND THE CONTINUUM DYNAMICS, INC. (CDI) DESIGN AND STRESS EVALUATION OF NMP2 STEAM DRYER MODIFICATIONS FOR EPU OPERATION (NON-PROPRIETARY)

Nine Mile Point Nuclear Station, LLC
July 30, 2010

ATTACHMENT 2
SUMMARY OF THE STRUCTURAL INTEGRITY ASSOCIATES, INC (SIA) EVALUATION OF RECENTLY IDENTIFIED INDICATIONS IN THE STEAM DRYER HOOD SUPPORT ATTACHMENT AND THE CONTINUUM DYNAMICS, INC. (CDI) DESIGN AND STRESS EVALUATION OF NMP2 STEAM DRYER MODIFICATIONS FOR EPU OPERATION (NON-PROPRIETARY)

The following provides a summary of the evaluation of the recently identified indications in the steam dryer hood support attachment for the extended power uprate (EPU) conditions. This evaluation applies Revision 4.1 of the Continuum Dynamics Inc.'s (CDI's) Acoustic Circuit Model (ACM) stress analysis results discussed in the Nine Mile Point Nuclear Station, LLC (NMPNS) responses to Requests for Additional Information (RAI) NMP2-EMCB-SD-RAI-6 and NMP2-EMCB-SD-RAI-8 provided in the submittal dated June 30, 2010. The flaw evaluation is included as Attachments 4 and 11.

In addition, the attached CDI Report 10-12 (Attachments 5 and 12) provides an updated list of modifications required for the steam dryer to meet the 100% margin at EPU conditions. The modifications listed in CDI Report 10-12 supersede the previous list of modifications discussed in CDI Report 09-26 provided as Attachments 5 and 15 in the NMPNS Response to Requests for Additional Information dated December 23, 2009.

Summary:

The structural model predicts that the design weld geometry for the steam dryer hood support attachment has sufficient margin to the fatigue crack endurance limit such that crack initiation from flow induced vibration (FIV) is not predicted by current loads or by EPU loads. The steam dryer inspection performed during the recent refueling outage identified that the as-welded condition at the bottom of the hood supports shows evidence of rework, grinding and fit-up induced biased mean stresses that have created localized regions above the endurance limit. This is consistent with similar cracking at this location documented in BWRVIP-139-A, "Steam Dryer Inspection and Flaw Evaluation Guidelines." The BWRVIP-139-A operating experience (OE) discussion notes "indications are associated with local stress concentration points, but appear to grow only a limited length. The distribution of the crack locations and relatively consistent length of these cracks suggest that they are stable (not growing) and might have resulted from relieving residual fabrication stresses or stresses from initial thermal expansion of the steam dryer."

The cracking at the hood support locations was first detected in April 2010 with the first baseline inspection of this location since the dryer was placed in service in 1988. This steam dryer was operated at Original Licensed Thermal Power (OLTP) conditions for 6 years before Nine Mile Point Unit 2 (NMP2) implemented the 5% stretch power uprate. NMP2 has operated at 105% OLTP from 1995 through 2010. The flaw evaluation for this location at Current Licensed Thermal Power (CLTP) conditions predicts that the FIV related stresses are sufficient to propagate a flaw in the location identified in the outer hood locations. The flaw evaluation concludes that crack propagation of approximately 2.25 inches is predicted in a relatively short period of time (months) and that the loading is displacement controlled such that the loads drop significantly as the flaw grows. The maximum observed flaw of less than 2 inches after 22 years of operation (15 years at current 105% stretch power uprate steam flow rates) is consistent with the flaw evaluation and the BWRVIP-139-A assessment of similar OE.

The cracks on the middle and inner hoods are less than ≈ ½ inch in length and the cracking orientation is characteristic of the primary initiation mechanism as fabrication. The small cracks and relatively non-uniform orientation indicates non-FIV crack growth early in operation likely dominated by fabrication stresses or thermal loading as the crack initiation mechanism. The results of the shell model stress comparison discussed in CDI Report 10-12 (Attachments 5 and 12) demonstrate that the stress, after 2 shell elements are disconnected (approximately 4 inches), is reduced to one third that of the outer hood which is indicative of displacement-controlled stress.

Conclusions:

The potential crack growth is anticipated to be minor and not affect the integrity of the hood support to ensure margin for EPU service conditions. Repair is warranted to provide robust margin to further crack growth under EPU conditions consistent with the NRC mandated 100% margin to the onset of FIV cracking. As access to the steam dryer is required to complete the hood support repairs, NMPNS will also implement the group 4 repairs previously noted as non-mandatory in the NMPNS submittal dated June 30, 2010.

In the NMPNS submittal dated June 30, 2010, CDI Report 10-11P identified that with ACM Revision 4.1 loads, additional modifications to the steam dryer are required. It included scoping modifications that demonstrated that NMP2 can meet the NRC mandated 100% margin. The attached CDI Report 10-12 (Attachments 5 and 12) provides an updated list of modifications required for the steam dryer to meet the 100% margin at EPU conditions and associated detailed submodel results. The modifications listed in CDI Report 10-12 supersede the previous list of modifications discussed in CDI Report 09-26 provided as Attachments 5 and 15 in NMPNS Response to Requests for Additional Information dated December 23, 2009.