This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment (3).

Exelon Generation.

Christopher R. Costanzo
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NMP2L2556

September 15, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
Docket No. 50-410

Subject: Response to Request for Additional Information Nine Mile Point Nuclear Station License Amendment Request for Maximum Extended Load Line Limit Analysis Plus

References:
(1) Letter from P. Swift (NMPNS) to Document Control Desk (NRC), License Amendment Request Pursuant to 10 CFR 50.90: Maximum Extended Load Line Limit Analysis Plus, dated November 1, 2013 (ML13316B107)

(2) Letter from B. Vaidya (USNRC) to C. Costanzo (NMPNS), Nine Mile Point Nuclear Station, Unit No. 2 – Third Round of Request for Additional Information Regarding License Amendment Request Pursuant to 10 CFR 50.90: Maximum Extended Load Line Limit Analysis Plus (MELLLA+) (TAC No. MF3056), dated August 14, 2014 (ML14204A369)

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits supplemental information requested by the NRC in support of a previously submitted request for amendment to the Nine Mile Point Unit 2 (NMP2) Renewed Facility Operating License NPF-69. The initial request, dated November 1, 2013 (Reference 1), included a proposed expansion of the operating boundary to allow operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain and the use of the General Electric Hitachi Nuclear Energy (GEH) analysis code TRACG04.

The supplemental information, provided in Attachment (2) (non-proprietary) and Attachment (3) (proprietary) to this letter, responds to the request for additional information that was provided in a letter from the NRC Staff to NMPNS on August 14, 2014 (Reference 2).

A001
NRR

This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment (3).

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

This supplement information does not change the initial determination of "no significant hazards consideration" justified in the original amendment request, Reference (1). Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this supplemental information to the appropriate state representative.

This letter contains no new regulatory commitments.

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Acting Manager – Site Regulatory Assurance, at (315) 349-5245.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of September, 2014.

Sincerely,

Christopher R Costanzo

Christopher R. Costanzo

CRC/KJK/STD

Attachments:

(1) Affidavits from GE-Hitachi Nuclear Energy Americas LLC
(2) Response to NRC Request for Additional Information (Non-Proprietary)
(3) Response to NRC Request for Additional Information (Proprietary)

cc: Regional Administrator, Region I, USNRC
Project Manager, USNRC
Resident Inspector, USNRC
A. L. Peterson, NYSERDA

ATTACHMENT 1

AFFIDAVITS FROM

GE-HITACHI NUCLEAR ENERGY AMERICAS LLC

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Linda C. Dolan, state as follows:

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

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, GE-PPO-1GYEF-KG1-740, "GEH Responses to NMP2 MELLLA+ SRXB(2) RAIs 1, 2, 3, 8-14, 18, 20, 22, & 23," dated August 29, 2014. The GEH proprietary information in Enclosure 1, which is entitled "Responses to SRXB(2) RAIs in Support of NMP2 MELLLA+ LAR," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and large objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

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

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of the Maximum Extended Load Line Limit Analysis Plus analysis for a GEH Boiling Water Reactor ("BWR"). The analysis utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of Maximum Extended Load Line Limit Analysis Plus for a GEH BWR.

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

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

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

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

I declare under penalty of perjury that the foregoing affidavit it true and correct.

Executed on this 29th day of August 2014.

Linda C. Dolan

Linda C. Dolan
Manager, Regulatory Compliance
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
Wilmington, NC 28401

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Peter M. Yandow, state as follows:

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

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, GE-PPO-1GYEF-KG1-741, "GEH Responses to NMP2 MELLLA+ SRXB(2) RAIs 4, 6, 15, 16, & 19," dated September 4, 2014. The GEH proprietary information in Enclosure 1, which is entitled "Responses to SRXB(2) RAIs in Support of NMP2 MELLLA+ LAR," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and large objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

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

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of the Maximum Extended Load Line Limit Analysis Plus analysis for a GEH Boiling Water Reactor ("BWR"). The analysis utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of Maximum Extended Load Line Limit Analysis Plus for a GEH BWR.

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

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

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

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

I declare under penalty of perjury that the foregoing affidavit it true and correct.

Executed on this 3rd day of September 2014.

Peter M. Yandow

Peter M. Yandow
Vice President, NPP/Services Licensing
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ATTACHMENT 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

(NON-PROPRIETARY)

This attachment provides supplemental information from two GEH letters.

Enclosure 1 to GEH letter GE-PPO-1GYEF-KGI-740, "GEH Response to NMP2 MELLLA+ SRXB (2) RAIs 1, 2, 3, 8-14, 18, 20, 22 & 23," dated August 29, 2014.

Enclosure 1 to GEH letter GE-PPO-1GYEF-KGI-741, "GEH Response to NMP2 MELLLA+ SRXB (2) RAIs 4, 6, 15, 16, & 19," dated September 4, 2014.

The affidavits detailing the reasons for the request to withhold the proprietary information are provided in Attachment 1.

By letter dated November 1, 2013, Nine Mile Point Nuclear Station, LLC (NMPNS) requested NRC approval to implement a proposed expansion of the operating boundary to allow operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain and the use of the General Electric Hitachi Nuclear Energy (GEH) analysis code TRACG04. This Attachment provides supplemental information that responds to the NRC's request for additional information that was provided in a letter to NMPNS dated August 14, 2014. Each individual NRC question is repeated (in italics), followed by the NMPNS response.

SRXB(2)-1.0: POWER DENSITY> 50 MWT/MLBM/HR

The Methods safety evaluation report (SER) states, "Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased." The power distribution root mean squared, RMS, data provided to support the Methods SER (SER Figure 3-4) ranged from [[ ]], and an extrapolation to [50 MWt/Mlbm/hr] was allowed based on the Safety Limit Minimum Critical Power Ratio (SLMCPR) adders.

The statement in Section 2.2.5 of the Safety Analysis Report (SAR): "The incorporation of this limitation duplicates the intent of the Methods SER..." does not seem to be justified by the SER language and intent.

Since NMP2 power density is 51.86 MWt/Mlbm/hr (>50), provide a power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.

Response SRXB(2)-1.0

The 50 MWt/Mlbm/hr may be exceeded as long as additional conservatism is applied to the cycle-specific SLMCPR. The requested action to justify that the neutronic methods axial and nodal power distribution uncertainties have not increased cannot be adequately performed until measured data is available.

The additional conservatism is included by incorporation of the +0.02 SLMCPR adder specified by the Methods Licensing Topical Report (LTR) Safety Evaluation Report (SER) Limitation and Condition 9.5 (Reference 1-1) for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) plants with planned operation above a Power-to-Flow Ratio (P/F) of 42 MWt/Mlbm/hr.

The SLMCPR evaluation specified by the MELLLA+ LTR SER Limitation and Condition 12.6 (Reference 1-2) specifically requires an evaluation of the Two Loop Operation (TLO) SLMCPR using the highest NRC-approved core flow uncertainties (i.e., Single Loop Operation (SLO) uncertainties) at the off-rated condition that has the highest P/F in the MELLLA+ operating domain.

Therefore, additional conservatism and justification that the power distribution uncertainties are acceptable is based on inclusion of the following:

- +0.02 SLMPCR adder for P/F > 42 MWt/Mlbm/hr,
- TLO SLMCPR evaluation at statepoint M (P/F = 51.86), and
- TLO SLMCPR evaluation with SLO uncertainties.

SLO is not allowed in the MELLLA+ operating domain, but the SLO uncertainties are used to evaluate the TLO SLMCPR for required statepoints in the MELLLA+ operating domain. The SLO SLMCPR evaluation is lower than the TLO SLMCPR due to the off-rated evaluation with SLO uncertainties. TLO SLMCPR and SLO SLMCPR are requested to be set equivalent in the technical specifications to avoid a SLO SLMCPR lower than a TLO SLMCPR.

See Figure 1-1 for a graphical depiction of the SLMCPR evaluation statepoints.

Also shown in the figure is the Traversing Incore Probe (TIP) exit 5% bypass voiding curve which represents the power to flow condition which is expected to have hot channel bypass voiding of 5% at the highest TIP elevation. It is recommended that TIP data gathered to the left of this line not be utilized for Local Power Range Monitor (LPRM) adaption. Because statepoint M (P/F = 51.86 MWt/Mlbm/hr) is to the left of the TIP exit 5% bypass voiding curve, the TIP data captured near this region may be utilized for power distribution uncertainties assessments, but should not be utilized from LPRM adaption.

The expected duration of operation in the region around statepoint M is limited by the expected duration of a xenon transient.

References

1-1. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

1-2. GE Hitachi Nuclear Energy, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," NEDC-33006P-A, Revision 3, June 2009.

1-3. Letter from James F. Harrison (GEH) to NRC, "Implementation of Methods Limitations - NEDC-33173P," MFN 08-693, September 18, 2008.

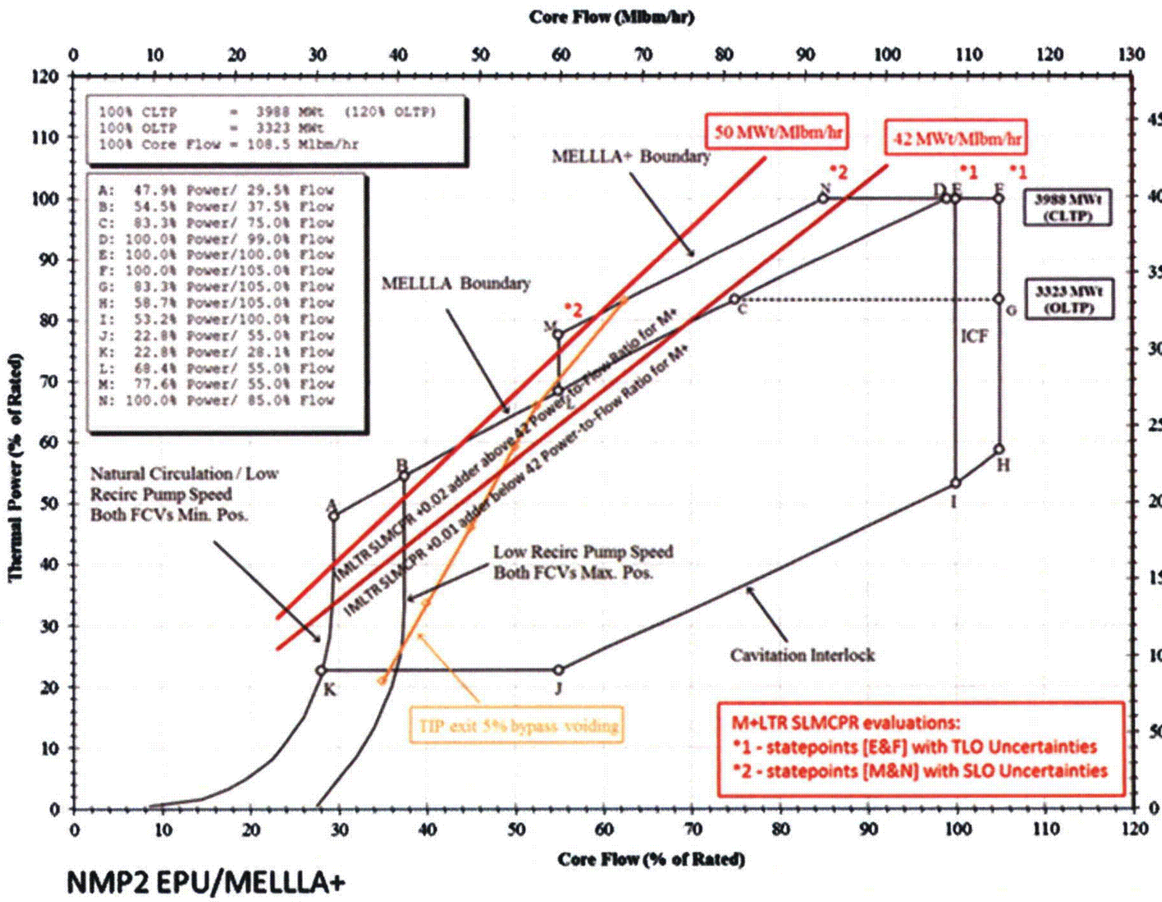


Figure 1-1. Graphical Depiction of the SLMCPR Evaluation Statepoints

SRXB(2)-2.0: SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (SLMCPR) ADDERS

Section 2.2.1 "Safety Limit Minimum Critical Power Ratio" states that "a +0.02 SLMCPR adder will be added to the cycle-specific SLMCPR."

1. *Provide a list of SLMCPR adders in MELLLA+ with respect to Original Licensed Thermal Power (OLTP) conditions.*
2. *Specify which adders are part of the Extended Power Uprate (EPU), and which are MELLLA+ specific.*
3. *In addition, the Methods SER specifies a SLMCPR adder of 0.03. Please explain the difference between the 0.03 and 0.02 values.*

Response *SRXB(2)-2.0*

Table 2-1 specifies the applicable SLMCPR adders defined by the Methods Licensing Topical Report (LTR) Safety Evaluation Report (SER) (Reference 2-1) for operation with Extended Power Uprate (EPU)/Maximum Extended Load Line Limit Analysis (MELLLA) (Limitation and Conditions 9.4) and EPU/MELLLA+ (Limitation and Conditions 9.5).

Figure 2-1 shows a graphical depiction of the SLMCPR evaluation statepoints and the Methods LTR penalties applied based on Power-to-Flow Ratio (P/F) value. Statepoints M&N [*2] in Figure 2-1 apply a +0.02 SLMCPR adder. Statepoints E&F [*1] in Figure 2-1 apply a +0.01 SLMCPR adder. Statepoints N, E, & F are evaluated at the rated power condition of 3988 MWth (100% CLTP or 120% OLTP). Statepoint M is evaluated at the off-rated power condition of 3095.5 MWth (77.6% CLTP or 93.2% OLTP).

The most limiting SLMCPR of the four evaluation statepoints at Beginning of Cycle (BOC)/Middle of Cycle (MOC)/ End of Cycle (EOC) exposure conditions will be used to set the technical specification SLMCPR. The Two Loop Operation (TLO) SLMCPR and Single Loop Operation (SLO) SLMCPR is requested to be set equivalent in the technical specifications to avoid a SLO SLMCPR lower than a TLO SLMCPR.

NMP2 has elected to conservatively apply a +0.02 SLMPCR adder to all evaluation points as noted by the statement in the M+SAR that "a +0.02 SLMCPR adder will be added to the cycle-specific SLMCPR." The NMP2 Cycle 15 Tech Specification change letter (Reference 2-2) Table 3 notes the application of this adder to each evaluation statepoint shown in Cycle 15.

The Methods LTR SER Limitation and Condition 9.4 (SLMCPR 1) and Limitations and Condition 9.5 (SLMCPR 2) were revised in Revision 4 of NEDC-33173P-A (Reference 2-1) with SER approval shown in Appendix I on Page I-22.

The 0.03 adder was eliminated in Revision 4 of NEDC-33173P-A. This is the revision of the Methods LTR that is utilized in Cycle 15 for NMP2.

References

2-1. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

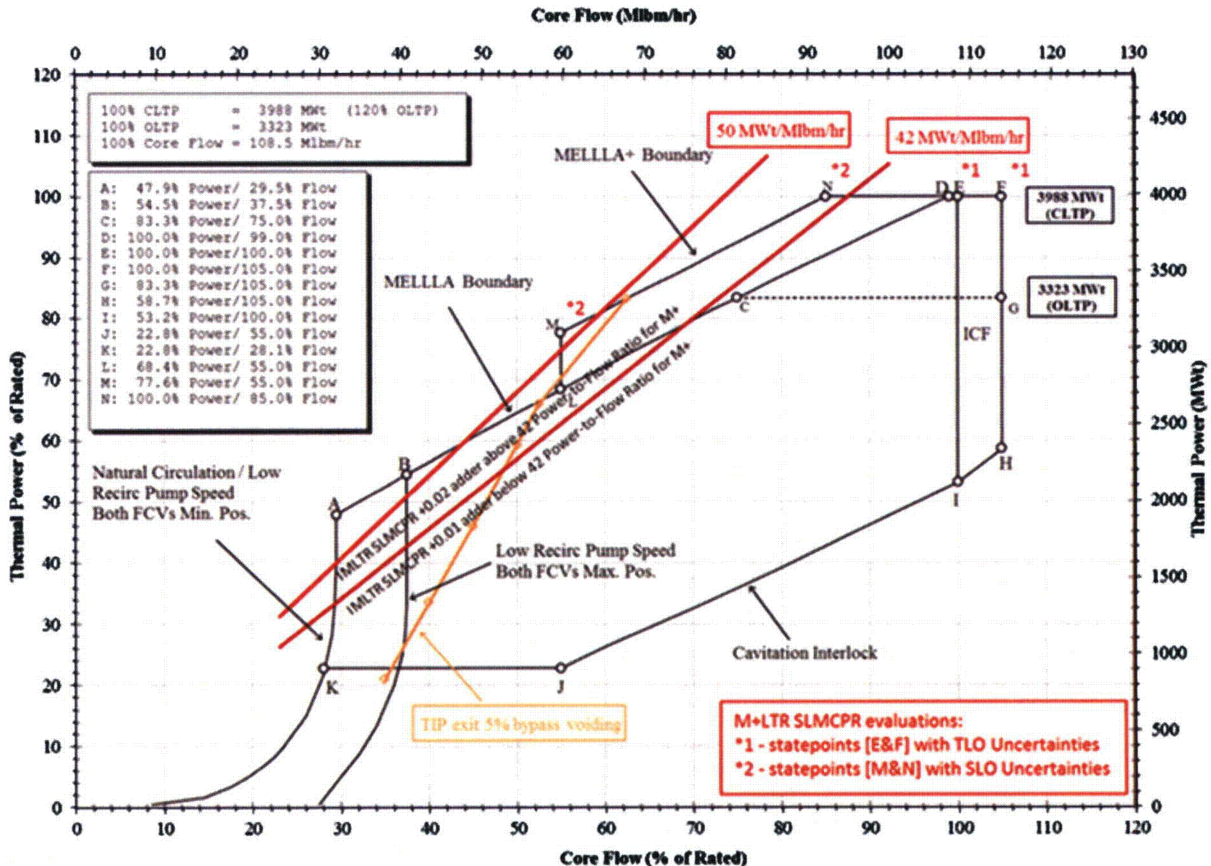
2-2. Global Nuclear Fuel, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Nine Mile Point 2 Cycle 15," GNF-0000-0156-7490-R0-P, Revision 0, August 2013.

Table 2-1. Applicable SLMCPR Adders

Licensing Basis	Methods LTR SER L&C	Cycle 14[1] TLO & SLO SLMCPR Adder	Cycle 15[2] TLO & SLO SLMCPR Adder
EPU / MELLLA (PUSAR)	9.4	0.02	0.00
EPU / MELLLA+ (M+SAR)	9.5	0.03	N/A
EPU / MELLLA+ with ≤ 42 MWt/Mlbm/hr (M+SAR)	9.5	N/A	0.01
EPU / MELLLA+ with > 42 MWt/Mlbm/hr (M+SAR)	9.5	N/A	0.02

1 The following adders applied at the start of Cycle14 based on NEDC-33173P-A Revision 3, EPU/MELLLA+ was not approved for Cycle 14 operation.

2 The following adders apply at the start of Cycle15 based on NEDC-33173P-A Revision 4.



NMP2 EPU/MELLLA+

Figure 2-1. Graphical Depiction of the SLMCPR Evaluation Statepoints

SRXB(2)-3.0: VOID FRACTION

Figures 2-2 through 2-5 of the SAR show an unusual behavior towards the end of cycle in NMP2 (significant increase in flow and reduction in void past 16.5 GWD/ST).

1. *Is this behavior caused by an end of cycle (EOC) stretch with increased core flow (ICF)?*

2. *The hot power bundle flow (Fig 2-2) increases by -25% whereas the increased core flow has a maximum value of 5% (Fig 1-1). Please explain the unusual NMP2 behavior in Figs 2-2 through 2-5.*

Response SRXB(2)-3.0

1. The behavior is from elevated core flows on the approach to end of cycle. The increase in core flow is from 85% to 105%, effectively 20% change in core flow. The behavior is not unusual. The comparison plant data is from the Methods Licensing Topical Report (LTR) (Reference 3-1). Several of the other plants data on the figures show similar trends. The other plant tracking data shows nominal core flow conditions as operated, not necessarily minimum core flow conditions at rated power as shown for NMP2 Maximum Extended Load Line Limit Analysis Plus (MELLLA+).

2. The last four data points shown in the MELLLA+ SAR Figure 2-1 through Figure 2-6 are at higher core flow conditions. The earlier data points with cycle exposures between 0.00 GWd/ST and 16.42 GWd/ST are at rated power with 85% core flow. The fourth to the last data point is at rated power with 92.5% core flow. The third to the last data point is at rated power and 100% core flow. The second to the last data point is at rated power and 105% core flow (Increased Core Flow (ICF) cycle extension). The last data point is at de-rated power (power coastdown cycle extension) and 105% core flow.

References

3-1. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

SRXB(2)-4.0: BACKUP STABILITY SOLUTION

Section 2.4.3 Backup Stability Protection (BSP) describes that the Detect and Suppress Solution - Confirmation Density (DSS-CD) licensing topical report (LTR) provides two options: (1) BSP manual regions and (2) BSP implemented with average power range monitor (APRM) flow bias scram. This section of the NMP2 SAR appears to be a summary of the DSS-CD LTR, but it is not clear which of the two options will be implemented by NMP2.

1. *Which option will NMP2 use for the first MELLLA+ cycle?*

2. *Provide the BSP regions for the NMP2 equilibrium cycle.*

Response SRXB(2)-4.0

1. Both Backup Stability Protection (BSP) options are used for the first Maximum Extended Load Line Limit Analysis Plus (MELLLA+) cycle. As described in Section 7.5.2 and in the Technical Specification (TS) changes documented in the approved Detect and Suppress Solution - Confirmation Density (DSS-CD) Licensing Topical Report (LTR) NEDC-33075P-A, Revision 8 (Reference 4-1), if the DSS-CD solution is inoperable, the Automated BSP option requires that the licensee implement the Automated BSP scram option within 12 hours. During this time Manual BSP is used as it takes some time to switch from DSS-CD to the automated BSP protection. If the Automated BSP option cannot be implemented, the TS require the licensee to implement the Manual BSP Region and BSP Boundary. These TS actions are defined for Nine Mile Point Unit 2 (NMP2) as actions F and J; see Reference 4-2, the May 14, 2014 updated submittal with responses to RAI STSB-1 and RAI STSB-2.

2. Sample BSP Region demonstration analyses have been performed for the NMP2 equilibrium cycle. The Nominal Feedwater Temperature (NFWT) results of the Manual BSP Region I (Scram), Manual BSP Region II (Controlled Entry), and BSP Boundary are shown in Figure 4-1.

[[

]]

Figure 4-1 NMP2 MELLLA+ Equilibrium Cycle Sample BSP Regions for NFWT

References

4-1 GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

4-2 Letter from J. Stanley (NMPNS) to Document Control Desk (USNRC), "License Amendment Request Pursuant to 10 CFR 50.90: Maximum Extended Load Line Limit Analysis Plus– Response to RAI STSB-1 and RAI STSB-2," dated May 14, 2014.

SRXB(2)-5.0: REACTOR CORE ISOLATION COOLING (RCIC)

Section 3.9.3 "[RCIC] Net Positive Suction Head" states that "For ATWS (Section 9.3) and fire protection (Section 6.7), operation of the RCIC system at suppression pool temperatures greater than the operational limit may be accomplished by using the condensate storage tank (CST) volume as the source of water"

1. *Is the CST available for RCIC even under containment isolation conditions?*

2. *If the suppression pool temperature reaches the operational limit, what indication/training does the operator have to switch from suppression pool to CST inlet?*

Response SRXB(2)-5.0

1. The RCIC system suction is normally aligned to the CST. The CST suction valve does not automatically close on any Group 1 to 12 containment isolation signals. The CST suction valve automatically closes only when CST tank level is below the low level trip setpoint (after the suppression pool suction valve is full open). If the RCIC system is operated with suction from the suppression pool, the RCIC system must be shutdown prior to returning its suction to the CST.

2. The RCIC system suction is normally aligned to the CST. With a large CST inventory, it is unlikely that RCIC system suction will need to be aligned to the suppression pool during an ATWS.

   Emergency Operating Procedures (EOPs) for Reactor Pressure Vessel (RPV) Controls, Steam Cooling, and Failure to Scram have an explicit operation limit of 140°F for the RCIC system. Suppression pool temperature indication is available in the control room. Since the operators are trained to the EOPs, RCIC suction will be aligned to the CST prior to exceeding the 140°F limit. This action requires the RCIC system to be shutdown prior to suction switch-over.

   In the event that the RCIC system is aligned to the suppression pool, the Net Positive Suction Head Available (NPSHA) = 28.46 ft at the RCIC system design specification maximum operating temperature of 170°F with the Net Positive Suction Head Required (NPSHR) = 23 ft. The maximum ATWS suppression pool temperature is 160°F.

SRXB(2)-6.0: ANTICIPATED OPERATIONAL OCCURANCE (AOO) IMPACT OF FLOW

On a separate MELLLA+ submittal, data was provided to justify that AOOs have smaller change in minimum critical power ratio (ΔMCPR) at 80% core flow than at 105% core flow. The argument presented in the past is a shift in power towards the bottom as the voids increase for the 80% flow case, which results in increased control rod performance that offsets the higher void reactivity coefficients at higher void levels.

NMP2 uses a core flow window from 85% to 105%. Provide the initial axial power shapes for the events in Table 9-1 of the SAR at 85% and 105% flow.

Response SRXB(2)-6.0

The initial axial power shapes for the Generator Load Rejection Without Bypass (LRNBP), Turbine Trip Without Bypass (TTNBP), and Feedwater Controller Failure (Maximum Demand) (FWCF) events documented in Table 9-1 are provided in the following figure for the 85% flow and 105% flow cases.

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Node numbers correspond to axial divisions of the active core with Node 1 at the bottom of the active core.

SRXB(2)-7.0: BI-STABLE FLOW

Is NMP2 susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

Response SRXB(2)-7.0

NMP2 is susceptible to bi-stable flow in the recirculation piping. The current 100% EPU power MELLLA boundary requires core flows between 99% and 105% rated core flow. To achieve this core flow, NMP2 requires recirculation drive flows between 44,000 gallons per minute (gpm) and 46,000 gpm, which results in the flow control valve in the 90 to 100% position. At NMP2 the magnitude of the bi-stable flow condition is reduced as the recirculation flow control valve position is increased. Thus, no maximum recirculation flow limit is required to ensure thermal power remains within the 100% EPU rated thermal power averaging time constant.

NMP2 did not have a flow limitation related to bi-stable flow when operating at pre-EPU licensed thermal power along the MELLLA boundary with core flows down to 80% rated core flow. The MELLLA+ testing will monitor the recirculation flow control and the bi-stable flow magnitude when operating at 100% power with core flows from 85% to 99%. The expectation is that the previous bi-stable flow magnitude observed during MELLLA operations at pre-EPU licensed thermal power will remain the same and that the associated thermal power changes will remain acceptable. Routine monitoring during normal operations will determine if the bi-stable flow will require reactor engineering to make slight adjustments in the rod line. Specific high or low limitations on recirculation flow are not anticipated.

SRXB(2)-8.0: PLANT DESIGN PARAMETERS

1. *Provide plant design parameters relevant to the ATWS calculations in Section 9 of the SAR. Specifically: turbine bypass capacity, sources of high-pressure injection and their operability issues (e.g., steam is lost after isolation...), sources of low-pressure injection and their operability issues (e.g. CST pumps...).*

2. *Provide vessel component elevations in units comparable to the ones used for water level in the Section 9 figures (include separators, feedwater spargers, nominal level, level setpoints for actuations, top of active fuel ...).*

Response SRXB(2)-8.0

1. The turbine bypass capacity is 18.5%.

   The source of high pressure injection credited in the analyses is the feedwater system. During an Anticipated Transient Without Scram Instability (ATWSI)-Turbine Trip With Bypass (TTWBP), there is an automatic feedwater runback, but feedwater is still available. The Reactor Core Isolation Cooling (RCIC) is also a high-pressure makeup system. RCIC is credited when feedwater is unavailable such as in the Main Steam Isolation Valve Closure (MSIVC) and Pressure Regulator Failure-Open (PRFO) analyses. No other high-pressure injection system is credited in the ATWS analyses.

   In the ODYN MSIVC and PRFO analyses, additional flow is included to control reactor water level to TAF + 5 feet to avoid depressurization as required per the ODYN analysis.

   Because depressurization does not occur in the ATWS calculations, low pressure injection systems are not credited in the analyses.

   One of the limitations of the feedwater system following a turbine trip is the loss of feedwater heating. This loss of heating is included in the ATWSI analysis.

2. All level readings in the figures are in units of inches in relation to the separator skirt elevation. This is 13.085 meters (515.16 inches) above vessel zero. For comparison, the following are other elevations in the same units in descending order:

• Upper tap	15.202 m (598.5 in) (83.3 inches above separator skirt)
• High Water Level (L8)	14.945 m (588.4 in) (73.2 inches above separator skirt)
• Nominal level	14.338 m (564.5 in) (49.3 inches above separator skirt)
• Low Water Level (L3)	13.386 m (527.0 in) (11.8 inches above separator skirt)
• Narrow Range tap	13.119 m (516.5 in) (1.3 inches above separator skirt)
• Separator Skirt elevation	13.085 m (515.2 in)
• Feedwater Nozzle	12.602 m (496.1 in) (-19.1 inches above separator skirt)
• Top of Active Fuel	9.304 m (366.3 in) (-148.9 inches above separator skirt)
• Wide Range tap	9.283 m (365.5 in) (-149.7 inches above separator skirt)

SRXB(2)-9.0: ATWS SEQUENCE OF EVENTS

Provide tables of the assumed sequence of events for the ODYN computer code licensing calculation, the ATWS/Stability calculation, and the ATWS/Stability/Reactor Pump Trip (RPT) calculation.

Response SRXB(2)-9.0

Sequence-of-Events tables are included as Tables 9-1 through 9-5. Tables 9-1 through 9-3 are ODYN licensing events, Table 9-4 is the Anticipated Transient Without Scram Instability (ATWSI)-Turbine Trip event, and Table 9-5 is the ATWSI–RPT event.

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SRXB(2)-10.0: ATWS WATER LEVEL STRATEGY

1. *Provide a detailed description of what water level control strategy (with emphasis on timing) was used for each ATWS calculation.*

2. *Describe the sources of water used to control the level. For the equipment used, describe automated actions (i.e., loss of extraction steam), and assumptions about operability (i.e., residual steam volume, if used) after the Main Steam Isolation Valve isolation occurs.*

Response SRXB(2)-10.0

1. The water level control strategy is basically reducing water level during an ATWS event below the feedwater spargers in order to decrease subcooling and power. This is consistent with SRP 15.8. The level below the feedwater spargers to which water level is reduced is not as important as the timing for when this occurs. The following tabulates the timing for several events:

    a. Anticipated Transient Without Scram (ATWS) – Main Steam Isolation Valve Closure (MSIVC) – Automatic feedwater runback is initiated in 38 seconds (33.0 second delay following the high pressure ATWS Recirculation Pump Trip (RPT) which is conservatively assumed to occur at 5.0 seconds).

    b. ATWS-Pressure Regulator Failure-Open (PRFO) – Automatic feedwater runback is initiated in 56 seconds (33.0 second delay following the high pressure ATWS RPT which occurs at 23 seconds).

    c. ATWSI-Turbine Trip With Bypass (TTWBP) – Automatic feedwater runback is initiated in 33.6 seconds (33.0 second delay following the high pressure ATWS RPT which occurs at 0.6 seconds).

    d. ATWSI-RPT – Water level reduction in 270 seconds.

2. Feedwater is used for all events. If feedwater is unavailable, RCIC is used. The RCIC flow enthalpy is 88 Btu/lbm.

    a. ATWSI-TTWBP – Feedwater is used during this event. After the turbine trip, feedwater heating is lost, so feedwater temperatures drop to the lowest expected main condenser temperature.

    b. ATWSI-RPT – Feedwater is used during this event. As a consequence of the reduction in steam flow, feedwater temperature is reduced accordingly.

SRXB(2)-11.0: DETAILED PLOTS

The neutron flux provided for the ATWS-Instability (RPT) calculation is core-average. Provide additional plots with hot channel powers at symmetric core locations showing the amplitude of the regional oscillations for the ATWS-Instability calculation.

Response SRXB(2)-11.0

Additional plots showing all hot channel powers in opposite regions are provided in Figures 11-1a through 11-1c. Three comparison plots are provided: hot channels 111 and 112, hot channels 113 and 114, and hot channels 115 and 116.

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SRXB(2)-12.0: PEAK CLADDING TEMPERATURE (PCT)

1. *Section 9.3.3 of the SAR specifies that the minimum stable film boiling temperature (T_{min}) correlation used is Shumway with clean Zirconium (Zr) credit (i.e., no Zr oxide) and no void credit. Is the TRACG quench model activated for these calculations or for the ATWS/Instability/RPT transient?*

2. *The ATWS/Instability/RPT calculation (Fig 9-12) shows a PCT heatup at ~140 sec when the power oscillations initiate. The PCT recovers and the rods seem to rewet at ~300 sec when the oscillations are mitigated by the flow/reduction. What mechanism allows for heatup and rewet?*

3. *Provide plots similar to Figure 9-12 that shows PCT superimposed with the calculated T_{min} value.*

4. *Provide tables showing the calculated margin between PCT and T_{min} for Figure 9-12.*

Response SRXB(2)-12.0

1. The quench model is activated for the Anticipated Transient Without Scram Instability (ATWSI) TRACG calculations.

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3. The "Peak Clad Temperature" in M+SAR Figure 9-12 shows the highest clad temperature in the core at each timestep. The location of the highest PCT can vary from different channels and/or levels. To address this RAI, The PCT, T_{min}, and total flow rate at a single location is provided. The chosen location is the location that resulted in the highest overall PCT. This location [[ ]] and is shown in Figure 12-2.

4. Figure 12-3 shows the margin for the limiting RPT case discussed in Response 12-3.

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SRXB(2)-13.0: TURBINE TRIP EVENTS

Results for turbine trip without bypass (TTNBP) during an AWTS-I event are not included in the SAR. Provide results for TTNBP in the MELLLA+ operating domain.

Response SRXB(2)-13.0

Typically, the TTNBP case is not performed for ATWS for the following reasons:

- The event is extremely unlikely by coupling the improbable event of a complete scram failure with an additional complete failure of the bypass,
- The Turbine Trip With Bypass (TTWBP) event displays similar results for ATWSI,
- The MSIVC event is performed and is generally more limiting than the TTNBP event during the initial pressurization part of the transient as discussed in NEDE-24222 (Reference 13-1), and
- For NMP2, the turbine trip events are not limiting due to the automatic feedwater runback.

To respond to this RAI, the TTNBP with scram failure is analyzed for instabilities and the results are shown in Figure 13-1. Because of the automatic feedwater runback, PCTs are low, and there are no oscillations in either event. The differences in PCT are due to the differences in pressure. PCTs follow the same trend as the saturated temperature.

References

13-1. NEDE-24222, "Assessment of BWR Mitigation of ATWS (NUREG 0460 Alternate No. 3)," December 1979.

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SRXB(2)-14.0: STEAM DRYER STRUCTURAL INTEGRITY

1. *Are the boundary conditions used in Acoustic Circuit Model, ACM, affected by MELLLA+ flow? Is there any impact on reactor water level and boundary conditions for annular region between dryer skirt and separator stand pipes, and annular region between reactor pressure vessel wall and dryer skirt? Is there any impact on dryer pressure loading used and on dryer structural analysis?*

2. *Are steam dryer stresses evaluated for EPU conditions bounding for plant operation at EPU conditions combined with MELLLA+ conditions?*

3. *Section 3.3.4 "Steam Line Moisture Performance Specification" states, "This increase resulted in a MCO value above the original moisture performance specification of 0.10 wt. %." ... "The amount of time NMP2 is operated with higher than the original design moisture content (0.10 wt %) is minimized by operations" ... "the NMP2 moisture carryover, MCO, will be monitored and controlled to < 0.25 wt. %". Provide a summary explanation of:*

   a. *What analyses were performed to determine the 0.25% permissible limit?*

   b. *What analyses were performed to determine the original 0.1% moisture carryover, MCO, under MELLLA+ conditions?*

   c. *What plant operations are used in NMP2 to minimize the moisture carryover, MCO?*

   d. *Provide a short physical explanation of what causes the increased moisture carryover, MCO, at lower flow. Is this mechanism predicted using an experimental correlation or a first principle analytical tool?*

   e. *How is the MCO monitored during operation? What is the typical surveillance period?*

Response SRXB(2)-14.0

1. 1st Question - Boundary conditions possibly affected by MELLLA+ occur at the: (1) steam/froth interface under the dryer; and (2) steam/water interface in the annular region between the skirt and the vessel. MELLLA+ is known to increase moisture content of the steam and decrease the pressure loads on the dryer under MELLLA+ operation. Thus, the current boundary conditions applied to MELLLA+ calculations would result in conservative loads. Also, wet steam damps acoustic oscillations.

   2nd Question - No, the difference in water height from the inside to the outside of the skirt is controlled by the change in pressure (Δp) across the skirt. As described in the MELLLA+ Licensing Topical Report the steam dryer Δp (RIPDS) for MELLLA+ is bounded by the EPU increased core flow condition, (i.e., the steam dryer Δp is less under MELLLA+ conditions than the EPU increased core flow conditions). The boundary conditions account for the effect of moisture.

   3rd Question - EPU defined a range of potential dryer pressure Reactor Internal Pressure Differences (RIPD) for normal, upset, emergency and faulted conditions. With an increased dryer outlet moisture performance specification up to 0.35 wt. %, the pressure drop between the dryer dome and the steam line is 0.72 psid, which is identical to that at 0.10 wt. % at two significant figures. The increased moisture content at the vessel outlet increases the dryer fluid density from 2.3320 lbs/ft^3 (0.10 wt. %) to 2.3376 lbs/ ft^3 (0.35 wt. %). This slight change in steam density (0.0056 lbs/ft^3), assuming a bounding moisture performance of 0.35 wt. %, does not increase steam dryer Δp loads above the EPU defined loads.

2. The steam dryer loads associated with MELLLA+ conditions for deadweight, reactor internal pressure differences, seismic, and SRV loads remain the same or are bounded by those considered in the EPU evaluation. Therefore, the structural integrity results of the EPU evaluation remain applicable for MELLLA+ as well as for normal, upset, emergency, and faulted conditions.

   The EPU acoustic steam dryer loads are established based on measured data from the Main Steam Line (MSL) strain gages. This data has established these loads are bounded by the steam line flow velocity squared relationship. The minor change in steam density (0.0056 lbs/ft^3) has an insignificant effect on the acoustic speed bounded by analysis uncertainty assumptions. The change in moisture carryover (MCO) has an insignificant impact on the steam dryer acoustic loads.

   NMP2 is a recirculation flow control design and achieves 85% core flow at constant recirculation pump speed. Therefore, pump vane passing frequency is unchanged from EPU conditions.

3.a. The NMP2 Main Steam Line (MSL) hardware was designed to operate with steam moisture content based on the original steam dryer outlet moisture performance specification of 0.10 wt. %. At this value of moisture carryover (MCO), the moisture in the steam at the Turbine Stop Valves is less than 1.0 wt. %, and all MSL components receive steam with moisture content below component design specifications.

   The 0.25 wt. % permissible limit for MCO in the steam leaving the Reactor Pressure Vessel (RPV) was determined using a conservative calculation of the pressure drop across the MSLs. The pressure in the MSLs decreases as the saturated steam flows from the RPV to the high-pressure turbine due to irreversible energy losses caused by surface friction, changes in direction and elevation, and flow through various MSL components. This reduction in pressure produces a corresponding increase in the moisture content of the steam as it flows through the MSL.

   The plant-specific analysis calculated the moisture content of the steam at affected MSL components as a function of steam dryer exit moisture. In the case of NMP2, the limiting MSL components were identified as the Outboard Main Steam Isolation Valves (MSIVs).

   The analysis determined the MCO value in the steam leaving the dryer that would produce a maximum moisture content of 0.50 wt. % at the Outboard MSIVs. This latter value was identified as the maximum moisture that the MSIVs could be subjected to during normal operations without an unreasonable expectation that the components would perform their intended design function. With the plant operating at the 0.25 wt. % permissible MCO limit, the steam moisture content at the Outboard MSIVs is just below 0.50 wt. % and remains below 1.0 wt. % at the turbine inlet. This analysis provides reasonable assurance that all MSL components will not experience accelerated degradation as a result of the increased moisture content of the steam.

3.b. The original 0.1 wt. % MCO functional performance design specification for the steam dryer was adopted primarily to ensure that the moisture content of steam does not exceed the design specifications of certain critical components located in the Main Steam Line (MSL). The 0.1 wt. % specification also maintains radiation levels in the Balance of Plant as low as reasonably achievable (ALARA). Critical MSL components have design specifications for operating conditions that specify a maximum steam moisture content. For example, the Main Steam Isolation Valves (MSIVs) typically have a design specification that limits steam moisture content to no more than 0.25 wt. %. This specification is satisfied when the MCO leaving the dryers is 0.10 wt. %. The 0.1 wt. % specification also limits the moisture content of the steam entering the turbine to well below the typical maximum inlet specification value of 1.0 wt. %. It should be noted that the 0.10 wt. % MCO functional design performance specification is not limited to M+ conditions, but applies to all operating conditions.

3.c. Historical and current reactor core design practice for each NMP2 reload is to limit radial peaking factor to less than 1.6. This is typically achieved thru the core loading and rod patterns prescribed for the operating cycle. This approach has resulted in MCO much less than 0.10 wt. %. The standard MCO trending measurement is monthly. The current NMP2 EPU MCO based on this trending is shown in Figure 14-1.

Going forward (beginning with NMP2 Cycle 15 MELLLA+ late cycle operations and Cycle 16 design), a moisture carryover software prediction tool will be utilized to assist in mitigating MCO. The tool is based on plant-specific modeling and utilizes actual MCO data for tuning. The predictions of MCO will allow core designers and reactor engineers to understand when the risk of increased MCO is significant. At the core design phase, rod patterns or core flow can be adjusted. For the rod patterns during the cycle where MCO is predicted and confirmed elevated above 0.10 wt. %, reactor engineering will optimize the rod patterns to the extent practical to minimize the MCO.

3.d. MCO increases at lower core flow as a result of the performance characteristics of the primary steam separators operating under M+ conditions (i.e., reduced core flow).

Separator performance, which is predicted using empirical correlations based on experimental test data obtained during developmental testing, is primarily affected by three parameters: (1) the immersion water level surrounding the separators; (2) the total mass flow rate passing through the separator; and (3) the quality of the steam entering the separator. Neglecting changes in the water level, which does not change during steady-state operation, optimal separator performance with regard to separator carryover (CO, defined as the amount of moisture in the steam exiting the separator) occurs over a relatively narrow band of steam separator inlet quality ranging from roughly 10 to 25 wt. %. This performance characteristic is illustrated by plotting typical test results of CO versus Separator Inlet Quality at a constant inlet flow rate as shown in Figure 14-2. CO increases rapidly when the inlet quality is beyond this optimal range, despite the decreasing moisture content of the steam.

The separators were designed to operate under a specific range of inlet qualities and mass flow rates. When the core flow is reduced in the M+ regime, the overall effect is to increase the quality of the steam exiting the fuel bundles in the core. Although some mixing of this steam occurs in the upper plenum region, the mixing is not sufficient to produce homogeneous conditions at the separator inlets. Consequently, the quality of the steam entering the separators will exhibit some spatial variation that depends largely on the quality of the steam leaving the fuel bundles beneath the separator inlets. Hence, the quality of the steam entering the central separators will generally be much greater than the quality of the steam entering the peripheral separators due to the higher power of the central fuel bundles and the reduced core flow.

At high inlet quality, the separators are not as effective in removing sufficient water content to meet the original design values such that more water passes through the separator to the dryer. If a significant number of the separators are affected in this way, high MCO conditions (>0.10 wt. %) can occur. The NMP2-specific M+ evaluation, which is a design basis hardware evaluation that incorporates additional margin into the analysis for conservatism, indicated the potential for several separators to operate with elevated separator CO under M+ conditions. If a sufficient number of separators pass high CO to the dryer, the increased moisture may exceed the local capacity of the dryer to remove the additional moisture due to localized flooding of the dryer vane channels. This condition, known as moisture breakthrough, is characterized by an abrupt increase in the moisture content of the steam exiting the dryer (i.e., MCO) above the original functional dryer performance specification of 0.10 wt. %. This situation is illustrated in Figure 14-3, which shows degraded central separator performance for a typical plant operating under hypothetical core power and flow conditions in the M+ regime. Separators with exit CO approaching the local breakthrough condition (i.e., greater than or equal to 80% of the local breakthrough value) are highlighted in yellow whereas separators with exit CO exceeding the local breakthrough condition are highlighted in red. As can be seen in Figure 14-3, the increase in the inlet steam quality that occurs under M+ core flow conditions causes increased carryover from the central separators.

Thus, under M+ operating conditions, the quality of the steam in the central separators may exceed the optimal range for inlet quality such that the separator CO is much higher than the 10 wt. % functional design specification. If a sufficient number of these separators have high CO, localized dryer breakthrough may occur, which leads to elevated MCO. However, also under M+ operating conditions, the quality of the steam in the peripheral separators may increase such that the separator CO is lower and thus reduce the local CO into some dryer banks and decrease the local MCO to improve dryer performance in these sections. The analyses described in Item 3a above confirm that that separator and dryer performance will remain acceptable under M+ conditions.

3.e The NMP2 chemistry procedure S-CAP-100 requires chemistry to take samples once per month. This procedure provides instructions for the determination of steam quality by Sodium-24 carryover analysis. This technique is consistent with BWRVIP-139 recommendations. The NMP procedure currently defines alert levels at 0.07 wt % for NMP2. The intent is to maintain the same alert levels based on the current EPU MCO levels based on the predictions which show the change in MCO should remain on average below the alert levels. The MELLLA+ testing includes obtaining initial baseline MCO for the MELLLA+ conditions. The alert levels may be adjusted based on the test baseline MCO.

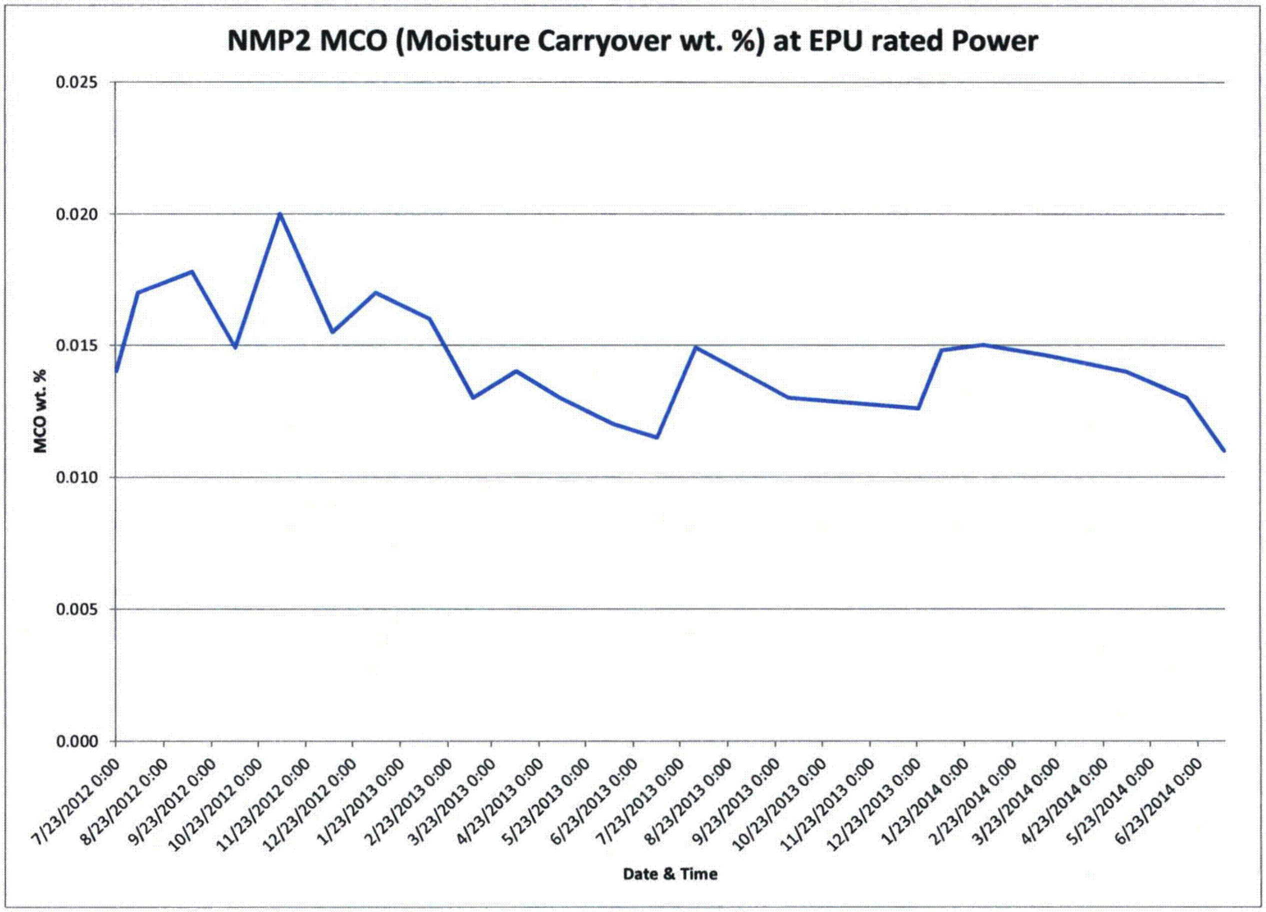

Figure 14-1. NMP2 MCO (Moisture Carryover Wt. %) at EPU Rated Power

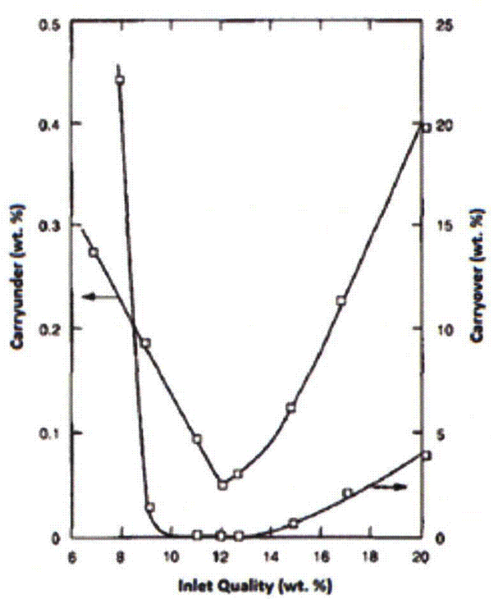

Figure 14-2. Typical separator performance characteristics with core effects omitted.

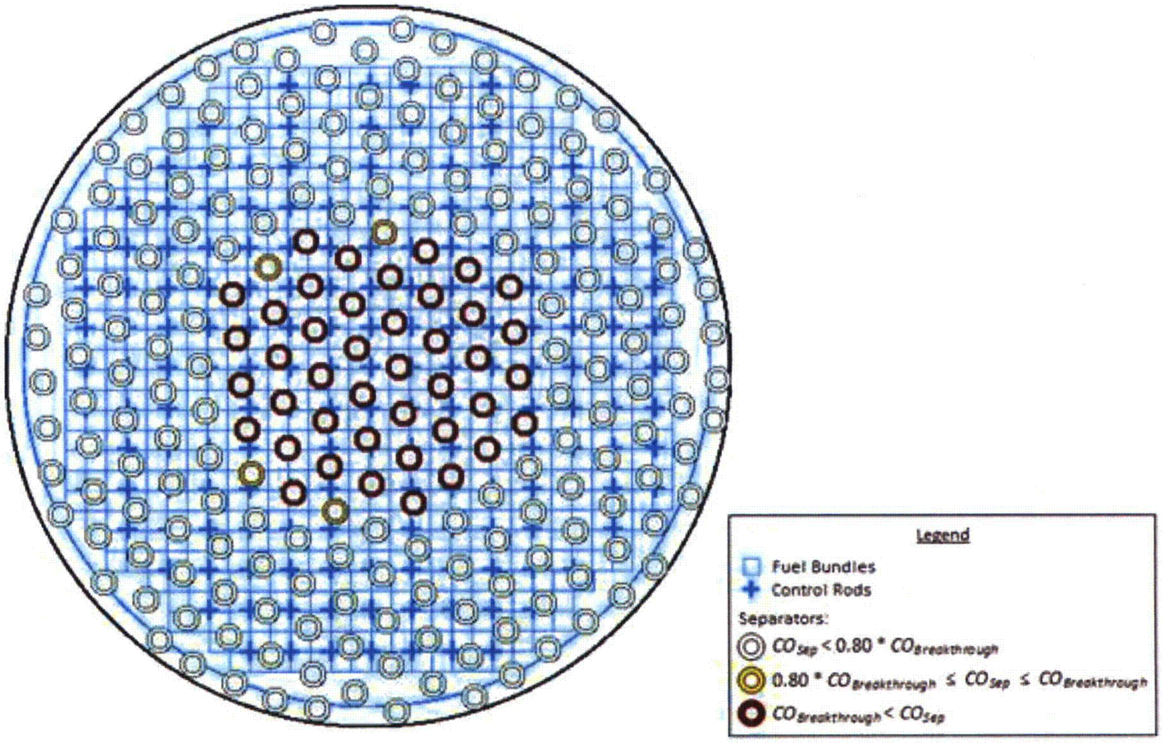

Figure 14-3. Core-Separator Map depicting central separators with degraded CO performance under hypothetical M+ operating conditions.

CO_{Sep} = moisture in the steam exiting the separator

$CO_{Breakthrough}$ = moisture in steam exiting the separator that exceeds localized dryer capacity to remove moisture to the dryer functional design specification of 0.10 wt. %

SRXB(2)-15.0: LARGE BREAK AND SMALL BREAK LOCA

1. *Section 4.3.1 Break Spectrum Response and Limiting Single Failure states,"A number of small break sizes were evaluated at the rated EPU power/MELLLA+ flow domain to determine the worst case small break." Provide a list of cases evaluated and indicate the limiting case.*

2. *The Small-Break LOCA results in Section 4.3.3 show the top-peaked axial power shape is limiting compared to the mid-peaked power shape. Why is Small-Break LOCA top-peak limited?*

3. *The results for Large-Break LOCA in Section 4.3.2 show the mid-peaked axial power shape being limiting. Why is Large-Break LOCA mid-peak limited?*

4. *In Section 4.3.2, explain the following regarding its results:*

   a. *The MELLLA+ LTR requires Small- and Large-Break LOCA analyses to include top-peaked and mid-peaked power shape in establishing the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and determining the PCT. Explain why only mid-peaked axial power shape is analyzed at 100% power and 100% flow.*

   b. *Explain why the mid-peaked axial power shape is limiting in terms of the PCT.*

   c. *What is the difference between mid-peak and top-peak values for the 1st peak results?*

   d. *Explain why the first peak is lower than the second peak for the mid-peaked axial power shape calculation in the Appendix K PCT analyses.*

   e. *Explain why no data is shown for the Nominal PCT cases.*

   f. *Please provide a plot of PCT versus time for Large-Break LOCA top- and mid-peaked axial power shape cases.*

Response SRXB(2)-15.0

1. The text referenced is from Section 4.3.1, second paragraph, regarding M+LTR Safety Evaluation Report (SER) Limitation and Condition 12.14 requiring a sufficient number of small recirculation line break sizes being analyzed at Current Licensed Thermal Power (CLTP) and Rated Core Flow (RCF). A list of small break size cases showing compliance is presented in a table of the M+SAR following Section 4.3.3.

   Investigations were conducted as directed by NEDC-33173P-A, "Applicability of GE Methods to Expanded Operating Domain" (Reference 15-1). As the project was initiated, the GESTR code was applied for fuel parameter input used by the Emergency Core Cooling System – Loss-of-Coolant Accident (ECCS-LOCA) evaluation model. According to the process approved by NEDO-33173, Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods" (Reference 15-2), the impact of the change of code would be implemented by evaluation model change reporting (per the 10 CFR 50.46 process). During progression of the analysis, transition to the PRIME code became available and to proactively address Limitations and Conditions 9.12 and 9.14 of NEDC-33173P-A (Reference 15-1), the analysis was augmented with limiting cases performed anew, explicitly incorporating the PRIME basis. These latter cases are the PRIME-based results reported in the indicated section of the M+SAR.

For completeness in this response, both the non-reported GESTR-based results as well as the PRIME-based small break cases of the M+SAR are shown below, noting the consistent trends. The PRIME-based results of the M+SAR are sufficient to show compliance to the acceptance criteria of 10 CFR 50.46, and fulfill M+LTR SER Limitation and Condition 12.11 to present the limiting small and large break Peak Cladding Temperatures (PCTs). The limiting break size, limiting axial power distribution (mid or top peaked) and demonstration of a bounding result with corresponding Nominal assumption case support the setting of the single licensing basis PCT, per the direction of M+LTR SER Section 4.3.1.5 (Reference 15-3).

There were no other small break cases analyzed than those listed below. The limiting small break case assumes a break size of 0.07 ft^2 with Appendix K assumptions and PCT of 1,561°F (bolded in the tables on the next page).

Small Break Summary Tables:

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2. The small break LOCA transient is characterized by a more slow progression of inventory loss compared to the more immediate expulsion of coolant from a design basis (or large break) event. For the limiting small break LOCA the core will not completely be uncovered. Rather, the ECCS response depends first on the break flow removing sufficient inventory that the level is reduced to the low level setpoint. Reaching that setpoint, the ECCS assets are mustered automatically,

considering the delay for systems to be actuated, pumps to be powered and running, and actual emergency cooling flow to become available to the core, this all factored into the analysis.

For analysis purposes, regulation requires assumption of the most limiting single failure. Loss of a diesel generator is postulated which removes high pressure systems, so no high pressure core spray is credited for Nine Mile Point Unit 2. Given that factor, then, the response of the Emergency Core Cooling System (ECCS) will rely subsequently upon the Automatic Depressurization System (ADS) timer, which also begins on low level setpoint, then, upon completion of that delay, the opening of safety relief valves to depressurize the vessel. As a sufficiently low back pressure is reached, the Low Pressure Coolant Injection (LPCI) is credited to begin delivery.

During this time, the ongoing loss of inventory will have resulted in the partial uncovering of the core from the top down. The extent of heatup for the core will be driven by the depth of uncovered cladding and the duration for which it remains uncovered for this period until cooling ECCS inventory arrives to recover it. A mid-peak power shape would show higher powered nodes covered with coolant water during much of this period with much less heating effect on cladding. If the power distribution is concentrated to the top, heat is generated for a longer time in uncovered spans of the bundles without cooling resource immediately present, resulting in greater heat up. Thus, for the small break LOCA, assuming top peaked power distribution, results show an overall higher Peak Cladding Temperature (PCT) at those higher levels, uncovered elevations of the core.

3. The nature of the design basis (or large break) case assumption is the severance of a recirculation line, or double-ended guillotine break. A blowdown of the vessel occurs and core flow begins to decrease immediately. Recirculation flow stops in the broken loop and recirculation flow coasts down in the intact loop. This first depressurization causes an expansion of resident water, nucleate boiling is lost, giving way to boiling transition (and occurrence of first peak PCT). The pressure trend is held up somewhat by the increased steaming. Then, as the level of water breaks suction on the jet pump, a communication path is established to outside ambient and with sudden escape of collapsible steam, the vessel depressurizes rapidly and the core is uncovered.

   ECCS delivery is accomplished following setpoints being reached on high containment pressure or low level, subject to single active failure, and initiation of pump operation with expected delays. But, in the condition of an uncovered core, two phenomena tend to define the elevation most vulnerable to heating.

   Counter current flow limitation on the bottom of the core bundles will tend to restrict the loss rate of all water from the bottom of the core, some amount being held up there awaiting drain to the lower plenum, affording conduction cooling on the bottom ends of the rods. Also, as steam resides in the core environment, or is generated from continued boiling on the cladding surface at the reduced vessel pressure, and more so as ECCS inventory begins to be delivered to the core during refilll, its effect as it traverses the length of the bundle up through the rods is beneficial to the higher elevation nodes it passes by absorbing additional heat (the so-called steam cooling effect) later in the transient. These two cooling effects leave the middle elevation of the core with the least effective means of cooling relief overall. Power concentrated in that elevation as an initial condition has been found historically to lead to the greatest increase in cladding temperature or resulting peak cladding temperature for large break assumption.

4. As explained in response to Item 1 in this RAI, the Nine Mile Point Unit 2 MELLLA+ analysis was initiated using the GESTR code for determination of fuel parameters input to the analysis. Then, as the transition to the PRIME code became available, limiting cases only were proactively performed anew to account for that basis in the analysis, sufficient to confirm bounding cases and results for establishing a licensing basis PCT. Presenting the GESTR results for the large break analysis in the table below, a discussion of these questions can then follow.

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4.a It was known from the interim results with GESTR that the 100% power and 100% flow would not be the limiting power/flow point for PCT assuming a large break. Sufficient for the PRIME demonstration was to show that by recalculating and presenting the two mid-peak cases [[ ]]. To satisfy the requirement of the M+LTR, this known limiting large break case was re-calculated with PRIME assuming a top-peaked axial power distribution, to confirm that the mid-peak power assumption, previously and historically seen as limiting, remained so.

4.b The response to Item 3 in this RAI, above, offers rationale as to the conditions of the predicted uncovered core and why the mid-peaked power shape is limiting for a large break. The mid-peaked power shape has been historically limiting for the SAFER model, particularly in the large break scenario, and is no different in this application.

4.c For the limiting large break cases above, the 1st peak PCTs are shown below for the mid-peak and top peak cases.

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The difference between these 1st peak PCT values is shown to be [[ ]].

4.d As noted in Item 3 in this RAI, above, the first peak is caused by the sudden pressure loss of the break, the expansion of the water in the vessel and loss of nucleate boiling. It is driven principally by the stored energy in the core at the time of the postulated accident. It is later, after the core has uncovered, that the heat up from decay heat being generated in the rods has a longer duration, greater effect, and more extensive heat up leading to the limiting 2nd peak PCT, until ECCS delivery is able to quench the rod and reverse the temperature excursion.

4.e The purpose of the span of cases is to identify the limiting case on which to report licensing basis PCT and claim compliance to acceptance criteria. A change of methodology imposed upon the evaluation model by the M+LTR SER Section 4.3.1.5, Item 1, is that the survey of potentially limiting power and flow conditions would be performed considering application of uncertainties, or, effectively, applying Appendix K assumptions to confirm the overall limiting case. The Nominal PCT remains significant, then, only to the extent of establishing uncertainty effects, confirming sufficient conservatism has been applied in deducing and reporting a licensing basis PCT per NEDE-23785-1-PA (Reference 15-4), the LTR for the evaluation model.

Note that during initial investigations with GESTR, before the limiting break size had been indicated, the large break Nominal case calculations had been performed as shown above. As noted, since it became known a priori that the large break with PRIME would not be limiting, the calculation of Nominal assumption cases for these break sizes became superfluous in the large break instance and were not re-calculated. (Also, note that Nominal assumption cases for determining the licensing basis PCT are shown with PRIME for the limiting power/flow, small break cases.)

4.f Plots of PCT vs. time for two large break cases, assuming top-peaked or mid-peaked axial power shape, are presented on the next page.

References

15-1 GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

15-2 GE Hitachi Nuclear Energy, "Implementation of PRIME Models and Data in Downstream Methods," NEDO-33173 Supplement 4-A, Revision 1, November 2012.

15-3 GE Hitachi Nuclear Energy, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," NEDC-33006P-A, Revision 3, June 2009.

15-4 General Electric Company, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident Volume III, SAFER/GESTR Application Methodology," NEDE-23785-1-PA, Revision 1, October 1984.

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Figure 15-1 – Peak Cladding Temperature – EPU Power / 85% Flow, Large Recirculation Suction Line Break, Mid-Peaked Power Shape, (Appendix K Assumptions)

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Figure 15-2 – Peak Cladding Temperature – EPU Power / 85% Flow, Large Recirculation Suction Line Break, Top-Peaked Power Shape, (Appendix K Assumptions)

SRXB(2)-16.0: OPRM ARMED REGION

The generic MELLLA+ flow domain is 80% - 100% flow at 100% power. NMP2 chose a different flow domain of 85% - 100% flow at 100% power. The generic Oscillation Power Range Monitor (OPRM) armed region value for the DSS-CD is 75% flow. Please explain why the generic value for the armed region for the DSS-CD was not adjusted commensurate with the adjustments to the generic flow domain? Please provide the criteria and methodologies used to set the OPRM armed region.

Response SRXB(2)-16.0

The reason for not adjusting the OPRM Armed Region is described below:

The OPRM Armed Region is defined generically as 75% Recirculation Drive Flow (RDF) for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) plants, regardless of the extent of the flow domain used.

The criteria used to set the Oscillation Power Range Monitor (OPRM) Armed Region are described in Section 4.5 of the reviewed and approved Detect and Suppress Solution - Confirmation Density (DSS-CD) Licensing Topical Report (LTR) (Reference 16-1).

The specified flow level is designed to disarm the trip and alarm functions during rated power operations. Because power oscillations are not expected at rated power operations and the reactor is operated at these conditions most of the time, disarming the trip function reduces the probability of unnecessary spurious scrams. In addition, the specified flow level is designed to arm the trip and alarm functions at a flow level that bounds the core conditions potentially susceptible to power oscillation. Historically thermal-hydraulic instability events in operating Boiling Water Reactors (BWRs) have not occurred operating at core flows close to or larger than 60% rated core flow. Therefore, the use of the generic 75% flow is already significantly bounding the region of the power/flow map where thermal-hydraulic instability events may occur and there is no need for it to be adjusted.

The criteria and methodologies used to set the OPRM Armed Region are the following:

Power Level - As instabilities are not expected to occur below 30% OLTP, the power threshold is generically set to be the MCPR monitoring threshold of 25% OLTP, which scales to a lower value for power uprated plants. [[

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Flow Level - As instabilities are not expected to occur at, or close to, rated conditions, the flow boundary threshold is generically set just below the minimum flow associated with rated power operation, specified in Reference 16-1 as 75% of RDF for MELLLA+ plants. For NMP2, the OPRM flow boundary of the OPRM Armed Region is set at 75% RDF. The NMP2 OPRM Armed Region power and flow boundaries are generically established in the NMP2 MELLLA+ Safety Analysis Report as shown in Figure 2-19 (Reference 16-2).

References

16-1 GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

16-2 GE Hitachi Nuclear Energy, "Safety Analysis Report for Nine Mile Point 2 Maximum Extended Load Line Limit Analysis Plus," NEDC-33576P, Revision 0, October 2013.

SRXB(2)-17.0: SIMULATOR UPDATE

1. *Describe any modifications to the training status of the key operator actions credited in the TRACG ATWS-Instability analysis.*

Response SRXB(2)-17.0

1. TRACG analysis predicts onset of greater than 25% oscillations for ATWS Instability events, which requires mitigation through the EOP level reduction strategy. The ATWSI analysis credits manual action to commence lowering water level based on a 250 second delay from the time the operator inserts the manual scram.

   Operations Simulator Training is scheduled to be conducted from September 2, 2014 through October 10, 2014. This will include the training on operator actions of scram insertion within 20 seconds and RPV water level reduction within the following 250 seconds for ATWSI events.

SRXB(2)-18.0: CORE DESIGN

1. *Please verify that the document from James J. Stanley to the NRC dated February 25, 2014, "License Amendment Request Pursuant to 10 CFR 50.90: Maximum extended Load Line Limit Analysis Plus - Core Reload and Safety Limit Supplemental Information" is the final supplemental reload licensing report (SRLR) for NMP2 MELLLA+ operating Cycle 15. If the document is not a final SRLR, when will the final SRLR be available to support NMP2 MELLLA+ operation?*

2. *When will the final Core Operating Limits Report be available to support NMP2 MELLLA+ operation?*

3. *Table 2-1 and Figures 2-1 through 2-6 of the SAR indicate the core design and fuel monitoring parameters for each exposure statepoint. Table 2-1 shows the peak nodal exposures ranging from 38.849 to 56.660 GWd/ST (52.003 GWd/ST for NMP2 MELLLA+) and Figures 2-1 through 2-6 show cycle exposure only up to 18 GWd/ST.*

    a. *Why do the figures show the data only up to 18 GWd/ST?*

    b. *Provide values for maximum bundle power, flow for peak bundle, exit void fraction for peak power bundle, maximum channel exit void fraction, core average exit void fraction, and peak LHGR at peak nodal exposure.*

4. *Please provide a detailed description and basis as to why the operational conditions for NMP2 in the MELLLA+ operating domain are within expected parameters based on the data shown in Figures 2-7 through 2-17.*

Response SRXB(2)-18.0

1. The NMP2 Cycle 15 SRLR provided to the NRC on February 25, 2014 supports Cycle 15 operation with Extended Power Uprate (EPU) /Maximum Extended Load Line Limit Analysis (MELLLA) (approved) and EPU/MELLLA+ (pending approval). This is the final SRLR for NMP2 MELLLA+ operation in Cycle 15.

2. The NMP2 COLR for NMP2 MELLLA+ Operation is drafted. However, it will not be approved and issued until after the EPU/MELLLA+ License Amendment Request, including the modified SLMCPR technical specification, is approved and issued by the NRC. Typically, COLRs are provided to the NRC approximately 30 days after implementation.

3. The NMP2 MELLLA+ SAR Figures 2-1 through 2-6 are plotted as a function of cycle exposure for the equilibrium core design consistent with the NMP2 Power Uprate Safety Analysis Report (PUSAR) and other plant EPU applications based on the Methods Licensing Topical Report (LTR) Safety Evaluation Report (SER) Limitation and Condition 9.24 (Reference 18-2).

    a. The NMP2 MELLLA+ SAR equilibrium core design has a maximum cycle exposure of 18.577 GWd/ST (20.478 GWd/MT), a maximum bundle exposure of 43.337 GWd/ST (47.771 GWd/MT), and a maximum nodal exposure of 52.003 GWd/ST (57.323 GWd/MT).

    b. Table 18-1 includes the requested data values for the NMP2 M+SAR curve supporting Figures 2-1 through 2-6. The requested peak Linear Heat Generation Rate (LHGR) at peak nodal exposure is different than the data in Figure 2-6. The bundles with peak nodal exposure is typically third cycle bundles on the periphery.

The Thermal-Mechanical Operating Limit (TMOL) curve for GE14 as defined by GESTAR-II (Reference 18-1) is satisfied independent of the operating domain using bundle design (pin-by-pin enrichment and gadolinia configuration) and core design (loading of fresh, once, and twice burnt streams). Figure 18-1 shows the nodal LHGR versus nodal exposure for all nodes in all bundles in the NMP2 M+SAR equilibrium core design operation from Beginning of Cycle (BOC) to End of Cycle (EOC).

4. The 2D core data in Figures 2-7 through 2-17 are representative of the NMP2 MELLLA+ equilibrium core design showing bundle power, LHGR, and Minimum Critical Power Ratio (MCPR) results. LHGR and MCPR are satisfied on a cycle-specific basis by design because these Specified Acceptable Fuel Design Limits (SAFDLs) are required. The equilibrium core design details shown in NMP2 M+SAR Figures 2-7 through 2-17 are an example of what to expect for NMP2 for the MELLLA+ operating domain. NMP2 intends to implement MELLLA+ in Cycle 15 when approved. The Cycle 15 results are similar to the equilibrium results.

References

18-1. Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-20 and NEDE-24011-P-A-20-US, December 2013.

18-2. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

18-3. GE Hitachi Nuclear Energy, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," NEDC-33006P-A, Revision 3, June 2009.

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Figure 18-1. NMP2 EPU/MELLLA+ (Equilibrium GE14 M+SAR)

Table 18-1. Requested Data Values for the NMP2 M+SAR Curve Supporting Figures 2-1 through 2-6

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SRXB(2)-19.0: STANDBY LIQUID CONTROL SYSTEM

1. *Provide rationale to revise the acceptance criterion in Surveillance Requirement 3.1.7.7 from a discharge pressure of ≥ 1,327 pounds per square inch gauge (psig) to ≥ 1,335 psig.*

Response SRXB(2)-19.0

For Technical Specification (TS) requirements, the maximum Standby Liquid Control System (SLS) pump discharge pressure for the limiting MELLLA+ Anticipated Transient Without Scram (ATWS) event is 1,331.3 psig, based on a reactor pressure of 1,226.3 psig and a SLS pressure drop of 105 psi at a Technical Specification 41.2 gpm minimum SLS pump flow rate. 1,331.3 psig (rounded up to 1,332 psig) is the minimum required pump discharge pressure to satisfy TS SR 3.1.7.7. In anticipation of future new fuel introductions and uncertainties, an additional 3 psi is added, resulting in the surveillance requirement of 1,335 psig.

SRXB(2)-20.0: CYCLE SPECIFIC SAFETY LIMITS

1 *It appears that both Figure 1 of Attachment 11 to the SAR and Figure 1 of the February 25, 2014, SRLR show the same core loading pattern. Please verify that Figure 1 in both documents is the same final core loading pattern. If they are not the same final core loading pattern, explain what they represent.*

2. *Figure 3 in Attachment 11 of the SAR shows Monte Carlo trials for the SLMCPR value through all of the uncertainty subroutines. Are there any Part 21 reports associated with SLMCPR?*

3. *Table 3 of Attachment 11 shows that the Final Estimated SLMCPR for a state-point at 100% power and 85% flow is [[ ]]. The Calculated Monte Carlo SLMCPR is [[ ]]. Explain why the 0.02 adder for MELLLA+ operation was applied to the Calculated Monte Carlo SLMCPR and not the Final Estimated SLMCPR.*

4. *Was the STERN test data used to improve the non-power distribution uncertainties shown in Table 4 in Attachment 11? If so, explain how the data was used. If not, explain why?*

Response SRXB(2)-20.0

1. The cycle-specific SRLR [Reference 20-1] dated February 25, 2014 supports Cycle 15 operation with Extended Power Uprate (EPU)/ Maximum Extended Load Line Limit Analysis (MELLLA) (approved) and EPU/MELLLA+ (pending approval). The tech specification change letter [Reference 20-2] supports Cycle 15 operation with EPU/MELLLA+ conditions, pending approval of the MELLLA+ License Amendment Request. The Cycle 15 loading is unchanged to support EPU/MELLLA operation at the start of BOC15 and EPU/MELLLA+ operation (when approved), so Figure 1 in the Cycle 15 SRLR and Figure 1 in cycle specific Attachment 11 to the M+SAR are the same.

2. Attachment 11 notes in Figure 3 that the calculation procedure for SLMCPR evaluations from the approved Figure 4.1 of NEDC-32601P-A is still the same. Attachment 11 also notes in Section 2.13 that there are no known 10CFR Part 21 factors that affect the NMP2 Cycle 15 SLMCPR calculations.

3. The Final Estimated Two Loop Operation (TLO) Safety Limit Minimum Critical Power Ratio (SLMCPR) using the MIPRIP Correlation and the Calculated Monte Carlo SLMCPR are values before the "Additional SLMCPR Licensing Conditions" (Methods LTR L&C 9.5 adder) are applied. The NMP2 M+SAR has elected to apply a conservative +0.02 adder at all statepoints instead of the revised Power-to-Flow ratio dependent limits of Methods LTR L&C 9.5 adder discussed in detail in the NMP2 MELLLA+ RAI 2.0 response. The TLO SLMCPR estimate using the MIPRIP Correlation is not the approved method to establish the numerical SLMCPR recorded in the technical specification.

4. The GE14 STERN test data was not used to improve the non-power distribution uncertainties approved in NEDC-32601P-A as noted in Table 4 of Attachment 11. Any changes to NEDC-32601P-A values would require a review and approval.

References

20-1. Global Nuclear Fuel, "Supplemental Reload Licensing Report for Nine Mile Point 2 (NMP2) Reload 14 Cycle 15 Extended Power Uprate (EPU) / Maximum Extended Load Line Limit Plus (MELLLA+)," 000N0123-SRLR, Revision 2, January 2014. [NMP2 Cycle 15 SRLR]

20-2. Global Nuclear Fuel, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Nine Mile Point 2 Cycle 15," GNF-0000-0156-7490-R0-P, Revision 0, August 2013. [NMP2 Cycle 15 Tech Spec Change Letter]

SRXB(2)-21.0: TECHNICAL SPECIFICATIONS

Please identify which version of GESTAR is used for TS 5.6.5.b.1.

Response SRXB(2)-21.0

The version of GESTAR that is utilized for TS 5.6.5.b.1 will be specified in the Core Operating Limits report as:

- General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-19, May 2012; and the U.S. Supplement, NEDE-24011-P-A-19-US, May 2012

The basis for this version is the NMP2 Supplemental Reload Licensing Report that was submitted to the NRC on February 25, 2014 (ML14064A321).

- Global Nuclear Fuel, Supplemental Reload Licensing Report for Nine Mile Point 2 (NMP2) Reload 14 Cycle 15 Extended Power Uprate (EPU)/Maximum Extended Load Line Limit Plus (MELLLA+), 000N0123-SRLR, Revision 2, dated January 2014 (ML14064A322)

SRXB(2)-22.0: LIMITING EVENTS ANALYZED IN A TWS VERSUS ATWS-I

1. *For NMP2, the limiting ATWS events analyzed were initiated from 100% current licensed thermal power, CLTP, and 85% rated core flow at beginning of cycle, BOC, and EOC exposure conditions. Why is peak reactivity exposure not analyzed as a limiting event for ATWS?*

Response SRXB(2)-22.0

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SRXB(2)-23.0: ATWS ANALYSIS RESULTS

Table 9-4 shows the key results of the ATWS analyses. Footnote 2 of Table 9-4 states "Coolable core geometry is ensured by meeting the 2200°F PCT and 17% local cladding oxidation acceptance criteria of 10 CFR 50.46." No calculation was performed for peak local cladding oxidation. Verify that the peak local cladding oxidation is insignificant under NMP2 MELLLA+ operation.

Response SRXB(2)-23.0

The full footnote for the local cladding oxidation (footnote 2) states the following, [[

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