William R. Gideon
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o: 910.832.3698

February 5, 2018

Serial: BSEP 18-0021 10 CFR 50.90

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion (CAC Nos. MF8864 and MF8865)

References:

1. Letter from William R. Gideon (Duke Energy) to the U.S. Nuclear Regulatory Commission Document Control Desk, *Request for License Amendment Regarding Core Flow Operating Range Expansion*, dated September 6, 2016, ADAMS Accession Number ML16257A410

2. NRC E-mail Capture, *Brunswick Unit 1 and Unit 2 - Request for Additional Information Related to the MELLLA+ LAR (CACs MF8864 and MF8865) (Nonproprietary),* dated January 5, 2018, ADAMS Accession Number ML18010A051

Ladies and Gentlemen:

By letter dated September 6, 2016 (i.e., Reference 1), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendment revises Technical Specifications (TSs) 3.1.7, 3.3.1.1, 3.4.1, 5.6.5, adds a new TS 5.6.7, and adds a new license condition to Appendix B of the operating license (OL), to support an expansion of the core power-flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)). On January 5, 2018, by electronic mail (i.e., Reference 2), the NRC provided a request for additional information (RAI) regarding the LAR.

Duke Energy's response to the RAI is provided in Enclosure 1. The questions and/or responses to several of these RAIs involve information considered proprietary to Framatome Inc. (i.e., formerly known as AREVA) or General Electric Hitachi (GEH).

As discussed with and agreed to by the NRC licensing project manager, Mr. Andrew Hon, NRC Question SNPB-RAI-2 will require additional time to develop a response. Accordingly, the response to SNPB-RAI-2 will be provided under separate letter no later than March 1, 2018.

Framatome and GEH, as owners of the proprietary information, have executed the affidavits provided in Enclosures 2 and 3, respectively, which identify the proprietary information that has

been handled and is classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to Duke Energy in Framatome and GEH transmittals that are referenced by the affidavits. The proprietary information has been faithfully reproduced in Enclosure 1 such that the affidavits remain applicable. The Framatome proprietary information is identified by single square brackets. **[** This sentence is an example. **]** The GEH proprietary information is identified by double square brackets. **[[** This sentence is an example.{3} **]]**. For the GEH proprietary information markings, the superscript notation {3} refers to Paragraph (3) of the affidavit provided in Enclosure 4 which provides the basis for the proprietary determination.

A non-proprietary version of the responses is provided in Enclosure 4.

No regulatory commitments are contained in this letter.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager - Regulatory Affairs, at (910) 832-2487.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on February 5, 2018.

Sincerely,

William R. Gideon

WRM/wrm

Enclosures:

1. Response to Request for Additional Information **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**
2. Framatome Affidavit Regarding Withholding Framatome Information Contained in the Enclosures to the letter BSEP 18-0021, entitled "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion (CAC Nos. MF8864 and MF8865)," dated January 31, 2018
3. GEH Affidavit Regarding Withholding GEH Information Contained in Attachment 1 of GEH Letter, GEH-PGN-MPLUS-149, "GEH Responses to NRC MELLLA+ Requests for Additional Information SRXB-RAI-1 through 7 and 9 through 11, SNPB-RAI-4c and 7, and EMIB-RAI-1," dated January 26, 2018
4. Response to Request for Additional Information (Non-Proprietary)

cc (with all enclosures):

U.S. Nuclear Regulatory Commission, Region II
ATTN: Ms. Catherine Haney, Regional Administrator
245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

U.S. Nuclear Regulatory Commission
ATTN: Mr. Gale Smith, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U.S. Nuclear Regulatory Commission
ATTN: Mr. Andrew Hon (Mail Stop OWFN 8G9A)
11555 Rockville Pike
Rockville, MD 20852-2738

cc (with enclosures 2, 3, and 4):

Chair - North Carolina Utilities Commission **(Electronic Copy Only)**
4325 Mail Service Center
Raleigh, NC 27699-4300
swatson@ncuc.net

Mr. W. Lee Cox, III, Section Chief **(Electronic Copy Only)**
Radiation Protection Section
North Carolina Department of Health and Human Services
1645 Mail Service Center
Raleigh, NC 27699-1645
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Framatome Affidavit Regarding Withholding Framatome Information
Contained in the Enclosures to the letter BSEP 18-0021,
entitled "Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324 Response to Request for
Additional Information Regarding Request for License Amendment
Regarding Core Flow Operating Range Expansion
(CAC Nos. MF8864 and MF8865),"
dated January 31, 2018

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the Enclosures to the letter BSEP 18-0021, Entitled, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion (CAC Nos. MF8864 and MF8865)," and referred to herein as "Documents." Information contained in these Documents has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and are of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents

be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in these Documents have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Alan F. Meg

SUBSCRIBED before me this 31st

day of January, 2018.

Hailey M. Siekawitch

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

GEH Affidavit Regarding Withholding GEH Information
Contained in Attachment 1 of GEH Letter GEH-PGN-MPLUS-149,
"GEH Responses to NRC MELLLA+ Requests for Additional Information
SRXB-RAI-1 through 7 and 9 through 11, SNPB-RAI-4c and 7,
and EMIB-RAI-1," dated January 26, 2018

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Lisa K. Schichlein, state as follows:

(1) I am a Senior Project Manager, 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 Attachment 1 of GEH letter, GEH-PGN-MPLUS-149, "GEH Responses to NRC MELLLA+ Requests for Additional Information SRXB-RAI-1 through 7 and 9 through 11, SNPB-RAI-4c and 7, and EMIB-RAI-1," dated January 26, 2018. The GEH proprietary information in Attachment 1, which is entitled "Response to SRXB, SNPB, and EMIB RAIs in Support of Brunswick Steam Electric Plant MELLLA+ LAR," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

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

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

(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 is true and correct.

Executed on this 26th day of January 2018.

Lisa K. Schichlein

Lisa K. Schichlein
Senior Project Manager, NPP/Services Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
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Lisa.Schichlein@ge.com

Response to Request for Additional Information
(Non-Proprietary)

Response to Request for Additional Information (Non-Proprietary)

By letter dated September 6, 2016, Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendment revises Technical Specifications (TSs) 3.1.7, 3.3.1.1, 3.4.1, 5.6.5, adds a new TS 5.6.7, and adds a new license condition to Appendix B of the operating license (OL), to support an expansion of the core power-flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)). On January 5, 2018, by electronic mail, the NRC provided a request for additional information (RAI) regarding the LAR. The response to the RAI is provided below.

The Framatome (i.e., formerly known as AREVA) proprietary information is identified by single square brackets. **[** This sentence is an example. **]** The General Electric Hitachi (GEH) proprietary information is identified by double square brackets. **[[** This sentence is an example.{3} **]]**. For the GEH proprietary information markings, the superscript notation {3} refers to Paragraph (3) of the GEH affidavit which provides the basis for the proprietary determination.

Regulatory Basis:

The following Title 10 of the U.S. Code of Federal Regulations (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC) are applicable:

- GDC 1, "Quality standards and records," requires, in part, that the structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that there is adequate assurance that a SSC performs its safety function.

- GDC 4, "Environmental and dynamic effects design bases," requires, in part, that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

- GDC 10, "Reactor design," requires that: The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 12, "Suppression of reactor power oscillations," requires that: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 28 "Reactivity limits, requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (a) result in damage to the reactor

coolant pressure boundary greater than the limited local yielding, or (b) sufficiently impair core cooling capability.

NRC SRXB-RAI-1

Please provide the BSEP-specific noise data and discussion of the licensee's evaluation of these data used to justify a Detect and Suppress Solution Confirmation Density (DSS-CD) amplitude discriminator setpoint (S_{AD}) value of **[[ ]]** to minimize the likelihood of spurious scram while ensuring that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded.

Response SRXB-RAI-1

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References

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

SRXB-1-2 "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," DUKE-0B21-1104-000(P), July 2016.

NRC SRXB-RAI-2

Please provide a basis for the use of a DSS-CD time period lower limit (T_{min}) of **[[ ]]**, which is higher than the approved value of **[[ ]]** given in the DSS-CD license topical report (LTR) and reduces the range of oscillations indicative of an anticipated reactor instability. This basis will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded. Please include the following components:

a. Provide justification, based on plant data, that indicates that the **[[ ]]** value may lead to an unacceptable or undesired likelihood of spurious scram in BSEP during Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operation; and

b. Demonstrate that thermal-hydraulic (T-H) instabilities are not expected to occur below a T_{min} of **[[ ]]**. As part of this demonstration, please include analysis based on the TRACG DSS-CD analyses that were provided in the Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus (M+SAR, Enclosure 6 of the LAR), as well as a justification that the T-H oscillation period would remain above **[[ ]]** in both units, at any cycle exposure, and at any other point in the DSS-CD armed region.

Response SRXB-RAI-2a

The Confirmation Density Algorithm (CDA) utilizes the Period Based Algorithm (PBA), which is designed to recognize periodic oscillatory behavior in Local Power Range Monitors (LPRMs) or Oscillation Power Range Monitor (OPRM) cell signals. PBA utilizes equivalent time period limits (T_{min} and T_{max}) to determine the oscillatory nature of the OPRM cell signal. These parameters define the limits within which successive oscillation periods may vary from the first (base) oscillation period to increment the number of confirmation counts.

Additionally, alarms are provided within the CDA to alert the operator of an increase in the number of confirmed period counts so actions can be taken to avoid a reactor scram. The CDA alarm setpoints, the successive confirmation count (N_{AI}) and the amplitude (S_{AA}) components are selected on a plant-specific basis to ensure that no spurious alarms occur during stable plant operation. The alarm setpoints may be applied to the leading OPRM cell. The alarm occurs when the successive period confirmation count for any single OPRM cell (in any OPRM channel) reaches N_{AI} and its amplitude exceeds S_{AA}. Alternatively, the alarm setpoint may be applied to the second confirming OPRM cell (i.e., provided a single OPRM cell exceeds N_{AI}, the alarm is generated when any additional OPRM cell in the same OPRM channel exceeds both N_{AI} and S_{AA}).

PBDA was the primary algorithm in the Option III stability Long Term Solution (LTS), and it is retained in DSS-CD with the defined parameter settings documented in Table 3-4 of Reference SRXB-2-1. PBDA will provide a scram if **[[**

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a. A stable reactor normally exhibits small, random deviations from the steady-state neutron flux conditions. The response of a stable reactor to global noise perturbations quickly becomes incoherent. Either the response rapidly decays to the background noise level due to the stable core conditions, or subsequent unrelated perturbations disturb the natural decay characteristics, thus resulting in a reset of the Successive Confirmation Counts (SCC).

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Figure SRXB-2-1 BSEP Unit 1 Channel 1 Cell 8 SCC Data

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Figure SRXB-2-2 BSEP Unit 1 Channel 1 Cell 8 Relative Signal Data

Response SRXB-RAI-2b

Significant flow reduction events from power operation may result in operating conditions that are unstable. This is more likely for Two Recirculation Pump Trip (2RPT) events that initiate from the rated power and minimum flow conditions. Because the reactor state condition is rapidly changing during the 2RPT event, the ensuing oscillations are not developed instantaneously. The transition to a coherent oscillation mode involves the alignment of the entire core, which not only requires some limited duration but also may exhibit transitional effects. The oscillation frequency, and therefore, the detected period for individual channels may exhibit modulated behavior.

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Table SRXB-2-1 TRACG04 DSS-CD Results for ATRIUM 10XM Fuel with an [[ ]]

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Figure SRXB-2-3 BSEP TRACG [[

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Figure SRXB-2-4 BSEP TRACG [[

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Figure SRXB-2-5 BSEP TRACG [[

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Figure SRXB-2-6 BSEP TRACG [[

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References

SRXB-2-1 GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

SRXB-2-2 "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," DUKE-0B21-1104-000(P), July 2016.

NRC SRXB-RAI-3

In the SAR, Anticipated Transient without Scram (ATWS) and ATWS Instability (ATWS-I) results were presented using the limiting fuel parameter sensitivity values to ensure that the ATRIUM 10XM fuel in BSEP was adequately modeled using GEH methods. Please justify the acceptability of **[[ ]]**, for the DSS-CD confirmatory analyses. This will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded for the ATRIUM 10XM fuel.

Please justify that the **[[ ]]** used in the DSS-CD approach conservatively bounds the uncertainty associated with ATRIUM 10XM fuel when the limiting fuel parameter values are considered.

Response SRXB-RAI-3

For DSS-CD applications, **[[**

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References

SRXB-3-1 GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

SRXB-3-2 "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," DUKE-0B21-1104-000(P), July 2016.

NRC SRXB-RAI-4

The following additional information is requested for the turbine trip with bypass (TTWBP) ATWS-I analyses to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event:

a. Please describe the approach used to ensure that the maximum steady-state Linear Heat Generation Rate (LHGR) for the TRACG TTWBP analyses was less than **[[ ]]** of the Maximum LHGR limit. Were the steady state conditions (e.g. control rod positions) used to initiate the TRACG TTWBP analyses consistent with the equilibrium cycle conditions provided by AREVA?

b. What value of 'dtmax' (maximum allowed timestep size) was used in the TRACG TTWBP analyses presented in the SAR? Was this 'dtmax' value conservative for ATWS-I analyses? If not, please provide revised TRACG TTWBP analyses with an appropriate value for 'dtmax' to replace the analyses presented in the SAR.

Response SRXB-RAI-4a

A hot rod is modeled with an additional rod group in the channel component selected to represent the hot bundle (highest radial power, highest harmonic, and highest hybrid power at the symmetrical section of the core). The hot rod (i,j) location in a bundle is **[[ ]]**. The hot rod relative power is selected such that in steady state at the intended core state point the peak LHGR of the hot rod is not less than **[[ ]]**, which is **[[ ]]** of the maximum LHGR limit. The rod relative power inputs determined for the hot rod are applied to all other TRACG channel groups.

Response SRXB-RAI-4b

A dtmax of **[[ ]]** was used in the TRACG TTWBP analyses presented in the SAR, but it is more appropriate to use a dtmax of **[[ ]]** per the GEH ATWS-I process. It is set **[[ ]]**. BSEP TRACG ATWS-I cases were re-analyzed with a dtmax of **[[ ]]** and the updated SAR results were provided in Reference SRXB-4-1. For the updated analysis, the core conditions and control rod pattern used to initiate the TRACG TTWBP analyses was consistent with the equilibrium cycle conditions provided by AREVA.

The updated associated paragraphs in SAR Section 9.3.3, and the updated Table 9-6, Figure 9-10 and Figure 9-11 were provided to the NRC in Reference SRXB-4-1.

Reference

SRXB-4-1. Letter from William R. Gideon (Duke Energy) to the NRC Document Control Desk, "Updates to Request for License Amendment Regarding Core Flow Operating Range Expansion (CAC Nos. MF8363 and MF8364)," BSEP 17-0089, November 1, 2017.

NRC SRXB-RAI-5

The following additional information is requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide a diagram of the TRACG channel grouping used for the regional and core-wide mode TTWBP ATWS-I analyses. Did the regional mode analyses result in higher PCT values than the core-wide analyses?

b. Figure 9-10 of the SAR provides total core values for neutron flux and inlet flow rate versus time; however, this does not sufficiently describe the local assembly behavior during regional mode oscillations. Please provide additional time-dependent results at symmetric core locations (including the limiting Peak Cladding Temperature (PCT) assembly) showing the amplitude of the regional oscillations for the revised TTWBP analyses requested in SRXB-RAI-4. Please include time-dependent assembly power, assembly inlet flow rate, and maximum cladding temperature for each of these two assemblies, as well as the axial location where the maximum PCT occurs.

Response SRXB-RAI-5a

Figure SRXB-5-1 and Figure SRXB-5-2 are the diagrams of the TRACG channel grouping used for the regional and core-wide modes for Turbine Trip with Bypass (TTWBP) Anticipated Transient Without Scram with Instability (ATWS-I) analyses at the Beginning-of-Cycle (BOC). The regional mode analyses result in a higher PCT value than the core-wide mode analyses.

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Figure SRXB-5-1 BOC TTWBP ATWS-I TRACG Channel Grouping (Core Wide Mode)

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Figure SRXB-5-2 BOC TTWBP ATWS-I TRACG Channel Grouping (Regional Mode)

Response SRXB-RAI-5b

Figure 9-10 in the MELLLA+ Safety Analysis Report (SAR) shows the core key parameters from ATWS-I TTWBP case results at BOC with Regional Mode. The local assembly behaviors are plotted for the limiting PCT assembly (i.e., CHAN116) and its symmetric CHAN115 (i.e., see Figure SRXB-5-3 and Figure SRXB-5-4.). The plots include time-dependent assembly power, assembly inlet flow rate, and maximum cladding temperature for each of these two assemblies.
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Figure SRXB-5-3 BOC TTWBP ATWS-I TRACG Assembly Response (Regional Mode)

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Figure SRXB-5-4 BOC TTWBP ATWS-I TRACG Assembly Response (Regional Mode)-Blowup Plot

NRC SRXB-RAI-6

Please provide the following information regarding the limiting fuel parameter sensitivities to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide the process used to determine the parameters used for the ATRIUM 10XM fuel parameter sensitivity study for the ATWS and ATWS-I analyses for BSEP MELLLA+. For each parameter, please provide information on how the sensitivity range was determined.

b. Please provide a table showing the maximum PCT value for the TRACG TTWBP analyses (including the revisions described in SRXB-RAI-4, if necessary) obtained by adjusting each fuel parameter individually to the minimum and maximum value within the appropriate sensitivity range to demonstrate each parameter's impact on the results.

Response SRXB-RAI-6a

The process used to determine the parameters in the ATRIUM 10XM fuel parameters sensitivity study involved creating the Phenomena Identification and Ranking Table (PIRT) that is used in the Code, Scaling, Applicability and Uncertainty (CSAU) methodology. First, the PIRT was developed by GE experts to identify and rank all phenomena that govern Anticipated Transient Without Scram (ATWS) responses. From these parameters, the highly ranked fuel-related parameters were down selected. From these, each parameter was screened to identify the fuel parameters that have an impact on ATWS application results. Any parameter that (1) did not have appreciable impact, or (2) are independent of the fuel type, were not considered further. The final list was examined by an expert review committee, and provided to Framatome Inc. from GEH for further investigation through a Design Input Request (DIR). This request provided the parameters for which bounding ranges were to be determined. In some cases requests for minimum and maximum values were made, and in others a suggested range was given.

The sensitivity ranges were determined as follows:

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Response SRXB-RAI-6b

Table SRXB-6-1 provides the fuel parameter impact by adjusting each fuel parameter individually to the minimum and maximum value within the appropriate sensitivity range. The fuel parameter bounding case is run with each fuel parameter combined and each is set at the end of the range that is the worst for Peak Cladding Temperature (PCT).

The void sensitivity base case **[[**
]]. The base case for the rest of the sensitivities **[[**

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The bounding case (Case 14) has a PCT of **[[ ]]**. This PCT is **[[ ]]** greater than the nominal value. The PCT is still well below 2,200°F. Because the bounding void sensitivity impact is **[[ ]]**, which is less than **[[ ]]**, void sensitivity impact is not included in the bounding PCT.

The fuel parameters shown in Table SRXB-6-1 are defined as below. **[[**

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Table SRXB-6-1 Fuel Parameter Impact

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				]]

Reference

SRXB-6-1 GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis," NEDE-32906P-A Revision 3, September 2006.

NRC SRXB-RAI-7

Based on recent NRC funded ATWS-I test experiments (KATHY), the failure to rewet (FTR) temperature is in reasonable agreement with homogeneous nucleation plus contact temperature and provides a reasonable representation of the cladding temperature behavior during ATWS-I oscillations. The homogeneous nucleation plus contact temperature is more conservative compared to Modified Shumway T_{min} model used in the SAR. Therefore, the following sensitivity studies are requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event:

a. Please provide TRACG sensitivity studies for the TTWBP event in BSEP using the homogeneous nucleation temperature plus contact temperature model for T_{min}. Sensitivity studies may include realistic assumptions for input parameters such as feedwater temperature versus time, operator action time to reduce water level, maximum initial LHGR value, and use of the TRACG quench model. Please also provide sensitivity studies using the limiting fuel parameter sensitivity values in conjunction with the homogeneous nucleation T_{min} model.

b. In the following paragraph, "limiting fuel parameter delta-PCT" refers to the PCT when using the limiting fuel parameter values minus the PCT when using nominal fuel parameter values. If the limiting fuel parameter delta-PCT in any of the cases performed in (a) was larger than the limiting fuel parameter delta-PCT reported in the SAR, please explain why the limiting fuel parameter delta-PCT was larger. This explanation may include consideration of the location of maximum PCT, the average or peak LHGR at this location, or other considerations as appropriate.

Response SRXB-RAI-7a

Based on the updated Safety Analysis Report (SAR) Turbine Trip with Bypass (TTWBP) case described in SRXB-RAI-4, sensitivity studies were performed using the TRACG homogeneous nucleation T_{min} correlation and realistic assumptions for input parameters such as feedwater temperature versus time and operator action time to reduce water level. For all cases with and without these sensitivities, the hot rod initial Linear Heat Generation Rate (LHGR) peaking is set to **[[ ]]** of the limiting LHGR value of 14.10 kW/ft, and the TRACG quench model is used. The nominal (realistic) feedwater temperature reduction rate is based on BSEP plant data and decreases at a slower rate of 0.5°F/second as compared to the licensing value of 1.3°F/second. The operator action time to reduce water level is reduced from 120 seconds to 96 seconds in one of the sensitivity cases. Sensitivity studies were also performed using the limiting fuel parameter sensitivity values in conjunction with the homogeneous nucleation T_{min} model and the above realistic input assumptions. PCT results are provided in Table SRXB-7-1 and Figures SRXB-7-1 (a and b) through SRXB-7-3 (a and b).

The following abbreviations are used in the Table SRXB-7-1 case names:

"TT" -Turbine Trip with Bypass

"HNT" -TRACG Homogenous Nucleation T_{min} Model

"NFWTR" -Nominal Feedwater Temperature Reduction Rate

"WL96" -Water Level Reduction Delay Time of 96 seconds

"B" -Bounding Fuel Parameter Uncertainty Included

[[

]]. This T_{min} correlation is used in the following sensitivity analysis. This correlation, as applied in TRACG, is defined in Equation (6.6-51) in Section 6.6.7 of the TRACG Model Description (i.e., Reference SRXB-7-1). The equation is

$$T_{min} = T_c + \left(T_c - T_\ell\right)\left(\frac{\rho_\ell k_\ell C_{p,\ell}}{\rho_w k_w C_{p,w}}\right)^{1/2}$$

where T_c is the critical temperature for water. **[[**

]]

The effect of the sensitivity cases on the output parameters of interest is consistent with the expected effect of the changes. **[[**

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Response SRXB-RAI-7b

The SRXB-RAI-4 response shows that the limiting fuel parameter delta-PCT is **[[ ]]**. When the TRACG homogeneous nucleation T_{min} model is used together with the nominal feedwater temperature reduction rate, the limiting fuel parameter delta-PCT is increased. The main reason for the larger delta-PCT is because T_{min} was exceeded when using the Homogeneous Nucleation T_{min}. After T_{min} is exceeded, the hot rod surface does not return to nucleate boiling during the flow oscillations and surface temperatures rapidly increase to high temperatures. The same values of the fuel-parameter uncertainties applied to the cases with temperatures above T_{min} are expected to cause a larger impact than the cases with temperatures below T_{min}. For example, **[[**

]]

References

SRXB-7-1 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

SRXB-7-2 GE Nuclear Energy, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047-A, June 1995.

Table SRXB-7-1 TTWBP-ATWS-I T_{min} Sensitivity PCT Results

Case Name	Transient FW Temperature Response	Operator Action Time to Reduce Water Level (sec)	PCT(K/F)
TT-HNT-NFWTR	0.5°F/sec	120	**[[**
TT-HNT-NFWTR-WL96	0.5°F/sec	96	
TT-HNT-NFWTR-B	0.5°F/sec	120	**]]**

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Figure SRXB-7-1(a) Key Parameter Responses – Case TT-HNT-NFWTR

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Figure SRXB-7-1(b) PCT Results – Case TT-HNT-NFWTR

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Figure SRXB-7-2(a) Key Parameter Responses – Case TT-HNT-NFWTR-WL96

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Figure SRXB-7-2(b) PCT Results – Case TT-HNT-NFWTR-WL96

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Figure SRXB-7-3(a) Key Parameter Responses – Case TT-HNT-NFWTR-B

[[

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Figure SRXB-7-3(b) PCT Results – Case TT-HNT-NFWTR-B

NRC SRXB-RAI-8

Please justify the feedwater temperature reduction rate that was used in the ATWS-I TTWBP analyses presented in the SAR, as well any reduced rate used in the ATWS-I sensitivity studies to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event. Justify this using appropriate plant data or simulator results, if available. If plant simulator data is provided, please justify the adequacy of the simulator for determining the feedwater temperature reduction rate.

Response

GEH assumed a feedwater temperature reduction rate of 1.3°F/sec in the ATWSI TTWBP analyses presented in the SAR. Based on plant data, this was considered a reasonable and conservative assumption for the ATWSI analyses. Further review of plant data confirmed that a temperature reduction rate of 0.5°F/sec was supported and remained conservative.

As reported in response to SRXB-RAI-7, TRACG sensitivity studies were performed using a feedwater temperature reduction rate of 0.5°F/sec. The sensitivity studies and the SAR analyses also modeled the plant-specific water level control strategy. For BSEP, this strategy requires that the "terminate and prevent" actions specified in the Emergency Operating Procedures (EOPs) are completed within 120 seconds. For the TTWBP event, the feedwater system continues to provide makeup to the reactor until terminated by plant operators at 120 seconds. For this reason, plant data is used to justify this analysis assumption of 0.5°F/sec. during this 120 second period.

Feedwater Temperature Change per unit Time

The feedwater temperature data collected following several BSEP turbine trip events was used to assess the rate of feedwater temperature reduction. The events selected were those where the majority if not all of the inventory following each plant trip was supplied to the reactor by the feedwater system. When the main turbine trips (i.e., typically at the beginning of each transient), the heat input to the feedwater heating system essentially stops. Since a large amount of hot feedwater and thermal energy remains trapped in the feedwater heaters themselves, the feedwater system will continue to provide heated feedwater following a turbine trip.

Plots of the relevant parameters for each plant transient are presented below. Each plot also includes a hypothetical line representing a temperature decrease of 0.5°F/sec and a vertical line marking the time of the turbine trip.

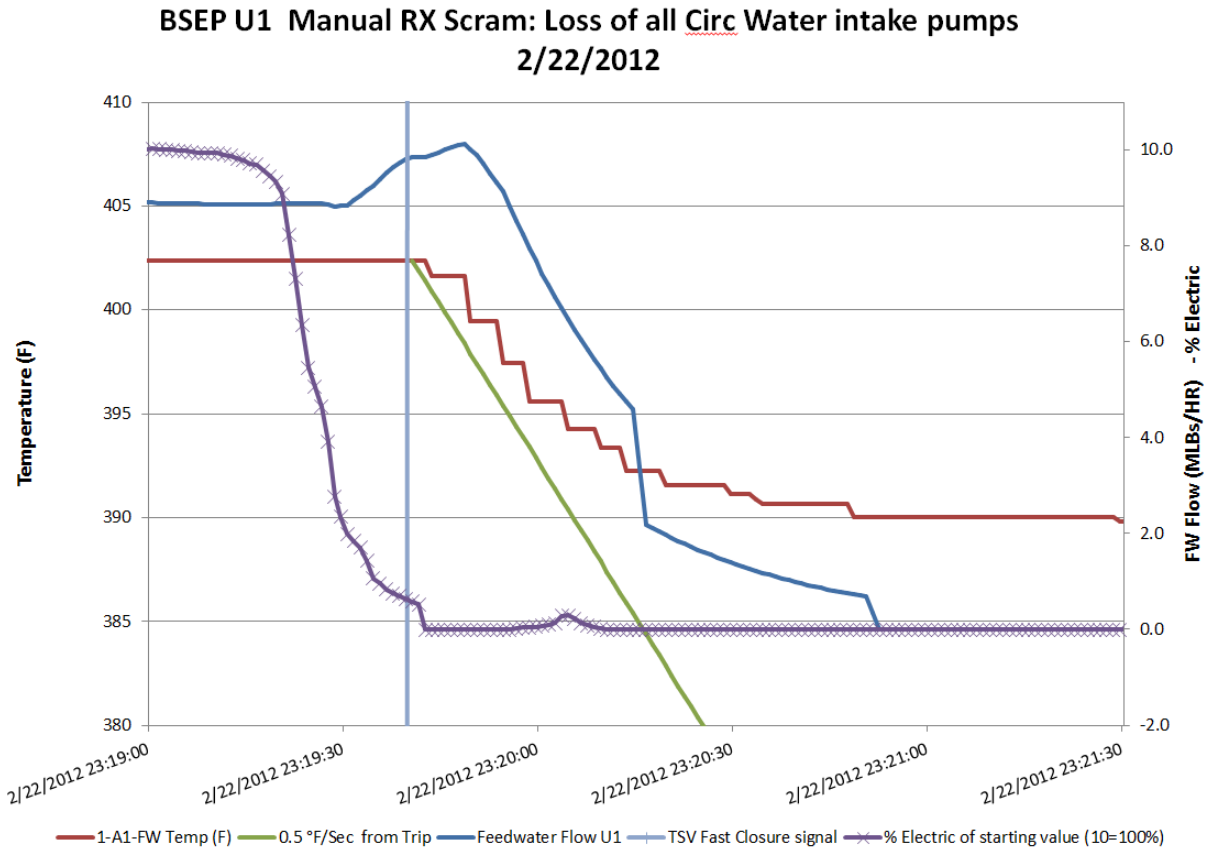

Figure SXRB-8-1 From ~70% Power

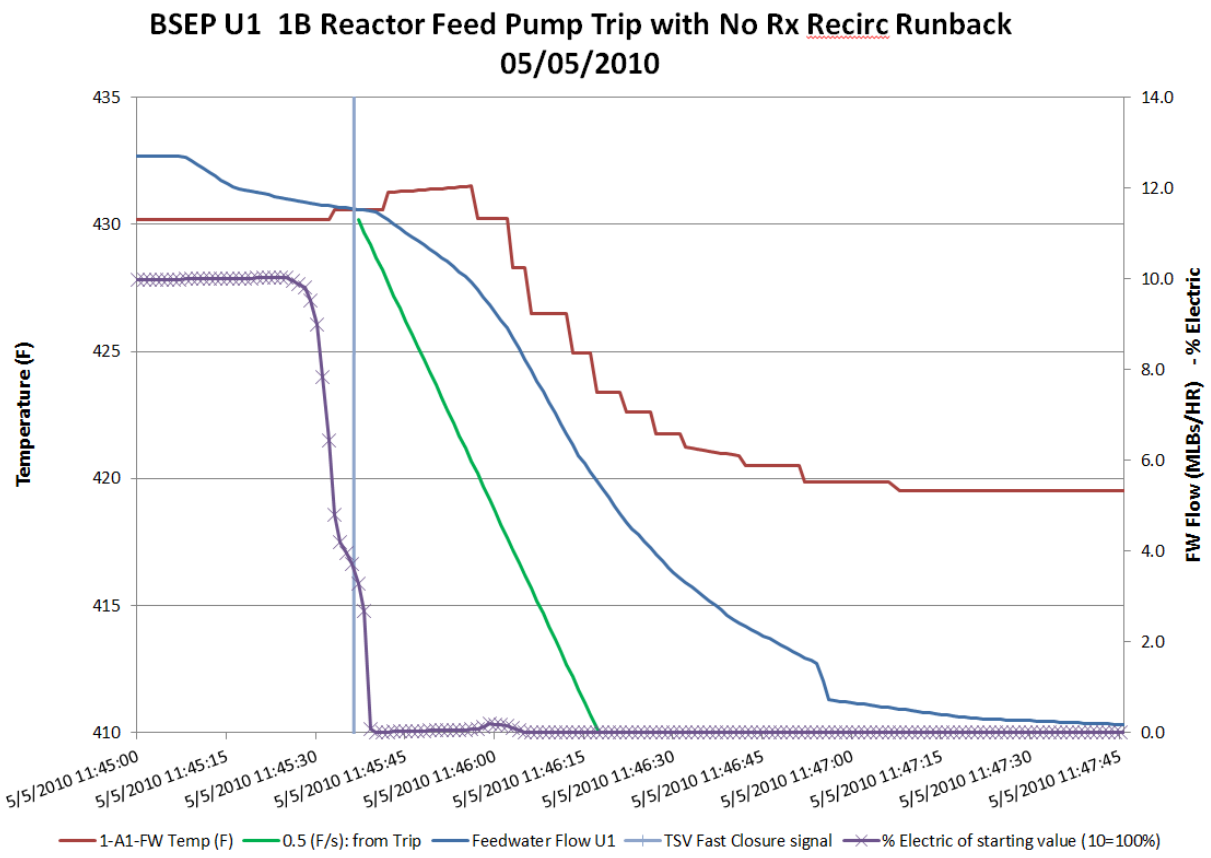

Figure SXRB-8-2 From ~100% Power

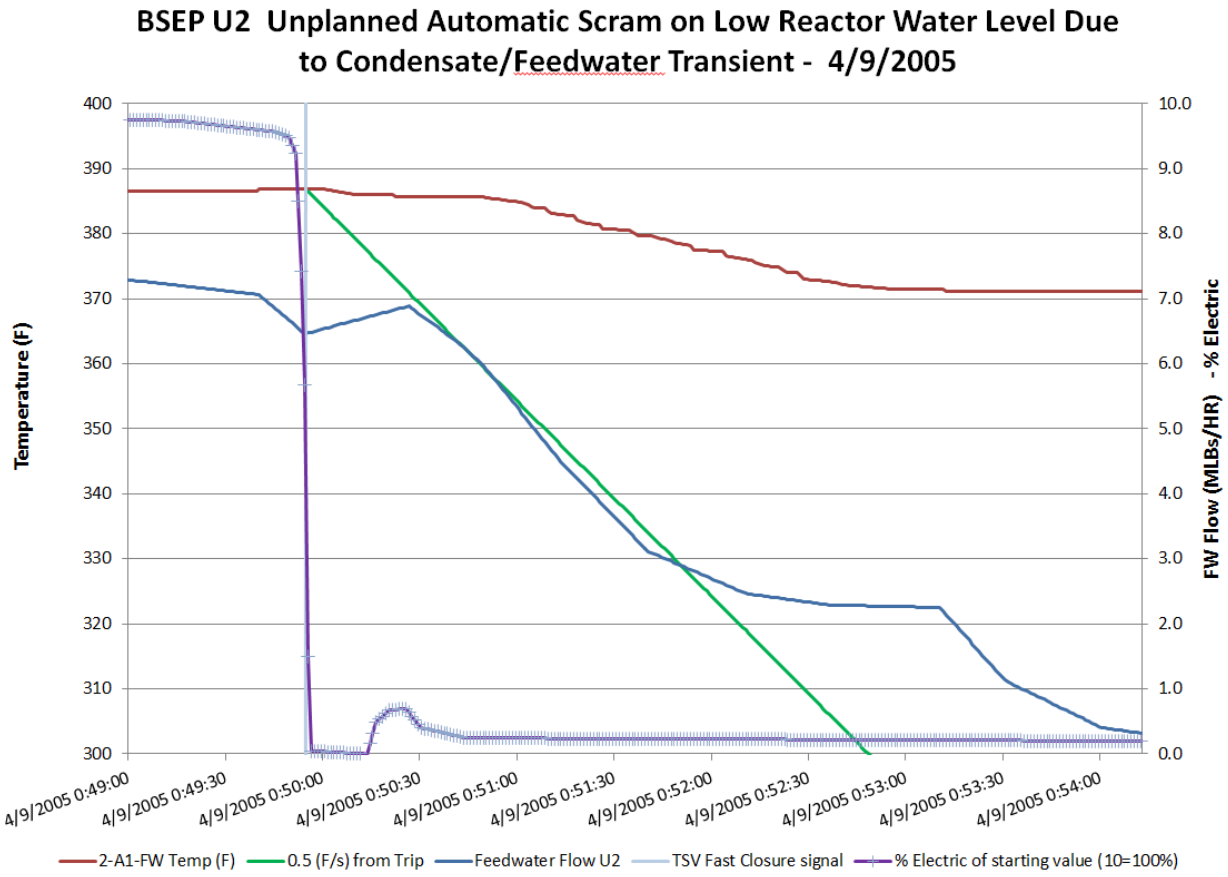

Figure SXRB-8-3 From 57% Power

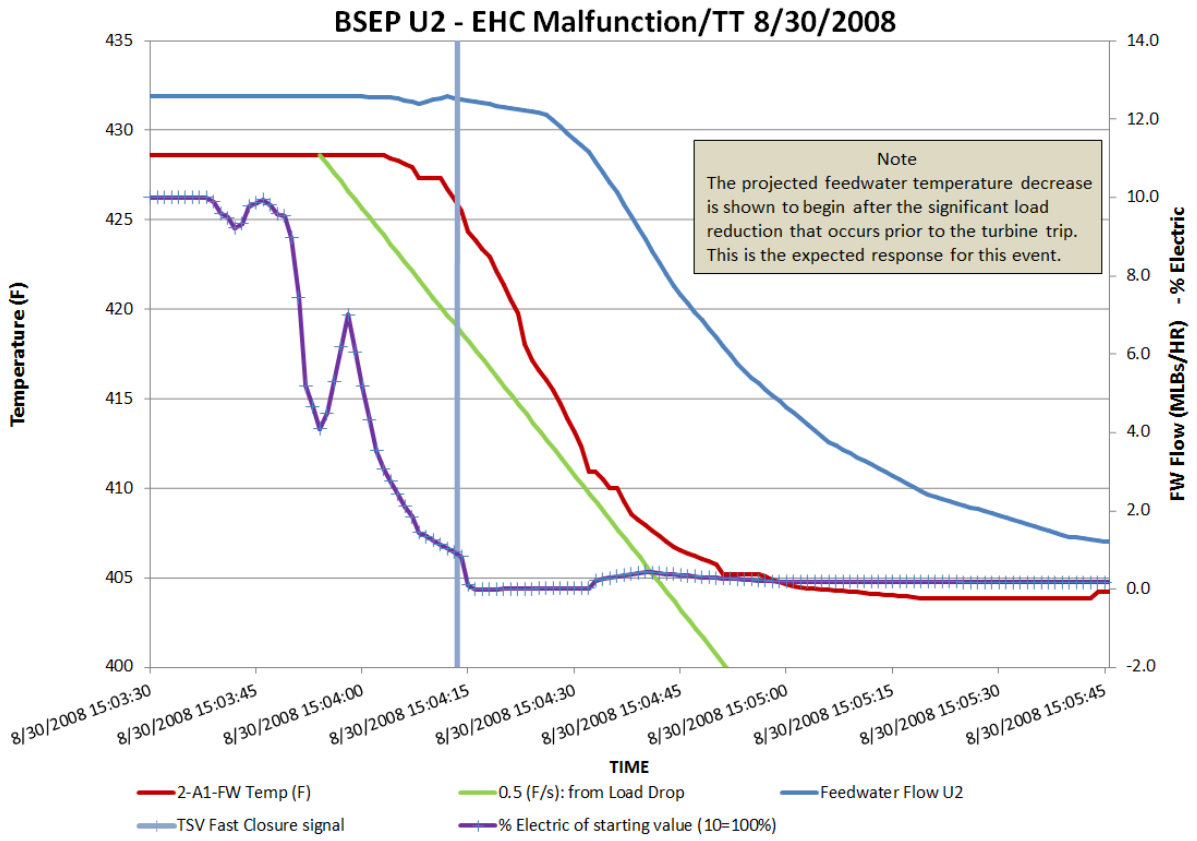

Figure SXRB-8-4 From 100% Power

The data presented in Figure SXRB-8-1 through Figure SXRB-8-4 demonstrate that the actual drop in feedwater temperature is less than the 0.5°F/sec. Thereby showing that an assumed feedwater temperature of 0.5°F/sec for ATWS analyses is conservative.

The event illustrated in Figure SXRB-8-4 was not as straight forward as the other events and deserves some additional discussion. This event is from a turbine electro-hydraulic control (EHC) system malfunction that led to a plant scram. Since this represents a corresponding drop in heat input to the feedwater heating system, the start of the 0.5°F/sec temperature line is placed just after the first significant turbine load decrease. For a brief period the rate of feedwater temperature reduction exceeded the 0.5°F/sec rate. This response is not surprising given the significant transient occurring with the secondary system. The initial feedwater temperature (i.e., 427°F) dropped to a relatively steady 404°F in approximately 100 seconds from the initial load drop. The overall temperature drop of approximately 25°F is significantly less than the 60°F temperature drop assumed in the ATWS-I sensitivity analyses (i.e., 0.5°F/sec over a two minute period). As shown in the next section, feedwater temperature as a function of feedwater mass during this event is consistent with other BSEP events.

Feedwater Temperature Change per unit Mass

During a high power ATWS event, the reactor will continue to operate at a reactor power as high as 60% of rated after the recirculation pumps are tripped. As operators implement the EOP reactor water level strategy during the first 120 seconds, a significant volume of feedwater is expected to be pumped to the RPV. This is in contrast to a plant scram from high power where the demand for feedwater is low. To provide an additional means for comparison, the temperature data from Figures SXRB-8-1 through SXRB-8-4 were plotted as a function of the mass pumped to the RPV. The goal of the mass flow versus temperature comparison is to establish a measurement of feedwater system heat removal that is independent of the rate that heat is removed from the system thereby providing a direct comparison between plant events and a hypothetical ATWSI scenario.

Based on rated feedwater flow (i.e., 12.755 Mlbm/hr) and assuming 60% power (i.e., from the TRACG analysis), the mass of feedwater required during the first 120 seconds of a high-power ATWS event is just over 250,000 lbm. Figure SXRB-8-5 shows the data from these events depicted in Figures SXRB-8-1 through SXRB-8-4 and shows the change in feedwater temperature on a per unit mass basis.

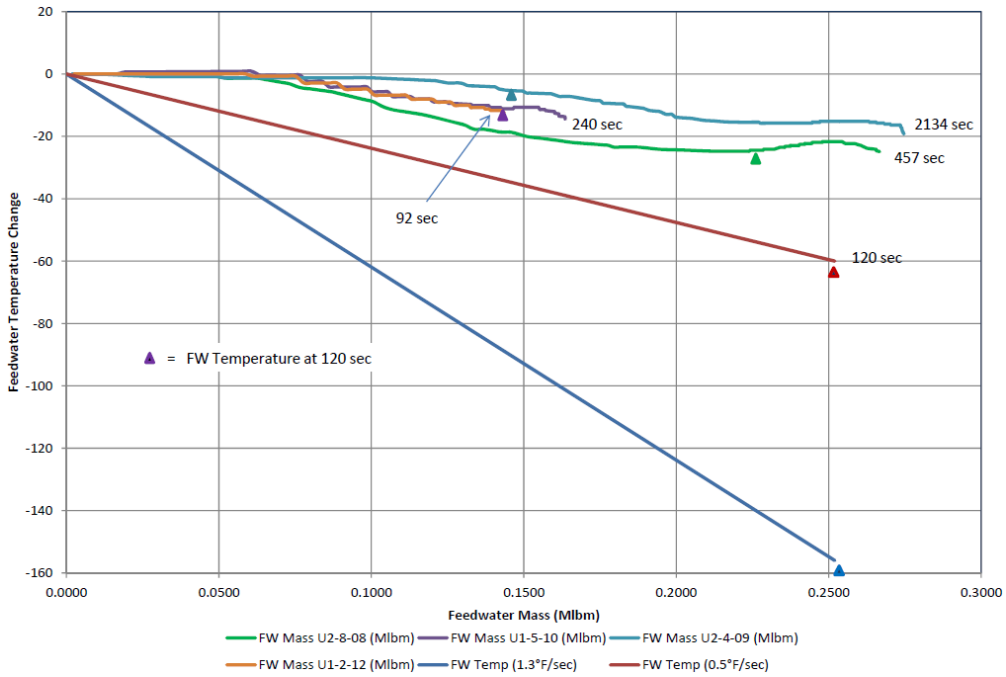

Figure SXRB-8-5 Feedwater Temperature Drop vs Mass

Observations from Figure SXRB-8-5:

- A significant volume of heated feedwater is available in the feed system such that there is not an immediate drop in feedwater temperature. For the data reviewed, feedwater

temperature remains within 5°F of the initial temperature until approximately 80,000 lbm is delivered to the vessel.

- The reduction in feedwater temperature associated with a pumped mass of 250,000 lbm after a turbine trip from high power conditions (i.e., >60% CTP) is expected to be less than 30°F in the first 120 seconds.

This approach provides a measure of feedwater temperature reduction that is independent of the rate that heat is removed from feedwater system thereby providing a direct comparison between plant events and a hypothetical ATWS-I event.

In conclusion, the reduction in feedwater temperature assumed in the ATWSI analyses of 0.5°F/sec (i.e., 60°F in 120 seconds) starting at the turbine trip bounds the drop observed in the plant data. Therefore, a feedwater temperature reduction rate of 0.5°F/sec or 1.3°F/sec combined with an operator action time of 120 seconds to terminate and prevent RPV injection are appropriate assumptions for ATWS-I TTWBP event analysis.

NRC SRXB-RAI-9

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the staff has the following questions regarding the use of the GEXL correlation for ATWS-I analyses:

a. Please discuss the GEXL correlation's applicability to oscillatory conditions.

b. For ATWS-I oscillations such as those calculated in the TTWBP analyses for BSEP MELLLA+, is the critical heat flux temperature (in the oscillation growth phase and in the limit cycle phase) during a given oscillation period typically determined by **[[**
]].

Response SRXB-RAI-9a

The GEXL correlation is used to determine the occurrence of boiling transition. For TRACG ATWS-I calculations, this is no different from its planned use and is consistent with the approved DSS-CD application with TRACG during oscillatory conditions (i.e., Reference SRXB-9-1). GEXL correlations developed for GE/GNF fuel products since GE9 have been assessed against oscillatory test data to confirm their performance in predicting the onset of boiling transition and the return to nucleate boiling.

Response SRXB-RAI-9b

Section 6.6.6 of the TRACG Model Description (Reference SRXB-9-2) describes the correlations used to determine when transition boiling occurs and how a rod returns to nucleate boiling. When GEXL applies (see Section 6.6.6 of Reference SRXB-9-2), transition boiling occurs **[[**

]]. Film boiling will always occur in TRACG if the minimum stable film boiling temperature (T_{min}) is exceeded regardless of the critical heat flux temperatures and critical qualities. Section 6.6.8 of the TRACG Model Description (i.e., Reference SRXB-9-2) describes the interpolation of heat transfer coefficients (HTCs) when the clad surface temperature is above T_{CHF} and below T_{min}.

The GEXL correlation does not determine the HTCs but does get used in determining the boundaries between correlations that are used to calculate HTCs. After a rod enters transition or film boiling, the GEXL correlation is only used indirectly as described below.

To return to nucleate boiling, the following three criteria apply:

1) **[[** **]]**
2) **[[** **]]**
3) **[[** **]]**

References

SRXB-9-1 GE Hitachi Nuclear Energy, "DSS-CD TRACG Application," NEDE-33147P-A Revision 4, August 2013.

SRXB-9-2 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P Revision 4, January 2008.

NRC SRXB-RAI-10

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the staff requests the following additional information regarding the use of R-factors in the TRACG ATWS-I analyses:

a. Please provide the R-factors used in each assembly in the TRACG TTWBP models, and justify how these R-factors were determined. If applicable, please provide TTWBP sensitivity results that indicate the effect that changing the GEXL critical power value has on maximum PCT during this event.

b. The stated range of validity of R-factors in the GEXL97 correlation for ATRIUM-10 fuel is **[[** **]]**. If an R-factor less than **[[** **]]** was used in the ATWS-I analyses, please justify the use of these ATRIUM 10XM R-factors outside of the stated range of validity. Does the GEXL correlation behave properly for R < **[[** **]]**? (For example, is the **[[** **]]** term intended to be used for R < **[[** **]]**?)

c. For safety analyses such as anticipated operational occurrences (AOOs), higher hot- rod R-factor values are typically more limiting. If a lower R-factor value was found to be more limiting in Part (a), please explain why it was more limiting or please provide additional sensitivity analyses using the following assumptions for assembly R-factors, to better understand the limiting combination of R-factors for the TTWBP event:

   1. Hot channel R-factor taken from bounding high value from AREVA inputs, and remaining R-factors taken as the low value

   2. Hot channel R-factor taken as the low value, and remaining R-factors taken as the bounding high value

   3. All R-factors taken as the low value

   4. All R-factors taken as the bounding high value

Response SRXB-RAI-10a

R-factors slightly lower than the specified applicability range of the GEXL97 correlation (Reference SRXB-10-1) are used: **[[ ]]** for each of the **[[ ]]** single channels and **[[ ]]** for the rest of the channel groups. Peak Cladding Temperature (PCT) results from the sensitivity analyses performed with the GEXL critical power uncertainty range **[[ ]]** show that that PCT impact is insignificant. A lower critical power corresponds to a higher R-factor and vice versa. For the lower critical power associated with the higher R-factor, the calculated PCT decreases slightly so use of the lower **[[ ]]** R-factor, which is slightly below the range of R-factors provided for ATRIUM 10, is slightly more conservative for the BSEP MELLLA+ ATWS-I analysis.

Response SRXB-RAI-10b

R-factor is used as an input for the GEXL correlation which is designed for a typical range of R-factors **[[ ]]**. There are no issues with an R-factor input less than **[[ ]]** that is within the specific design range for the correlation. GEXL calculates a critical quality based on local conditions and predicts the onset of Boiling Transition (BT). The R-factors used in GEXL evaluations have a minor impact on the calculated onset of BT which results in a minor impact on the calculated PCT results. The GEXL role in calculation of rewet is discussed in response to SRXB-RAI-9. Even though a lower R-factor of **[[ ]]** delays slightly the predicted BT onset, the sensitivity studies show that this results in a slightly higher calculated PCT. The sensitivity study shows that the average bundle R-factors have a small impact on the calculated PCT results.

Response SRXB-RAI-10c

For the BSEP MELLLA+ ATWS-I analysis, the GEXL critical power sensitivity shows that the higher initial Critical Power Ratio (CPR), (i.e., which should correspond to lower R-factor) causes a slightly higher PCT. However, the variation is very insignificant (around **[[ ]]**).

There is no discernible trend relating R-factor and GEXL critical power uncertainty with such minor changes in calculated PCT; therefore, no additional R-factor sensitivity studies are necessary because the sensitivities are already covered by the **[[ ]]** critical power sensitivity studies.

Reference

SRXB-10-1 GE Hitachi Nuclear Energy, "GEXL97 Correlation Applicable To ATRIUM-10 Fuel," NEDC-33383P, Revision 1, June 2008.

NRC RAI SRXB-RAI-11

Please provide steady-state core simulator comparisons for a representative BSEP MELLLA+ cycle using GEH and AREVA methods, to support the licensee's conclusion that the GEH methods have modeled ATRIUM 10XM in a satisfactory manner. Include comparisons of calculated results such as power and exposure distributions, hot eigenvalue, active or bypass flow rates, and core pressure drop.

Response

The requested steady-state core simulator comparisons for the representative BSEP MELLLA+ Equilibrium (EQ) cycle is provided in the subsequent figures to demonstrate the GEH/GNF methods using the NRC approved TGBLA06 lattice physics code and PANAC11 (P11) Three-Dimensional (3D) core simulator code have modeled the AREVA ATRIUM 10XM (A10XM) fuel and core characteristics at BSEP in a satisfactory manner.

The One Dimensional (1D) data series versus cycle exposure for the GEH/GNF P11 methods is shown as a blue line on the figures included in this RAI response. The 1D data series versus cycle exposure for AREVA MICROBURN (MB2) methods is shown as a red line on the figures. The 1D data series versus cycle exposure for the difference between methods is shown as black lines on the figures. The Two Dimensional (2D) power and exposure radial distributions and the 1D power and exposure profiles versus axial node are shown similarly as GEH/GNF P11 and AREVA MB2, and the difference between methods is shown as separate figures.

The GEH/GNF P11 hot eigenvalue is shown in Figure SRXB-11-1. The AREVA MB2 hot eigenvalue is shown in Figure SRXB-11-2. The hot eigenvalue difference between core methods is shown in Figure SRXB-11-3. It should be noted that eigenvalue comparisons between core methods are expected to be significantly different because of fundamental differences in the methods. Normally, eigenvalue comparisons are made between different fuel product lines with the same method at the same plant or the same fuel product line and method with a similar plant. As shown in Figure SRXB-11-4, the **[[**

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The GEH/GNF P11 active channel flow is shown in Figure SRXB-11-5. The AREVA MB2 active channel flow is shown in Figure SRXB-11-6. The active channel flow difference between core methods is shown in Figure SRXB-11-7. The GEH/GNF P11 bypass flow (which includes water

rod flow) is shown in Figure SRXB-11-8. The AREVA MB2 bypass flow (which includes water rod flow) is shown in Figure SRXB-11-9. The bypass flow difference between core methods is shown in Figure SRXB-11-10. The GEH/GNF P11 core pressure drop is shown in Figure SRXB-11-11. The AREVA MB2 core pressure drop is shown in Figure SRXB-11-12. The core pressure drop difference between core methods is shown in Figure SRXB-11-13. **[[**

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The radial power distributions at Beginning-of-Cycle (BOC) and End-of-Cycle (EOC) conditions are shown in Figure SRXB-11-14 and Figure SRXB-11-15, respectively, for GEH/GNF P11. The radial power distributions at BOC and EOC conditions are shown in Figure SRXB-11-16 and Figure SRXB-11-17, respectively, for AREVA MB2. The radial power distribution difference between core methods at BOC and EOC conditions is shown in Figure SRXB-11-18 and Figure SRXB-11-19, respectively. **[[**

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The GEH/GNF P11 radial exposure distributions at BOC and EOC conditions are shown in Figure SRXB-11-20 and Figure SRXB-11-21, respectively. The AREVA MB2 radial exposure distributions at BOC and EOC conditions are shown in Figure SRXB-11-22 and Figure SRXB-11-23, respectively. The radial exposure distribution difference between core methods at BOC and EOC conditions is shown in Figure SRXB-11-24 and Figure SRXB-11-25, respectively.

The GEH/GNF P11 axial power profile at BOC (0 MWd/MT cycle exposure) and End-of-Cycle Low Flow (EOCLF) (i.e., 17,700 MWd/MT cycle exposure) 100% power/85% flow All Rods Out (ARO) conditions before cycle extensions are shown in Figure SRXB-11-26 and Figure SRXB-11-27, respectively. The AREVA MB2 axial power profile at BOC and EOCLF conditions is shown in Figure SRXB-11-28 and Figure SRXB-11-29, respectively. The axial power comparison between core methods at BOC and EOCLF conditions is shown in Figure SRXB-11-30 and Figure SRXB-11-31, respectively. The GEH/GNF P11 axial exposure profile at BOC and EOCLF conditions is shown in Figure SRXB-11-32 and Figure SRXB-11-33, respectively. The AREVA MB2 axial exposure profile at BOC and EOCLF conditions is shown in Figure SRXB-11-34 and Figure SRXB-11-35, respectively. The axial exposure comparison between core methods at BOC and EOCLF conditions is shown in Figure SRXB-11-36 and Figure SRXB-11-37, respectively. **[[**

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**Figure SRXB-11-1 BSEP EQ ATRIUM 10XM
Hot Eigenvalue (GEH/GNF P11)**

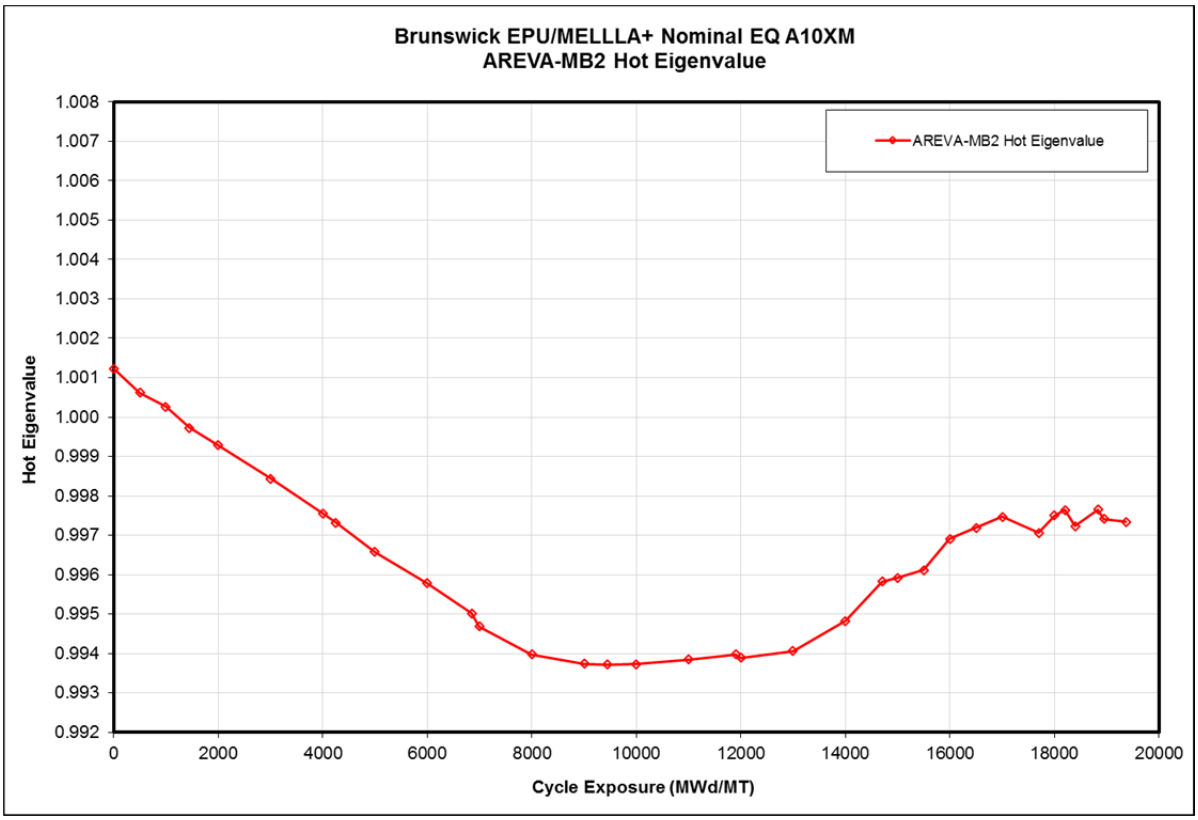

Figure SRXB-11-2 BSEP EQ ATRIUM 10XM Hot Eigenvalue (AREVA MB2)

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Figure SRXB-11-3 BSEP EQ ATRIUM 10XM
Hot Eigenvalue Difference

Note: Hot eigenvalues are not intended to be comparable between methods.

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Figure SRXB-11-4 BSEP EQ ATRIUM 10XM Hot Eigenvalue Comparison (ATRIUM 10XM versus GE14)

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Figure SRXB-11-5 BSEP EQ ATRIUM 10XM
Active Channel Flow (GEH/GNF P11)

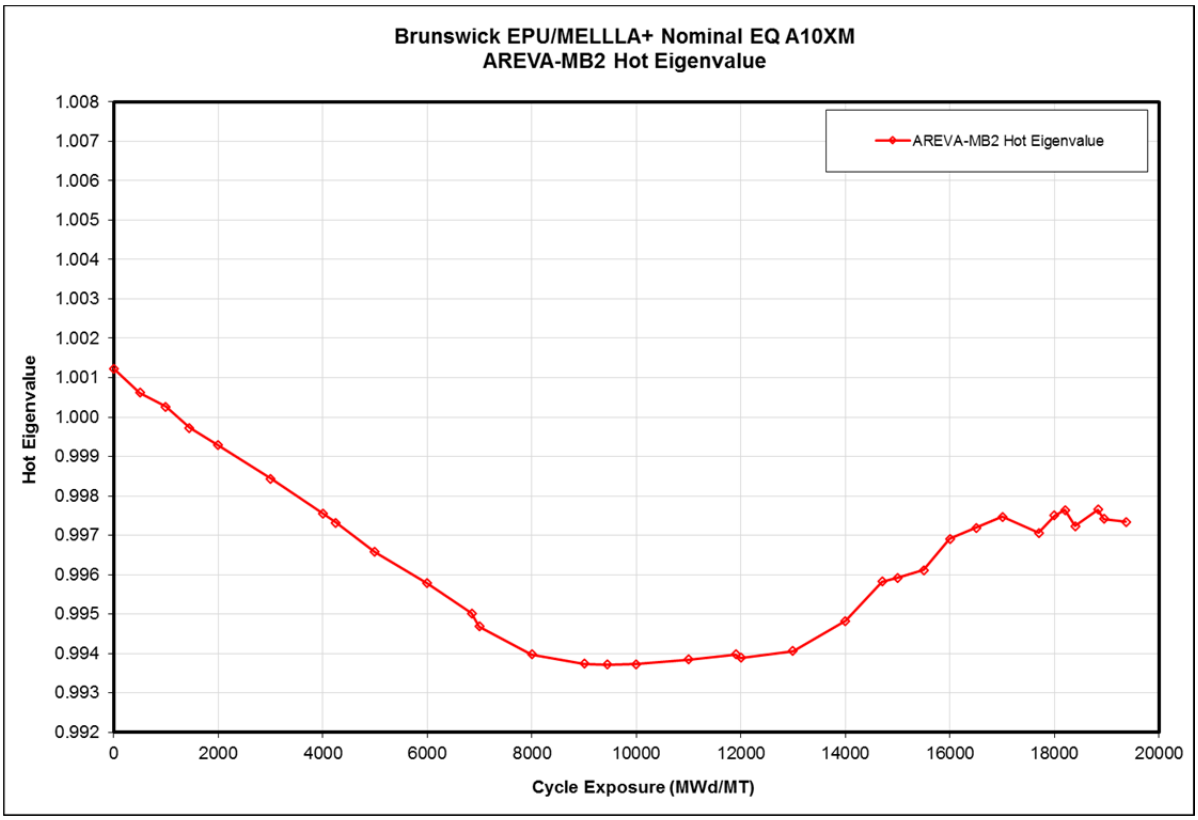

Figure SRXB-11-6 BSEP EQ ATRIUM 10XM Active Channel Flow (AREVA MB2)

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**Figure SRXB-11-7 BSEP EQ ATRIUM 10XM
Active Channel Flow Difference**

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Figure SRXB-11-8 BSEP EQ ATRIUM 10XM Bypass Flow (GEH/GNF P11)

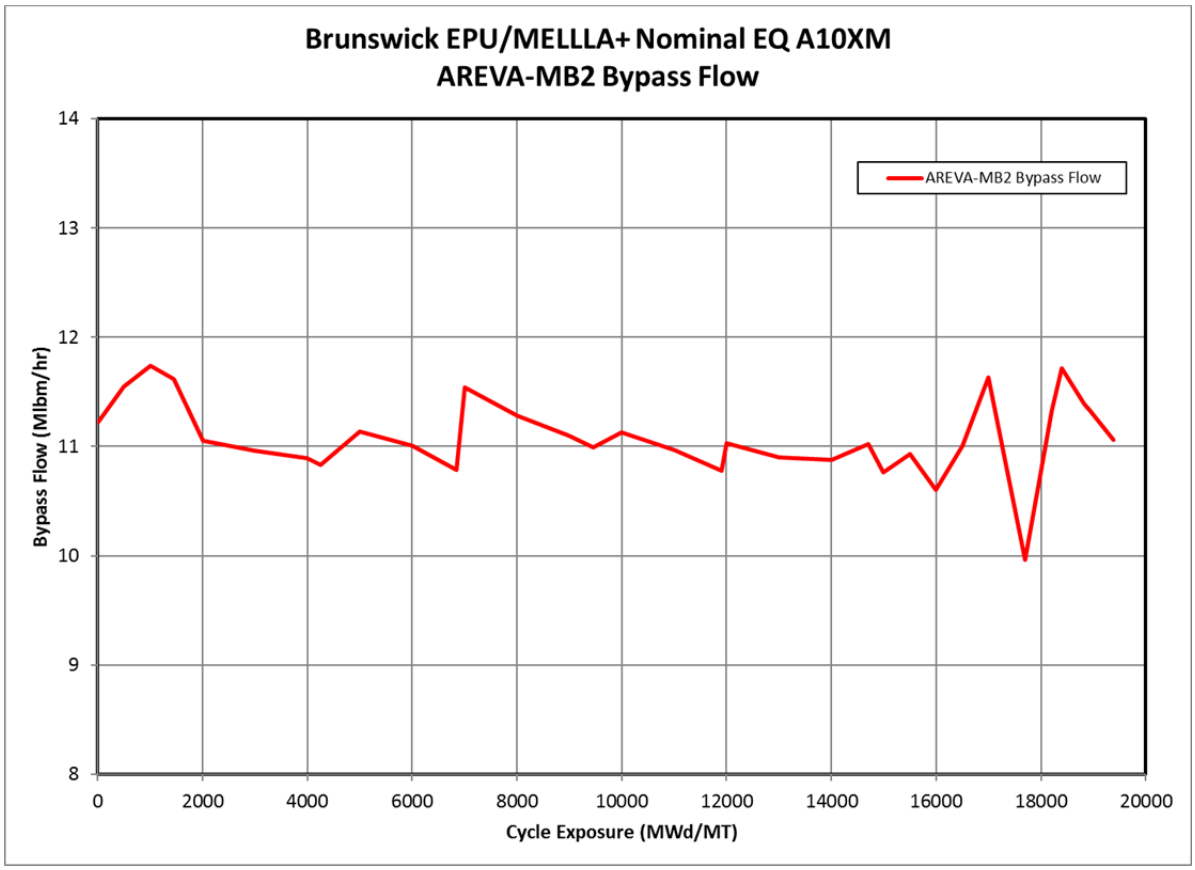

Figure SRXB-11-9 BSEP EQ ATRIUM 10XM Bypass Flow (AREVA MB2)

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Figure SRXB-11-10 BSEP EQ ATRIUM 10XM Bypass Flow Difference

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**Figure SRXB-11-11 BSEP ATRIUM 10XM
Core Pressure Drop (GEH/GNF P11)**

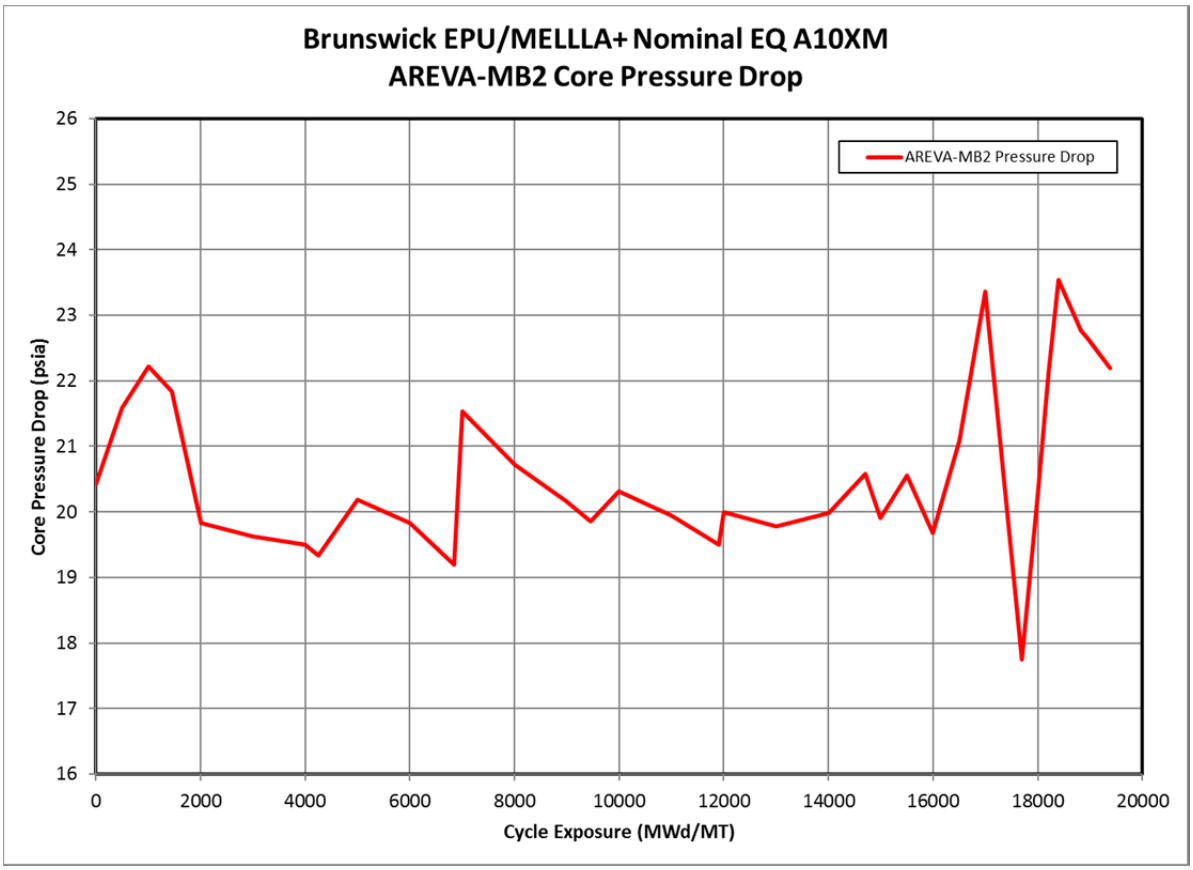

Figure SRXB-11-12 BSEP ATRIUM 10XM Core Pressure Drop (AREVA MB2)

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**Figure SRXB-11-13 BSEP ATRIUM 10XM
Core Pressure Drop Difference**

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Figure SRXB-11-14 BSEP ATRIUM 10XM BOC Radial Power Distribution (GEH/GNF P11)

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Figure SRXB-11-15 BSEP ATRIUM 10XM EOC Radial Power Distribution (GEH/GNF P11)

BOC Bundle Integrated Power

AREVA	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.293	0.340	0.401	0.440	0.466
2	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.408	0.526	0.608	0.792	0.827	0.842
3	0.000	0.000	0.000	0.000	0.344	0.431	0.559	0.672	0.866	0.938	0.993	1.019	1.027
4	0.000	0.000	0.000	0.396	0.684	0.811	0.909	0.993	1.055	1.093	1.099	1.127	1.084
5	0.000	0.000	0.349	0.688	0.763	0.991	1.084	1.125	1.094	1.087	1.188	1.151	0.964
6	0.000	0.000	0.461	0.816	0.993	1.097	1.143	1.223	1.118	1.157	1.212	1.221	1.001
7	0.000	0.000	0.564	0.913	1.086	1.142	1.237	1.230	1.261	1.235	1.289	1.256	1.257
8	0.000	0.412	0.674	0.995	1.126	1.223	1.231	1.269	1.237	1.254	1.270	1.314	1.274
9	0.296	0.529	0.868	1.055	1.094	1.117	1.261	1.243	0.926	0.998	1.274	1.275	1.218
10	0.343	0.623	0.939	1.093	1.086	1.157	1.235	1.255	0.997	1.002	1.266	1.326	1.209
11	0.400	0.794	0.994	1.098	1.187	1.210	1.291	1.278	1.276	1.269	1.331	1.328	1.335
12	0.439	0.826	1.019	1.126	1.149	1.221	1.265	1.316	1.284	1.327	1.324	1.337	1.296
13	0.465	0.840	1.026	1.082	0.962	0.995	1.257	1.274	1.219	1.205	1.334	1.291	0.925

Figure SRXB-11-16 BSEP ATRIUM 10XM BOC Radial Power Distribution (AREVA MB2)

EOC Bundle Integrated Power

AREVA	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.305	0.354	0.388	0.407
2	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.340	0.509	0.606	0.781	0.808	0.816
3	0.000	0.000	0.000	0.000	0.258	0.345	0.472	0.633	1.005	1.116	1.177	1.207	1.210
4	0.000	0.000	0.000	0.295	0.583	0.725	0.836	1.091	1.030	1.256	1.091	1.312	1.109
5	0.000	0.000	0.261	0.584	0.697	1.080	1.170	1.048	1.273	1.099	1.320	1.125	1.350
6	0.000	0.000	0.373	0.727	1.081	1.191	1.042	1.280	1.086	1.319	1.125	1.350	1.141
7	0.000	0.000	0.473	0.836	1.168	1.040	1.272	1.077	1.312	1.111	1.362	1.153	1.361
8	0.000	0.340	0.631	1.088	1.044	1.276	1.078	1.286	1.090	1.326	1.142	1.368	1.129
9	0.249	0.508	1.001	1.025	1.267	1.079	1.309	1.106	1.027	1.124	1.352	1.126	1.335
10	0.306	0.613	1.111	1.250	1.091	1.310	1.103	1.316	1.115	1.316	1.119	1.322	1.085
11	0.350	0.778	1.171	1.084	1.310	1.109	1.349	1.132	1.322	1.096	1.284	1.060	1.238
12	0.385	0.803	1.200	1.304	1.116	1.337	1.143	1.349	1.100	1.293	1.052	1.203	0.971
13	0.404	0.811	1.203	1.102	1.340	1.127	1.351	1.117	1.313	1.068	1.234	0.970	0.858

Figure SRXB-11-17 BSEP ATRIUM 10XM EOC Radial Power Distribution (AREVA MB2)

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Figure SRXB-11-18 BSEP ATRIUM 10XM BOC Radial Power Distribution Difference

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Figure SRXB-11-19 BSEP ATRIUM 10XM EOC Radial Power Distribution Difference

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Figure SRXB-11-20 BSEP ATRIUM 10XM BOC Radial Exposure Distribution (GEH/GNF P11)

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Figure SRXB-11-21 BSEP ATRIUM 10XM EOC Radial Exposure Distribution (GEH/GNF P11)

BOC Exposure (GWd/MT)

AREVA	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	45.293	45.777	44.787	42.956	40.079
2	0.000	0.000	0.000	0.000	0.000	0.000	0.000	45.773	37.492	37.953	19.822	21.659	21.867
3	0.000	0.000	0.000	0.000	45.731	45.006	37.807	35.053	0.000	0.000	0.000	0.000	0.000
4	0.000	0.000	0.000	45.826	17.932	20.173	21.005	0.000	19.770	0.000	23.966	0.000	24.127
5	0.000	0.000	45.828	18.001	34.925	0.000	0.000	21.827	0.000	24.703	0.000	24.146	0.000
6	0.000	0.000	37.950	20.232	0.000	0.000	24.587	0.000	24.816	0.000	24.654	0.000	24.684
7	0.000	0.000	38.024	21.074	0.000	24.691	0.000	24.744	0.000	25.305	0.000	24.548	0.000
8	0.000	45.658	35.162	0.000	21.865	0.000	24.681	0.000	24.001	0.000	24.361	0.000	24.938
9	45.380	37.584	0.000	19.840	0.000	24.850	0.000	22.141	30.176	23.073	0.000	25.630	0.000
10	45.886	35.079	0.000	0.000	24.704	0.000	25.311	0.000	23.133	0.000	24.182	0.000	24.728
11	44.236	19.862	0.000	23.999	0.000	25.043	0.000	24.035	0.000	24.125	0.000	23.940	0.000
12	43.017	21.715	0.000	0.000	24.207	0.000	24.191	0.000	25.204	0.000	24.329	0.000	23.456
13	40.186	21.913	0.000	24.195	0.000	25.196	0.000	25.063	0.000	25.151	0.000	24.031	30.301

Figure SRXB-11-22 BSEP ATRIUM 10XM BOC Radial Exposure Distribution (AREVA MB2)

EOC Exposure (GWd/MT)

AREVA	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	50.611	52.068	52.169	51.034	48.621
2	0.000	0.000	0.000	0.000	0.000	0.000	0.000	53.088	47.506	49.671	35.053	37.492	37.953
3	0.000	0.000	0.000	0.000	51.582	52.540	47.807	47.670	17.932	19.770	21.006	21.659	21.867
4	0.000	0.000	0.000	52.543	30.176	34.925	37.807	20.173	40.079	23.073	45.294	24.127	45.778
5	0.000	0.000	51.760	30.301	48.928	19.822	21.827	42.956	23.966	46.531	24.587	46.333	24.146
6	0.000	0.000	46.047	35.079	19.862	22.141	45.731	24.703	46.798	24.816	46.899	24.938	46.598
7	0.000	0.000	48.111	37.950	21.865	45.828	24.654	47.096	24.548	47.103	24.035	46.217	24.744
8	0.000	53.039	47.827	20.232	43.017	24.704	47.051	25.043	45.986	24.182	45.773	23.940	47.046
9	50.752	47.663	18.001	40.186	23.999	46.823	24.361	44.236	50.514	44.787	23.456	46.765	24.728
10	52.250	47.083	19.840	23.133	46.574	24.850	46.989	24.125	45.006	24.001	45.658	24.031	47.108
11	51.622	35.162	21.074	45.380	24.691	47.330	24.191	45.826	24.684	46.490	25.305	46.785	25.630
12	51.112	37.584	21.715	24.195	46.483	25.063	46.089	24.329	47.474	25.196	47.240	25.311	46.194
13	48.734	38.024	21.913	45.886	24.207	47.085	24.681	47.255	25.151	47.739	25.204	46.263	50.164

Figure SRXB-11-23 BSEP ATRIUM 10XM EOC Radial Exposure Distribution (AREVA MB2)

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Figure SRXB-11-24 BSEP ATRIUM 10XM BOC Radial Exposure Distribution Difference

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Figure SRXB-11-25 BSEP ATRIUM 10XM EOC Radial Exposure Distribution Difference

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**Figure SRXB-11-26 BSEP ATRIUM 10XM BOC
Axial Power (GEH/GNF P11)**

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**Figure SRXB-11-27 BSEP ATRIUM 10XM EOCLF
Axial Power (GEH/GNF P11)**

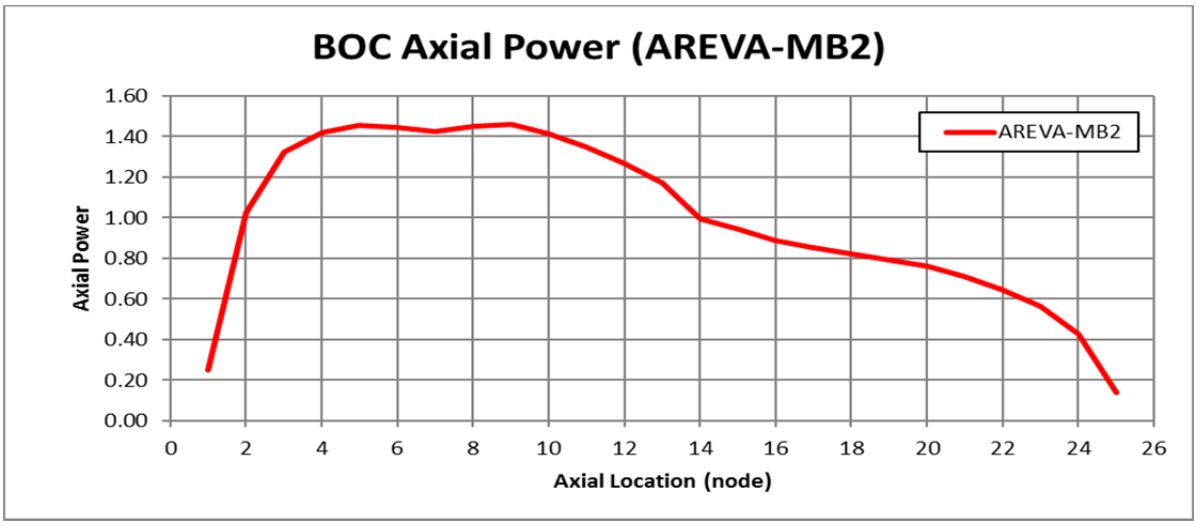

Figure SRXB-11-28 BSEP ATRIUM 10XM BOC Axial Power (AREVA MB2)

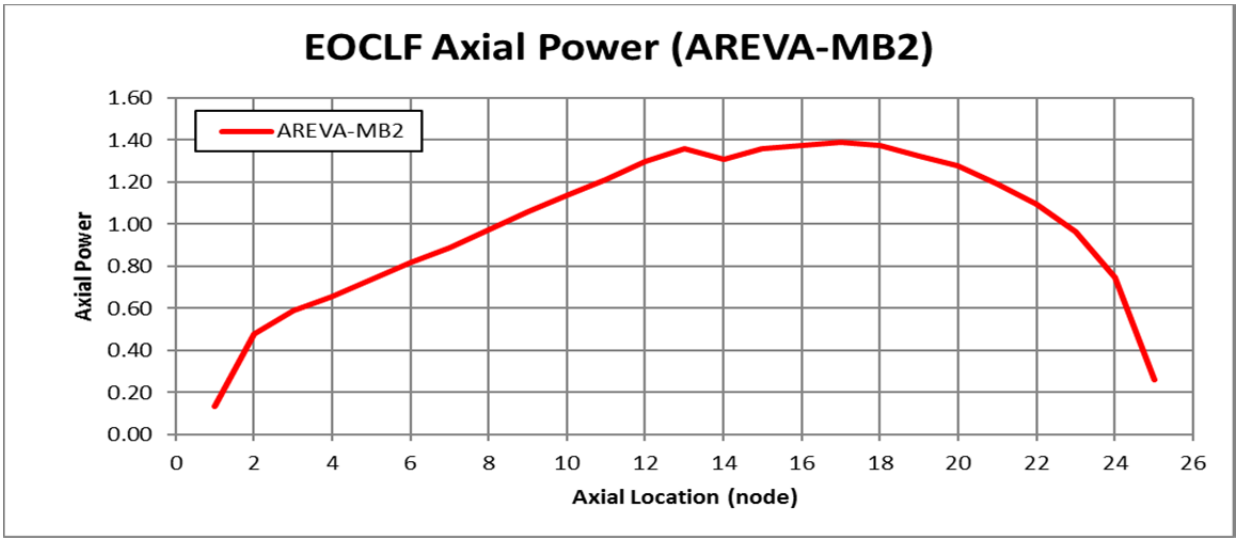

Figure SRXB-11-29 BSEP ATRIUM 10XM EOCLF Axial Power (AREVA MB2)

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**Figure SRXB-11-30 BSEP ATRIUM 10XM
BOC Axial Power Difference**

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**Figure SRXB-11-31 BSEP ATRIUM 10XM EOCLF
Axial Power Difference**

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Figure SRXB-11-32 BSEP ATRIUM 10XM BOC Axial Exposure (GEH/GNF P11)

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Figure SRXB-11-33 BSEP ATRIUM 10XM EOCLF Axial Exposure (GEH/GNF P11)

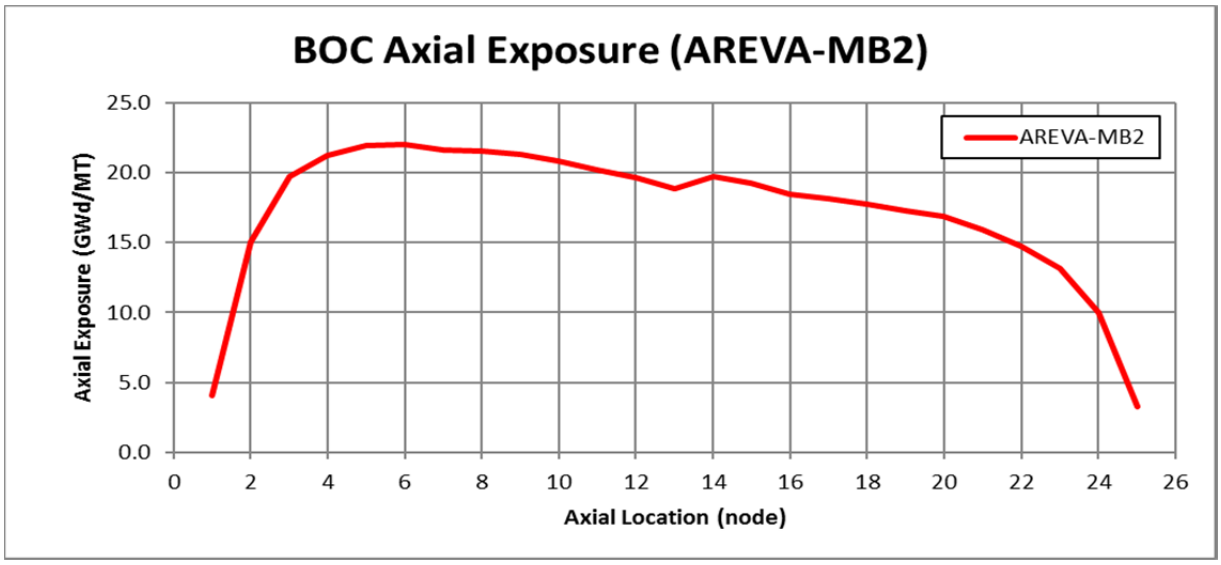

Figure SRXB-11-34 BSEP ATRIUM 10XM BOC Axial Exposure (AREVA MB2)

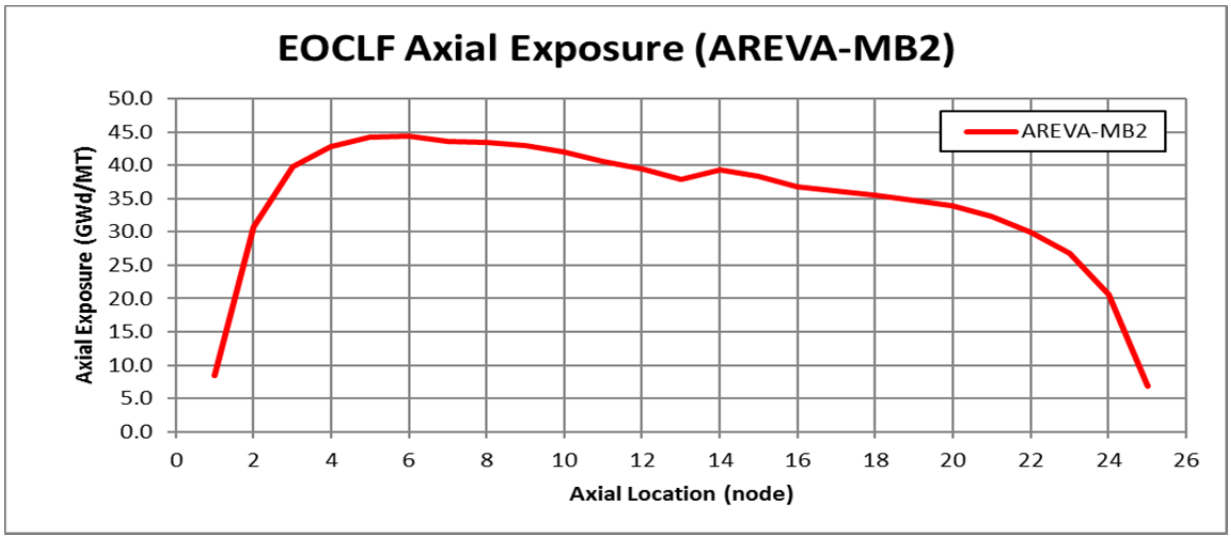

Figure SRXB-11-35 BSEP ATRIUM 10XM EOCLF Axial Exposure (AREVA MB2)

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Figure SRXB-11-36 BSEP ATRIUM 10XM BOC
Axial Exposure Difference

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Figure SRXB-11-37 BSEP ATRIUM 10XM EOCLF
Axial Exposure Difference

NRC SRXB-RAI-12

Please provide justification that safety limit minimum critical power ratio (SLMCPR) Penalty of 0.03 is not applicable to BSEP. Please provide data representative of the requested MELLLA+ Operating Domain for BSEP to justify the use of MICROBURN-B2 in the domain. If the data doesn't cover the entire operating domain, please identify the range that is not covered and identify any additional conservatisms that can be used to justify the adequacy of the method in this range.

Response SRXB-RAI-12

The 0.03 SLMCPR penalty described above is specified in Limitation and Condition 9.5 (SLMCPR 2) to NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domain." The following discussion expands upon supplemental information provided for NRC Request 1 in Duke letter BSEP 16-0101 dated November 9, 2016 (Reference SRXB-12-1) as well as information provided in a meeting held with the NRC on July 27, 2017.

Operation at the maximum core power and minimum core flow conditions allowed for Extended Power Flow Operating Domain (EPFOD) / Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain operations (100% rated power and 85% rated core flow), corresponds to a power to flow ratio of 44.66 MWt / Mlbm/hr which exceeds prior operating experience for Brunswick.

A Brunswick specific assessment of the power distribution uncertainties was presented in Section C.2 of ANP-3108P Revision 1, "Applicability of AREVA BWR Methods to Brunswick Extended Power Flow Operating Domain" (Reference SRXB-12-2). This previous assessment included plots of the 2D TIP uncertainty (δT'ij) for both C-lattice and D-lattice plants included in the MICROBURN-B2 licensing topical report, EMF-2158(P)(A) (i.e., Reference SRXB-12-3). Figure SRXB-12-1 through Figure SRXB-12-4 expand this comparison by providing the Brunswick specific δT'ij database. The following information is provided:

Figure SRXB-12-1	Brunswick 2D TIP Statistics (δT'ij) versus Cycle Number
Figure SRXB-12-2	Brunswick 2D TIP Statistics (δT'ij) versus Core Power/ Flow Ratio
Figure SRXB-12-3	Brunswick 2D TIP Statistics (δT'ij) versus Core Void
Figure SRXB-12-4	Brunswick 2D TIP Statistics (δT'ij) versus Core Power

These figures represent the same data with different independent variables. The database includes 214 full core gamma TIP measurements: 112 for BSEP Unit 1 (Cycles 14 through 20) and 102 for BSEP Unit 2 (i.e., Cycles 16 through 22).

Figure SRXB-12-1 through Figure SRXB-12-4 clearly demonstrate that the D-lattice radial TIP uncertainty reported on page 9-8 for "TIP Distribution Calculation" in the Reference SRXB-12-3 topical report is very conservative for Brunswick, **[**

]. Figure SRXB-12-2 through

Figure SRXB-12-4 also clearly demonstrate there is no correlation in the Brunswick specific uncertainty component due to the core power to flow ratio, core power, or core average void fraction. This is consistent with the conclusion of ANP-3108P Revision 1 based upon the data from EMF-2158(P)(A) topical report.

Figure C-18 and Figure C-24 in ANP-3108P Revision 1 (Reference SRXB-12-2) provide the 2D and 3D TIP statistics versus power to flow ratio for the C-Lattice plants in the EMF-2158(P)(A) topical report. These are repeated in this response as Figure SRXB-12-5 and Figure SRXB-12-6. As can be seen in these plots, fleet TIP data is available up to approximately 52 MWt / Mlbm/hr; however, the quantity of TIP data is limited above a ratio of 42 MWt / Mlb/hr. For this reason LPRM measurement data has also been evaluated. The primary purpose of the TIP system is to provide data for the periodic calibration of the individual LPRM detectors. Comparison of LPRM measured data with corresponding calculated data is therefore indicative of measured power distribution uncertainties.

The uncertainty in calculated-to-measured LPRM data as a function of power-to-flow ratio for the two Brunswick units is presented in Figure SRXB-12-7. LPRM data comparison for two BWR/6 plants that regularly operate at reduced flow is presented in Figure SRXB-12-8.

Figure SRXB-12-6 and Figure SRXB-12-8 demonstrate that there is no increase in the axial uncertainty for power with Framatome methods for power to flow ratios up to approximately 52 MWt / Mlbm/hr. Furthermore, there is no trend leading to higher uncertainties for higher Power/Flow ratios.

LPRM measurements are available at all operating statepoints. This comparison is more directly applicable to measured power distribution uncertainties since the LPRM instrumentation is used to determine the "measured" power distribution.

In order to put this data into perspective, Figure SRXB-12-9 presents the Brunswick Power/Flow map with a constant Power/Flow ratio equal to 52 MWt / Mlbm/hr line. There is only a very small corner of the EPFOD / MELLLA+ domain that includes operating states above Power/Flow ratios of 52 MWt / Mlbm/hr. Since there is no trend in the data a significant increase in the power distribution uncertainties is considered to be highly unlikely. Operation in this region is very infrequent and for very short periods of time.

Figure SRXB-12-1 presents the Gamma TIP uncertainties for operating statepoints for the Brunswick units. These results show significant margin to the uncertainty value used in the SLMCPR analysis (i.e., shown with a red line). This significant conservatism in the SLMCPR analysis is due to not taking any credit for the use of gamma TIP's at Brunswick as allowed by Framatome methodology.

The primary concern about EPFOD / MELLLA+ domain has been increased void fractions due to operation at higher power/flow ratios. While the use of MELLLA+ at Brunswick can result in ratios as high as approximately 53.6 MWt / Mlbm/hr, this only slightly exceeds the existing Framatome fleet database of approximately 52 MWt / Mlbm/hr. Furthermore, no adverse trend in power distribution accuracy has been noted for these higher power/flow ratios.

The following items are considered to be additional mitigations to the potential for increased power uncertainties:

- A sensitivity analysis provided in Section C.2 of ANP-3108P Revision 1 was performed with SAFLIM3D that showed low impact on the SLMCPR due to uncertainty changes.
- The region of the EPFOD on the Brunswick power/flow map that exceeds the current Framatome database is extremely small as shown in Figure SRXB-12-9. Consequently, operation in this region is considered to be unlikely and would only extend for very short periods of time if it were to occur.
- As illustrated in Figure SRXB-12-1, there is significant conservatism in the power distribution uncertainties used in the Brunswick SLMCPR analysis compared to the actual plant data since no credit is taken for the Gamma TIP hardware.

Based upon the discussion above it is concluded that no SLMCPR penalties are warranted for operation within the entire EPFOD / MELLLA+ domain at Brunswick.

Figure SRXB-12-1 Brunswick 2-D δT'ij Gamma TIP Response vs. Cycle Number

Figure SRXB-12-2 Brunswick 2-D δT'ij Gamma TIP Response vs. Power/Flow Ratio

Figure SRXB-12-3 Brunswick 2-D δT'ij Gamma TIP Response vs. Core Average Void Fraction

Figure SRXB-12-4 Brunswick 2-D δT'ij Gamma TIP Response vs. Core Power

Figure SRXB-12-5 EMF-2158(P)(A) 2-D TIP Statistics
for C-Lattice Plants vs. Core Power/Flow

Figure SRXB-12-6 EMF-2158(P)(A) 3-D TIP Statistics
for C-Lattice Plants vs. Core Power/Flow

Figure SRXB-12-7 Brunswick Statistics for Comparison of Measured and Calculated LPRM Values

Figure SRXB-12-8 BWR/6 3D Statistics for Comparison of Measured and Calculated LPRM Values

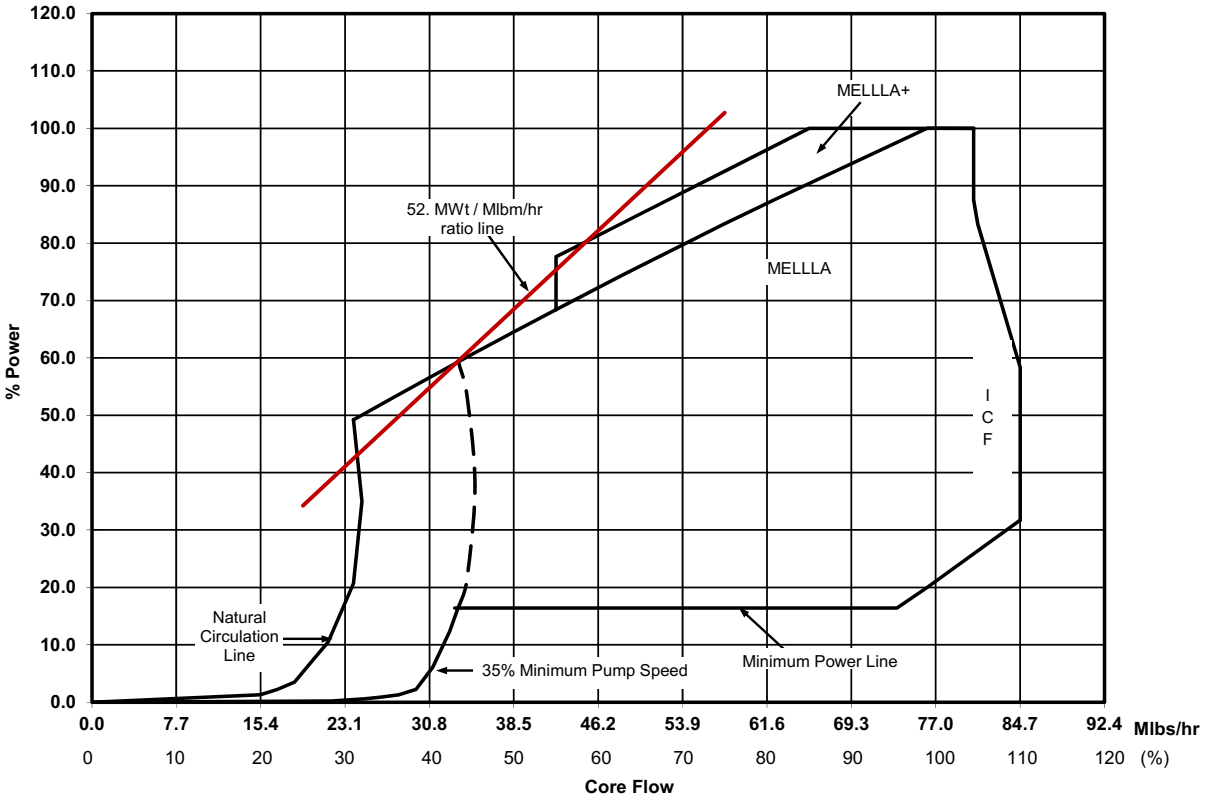

Figure SRXB-12-9 Brunswick Power Flow Operating Map with Constant P/F Line

References

SRXB-12-1 Duke Letter BSEP 16-0101, W. R. Gideon to USNRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Response to Request for Supplemental Information for License Amendment Request Regarding Core Flow Operating Range Expansion," dated November 9, 2016, Accession Number ML16330A504.

SRXB-12-2 Enclosure 12; ANP-3108(P) Rev. 1 to Duke Letter BSEP 16-0056, W. R. Gideon to USNRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Request for License Amendment Regarding Core Flow Operating Range Expansion," dated September 6, 2016, Accession Numbers ML16257A410 for the LAR and ML16257A412 for Enclosure 13 which is the non-proprietary version of the referenced report.

SRXB-12-3 EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.

NRC SRXB-RAI-13

AREVA ANP-3280P - *Brunswick Unit 1 Cycle 19 MELLLA+ Reload Safety Analysis, May 2016*, includes reference reload analyses for a mixed ATRIUM 10XM/ATRIUM-10 core. However, Section 1.0 of ANP-3280P indicates that ATRIUM-10 limits and analyses are not included in these reload analyses. Please explain this approach in more detail to ensure that it is appropriate for a full core of ATRIUM 10XM which will be the fuel type when MELLLA+ is implemented. For example, were the geometric and performance features of the ATRIUM-10 fuel assemblies modeled explicitly, while the SLMCPR calculations (such as those shown in Table 4.2 of ANP-3280P) considered only the ATRIUM 10XM fuel assemblies for Cycle 18?

How does Section A.1 of AREVA ANP-3108P *Applicability of AREVA NP BWR Methods to Brunswick Extended Power Flow Operating Domain, July 2015*, regarding mixed cores relate to the analyses performed in ANP-3280P?

Response SRXB-RAI-13

Brunswick Unit 1 Cycle 19 was selected to be the representative reload for the MELLLA+ License Amendment Request (LAR). As described in Section 1.0 of ANP-3280P, the Cycle 19 core consisted of a mix of ATRIUM-10 and ATRIUM 10XM fuel. The analyses supporting ANP-3280P were performed for the cycle-specific loading and use input data appropriate for each fuel type in the core. As noted, at the time of MELLLA+ implementation, only the ATRIUM 10XM fuel design will be resident in the core. As such, fuel specific ATRIUM-10 results are removed from ANP-3280P. The general discussion provided in Section A.1 of ANP-3108P presents the application of Framatome methods for mixed core configurations.

Specifically for the SLMCPR calculation presented in ANP-3280P, the analysis explicitly models each fuel type present in the core. The results provided in Table 4.2 of ANP-3280P were obtained from SLMCPR analyses based on the cycle-specific core loading.

NRC SNPB-RAI-1

Section 6.0 of ANP-3108 *Mechanical Limits Methodology* describes how the fuel mechanical design criteria are satisfied. Provide the details for the following aspects of the mechanical design for the extended power/flow operation at the Brunswick units:

a. Provide a summary description how fuel rod design criteria was applied for Extended Power/Flow Operating Domain (EPFOD) operation.

b. How the fuel design limits such as LHGR and burnup are established for the EPFOD operation.

c. Provide details of how the uncertainties (operating power, code model parameters, and fuel manufacturing tolerances) are utilized in the mechanical design analysis. Also, list the values and source of uncertainties utilized in the analysis.

Response SNPB-RAI-1a

As described in ANP-3108P, Section 6.0, the demonstration of compliance to thermal-mechanical limits for operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) or EPFOD is no different from operation in the MELLLA operating domain. Responses to specific questions are:

a. A summary of the application of the analysis methodology to the ATRIUM 10XM design is provided in Section 3.2 of Reference SNPB-1-1.

b. The determination of the Fuel Design Limit (FDL) and fuel rod exposure limit are unchanged for operation in MELLLA+.

c. The utilization of uncertainties (operating power, code model parameters and fuel manufacturing tolerances) is fully described in Reference SNPB-1-2.

There are several components to the reactor power uncertainty and these are described in Section 5.2. The treatment of power variations due to channel bow are addressed in an RAI response to BAW-10247PA [Reference SNPB-1-2, BAW-10247Q3(P), Appendix B].

The uncertainties for the model parameters are shown in an RAI response to BAW-10247PA [Reference SNPB-1-2, BAW-10247Q4P, Revision 000, Tables C, D and E].

The manufacturing uncertainties are compiled on an annual basis and statistically abstracted on a triennial basis. The most recent triennial statistics are shown in Table SNPB-1-1. These statistics typically show very little drift from year to year.

Table SNPB-1-1 Manufacturing Uncertainties for 2014/2015/2016

described in Topical Report (TR), XN-NF-80-19(P)(A) Volume 1 Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", (1983). The licensee has reported that that the maximum fuel rod enthalpy is less than the NRC threshold of 280 cal/g.

Appendix B of Standard Review Plan (NUREG-0800, March 2007) provides interim acceptance criteria for the reactivity initiated accidents, such as, CRDA for BWRs. The technical and regulatory basis for the interim criteria is documented in a memorandum dated January 19, 2007 (ADAMS Number ML070220400). This memorandum, with respect to the criteria on fuel enthalpy for maintaining coolable geometry, states that the 280 cal/g criteria has been found inadequate to ensure fuel rod geometry and long term coolability. NUREG-0800 Section 4.2 defines reactivity-initiated accident (RIA) fuel clad failure criteria as (1) radial average fuel enthalpy greater than 170 cal/g for BWR at zero or low power, and (2) local heat flux exceeding fuel thermal design limits (CPR) at-power events in BWRs. The technical basis for the BWR fuel failure criteria is detailed in the January 19, 2007 memorandum.

The draft regulatory guide (DG-1327 of November 2016) defines fuel cladding failure thresholds, analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits. The empirically based pellet-cladding mechanical interaction (PCMI) failure thresholds are shown in Figures 2 through 5 for fully recrystallized annealed (RXA) and stress relief annealed (SRA) cladding types at both low and high temperature reactor coolant conditions. The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content in weight parts per million (wppm).

Based on the background information given, please provide an evaluation to show that the fuel failure criteria as proposed by either SRP Section 4.2 (2007) or DG-1327 (2016) can be met for the control rod drop accident analysis at BSEP Units 1 and 2.

Response SNPB-RAI-2

As discussed with and agreed to by the NRC licensing project Manager, Mr. Andy Hon, NRC Question SNPB-RAI-2 will require additional time to develop a response. Accordingly, the response to SNPB-RAI-2 will be provided under separate letter no later than March 1, 2018.

NRC SNPB-RAI-3

Appendix B of ANP-3108P describes void-quality correlations, **[ ]** and Ohkawa-Lahey, and AREVA's independent validation of these correlations for application to ATRIUM 10XM and Brunswick plant operation at EPFOD conditions. Please provide responses for the following items.

a. Detailed description of AREVA's independent validation of the **[ ]** and Ohkawa-Lahey correlations using the test data from FRIGG experiments that generated Table B-1 and Figures B-1 through B-4 of ANP-3108P.

b. Supporting calculation detail that shows how the uncertainties and biases are utilized in the analyses described in Sections B.2 and B.3 of ANP-3108P.

Response SNPB-RAI-3a

Framatome's independent validation of the BWR void fraction correlations against the FRIGG experiments was performed to supplement and confirm the validation performed by the original developers of the correlations. For each of the experimental tests, the nominal test conditions were used to calculate the void fraction at each of the measurement locations. These calculated void fractions were then compared to the nominal measured void fractions and plotted in Figures B-1 and B-3 in Appendix B of ANP-3108P. The FRIGG benchmarks provide a basis for assessing the continued applicability of the licensed void fraction correlations to both modern fuel designs and operation of those fuel designs to EPFOD conditions.

Concurrent with the Framatome fuel design development that lead to the introduction of partial length fuel rods (PLFRs) and swirl vane spacer designs, fundamental testing was conducted to assess the adequacy of current methodologies in predicting steady-state behavior (critical power, pressure drop and void fraction) as well as dynamic behavior (i.e., channel decay ratio and instabilities). Framatome performed void fraction measurements to specifically assess the impact of the ATRIUM-10 fuel design attributes. The void fraction tests were performed at the KATHY test facility using a prototypical BWR CHF test assembly. The test assembly used 8 PLFRs, mixing vane grids, a **[** **]** typical of CHF tests. The tests were conducted using a scanning gamma densitometer that traversed across the test section. Void measurements were made at one of three different elevations in the assembly for each test point: just before the end of the part length fuel rods, midway between the last two spacers, and just before the last spacer.

To assess the high void fraction performance associated with EPFOD, AREVA performed gamma tomography measurements on the ATRIUM 10XM (i.e., the fuel design used in Brunswick). Again, the tests were performed at the KATHY test facility using a prototypical BWR CHF test assembly. The test assembly used the PLFR and grid spacer configuration of the ATRIUM 10XM and a **[** **]** in the test facility.

Plots presenting the comparisons between the void-quality correlations and measured data were presented in Figures B-2 and B-4 in Appendix B of ANP-3108P for the steady-state core simulator and transient methods. Visual examinations of these plots as well as comparisons of the mean and standard deviations in Table SNPB-3-1 demonstrate that the modern AREVA fuel design attributes and the potential operation at high void fractions for EPFOD result in no significant change in the void-quality correlation uncertainties relative to the historic validation.

Response SNPB-RAI-3b

The analyses in Sections B.2 and B.3 of ANP-3108P present the OLMCPR impact of biasing the nominal void fraction correlations over the uncertainty range relative to the inherent

conservatism of the approved licensing methodologies on the OLMCPR. Three cases are presented relative to the nominal licensing application:

Case 1: Adjustment of the steady-state simulator void fraction correlation to minimize the mean error over the void fraction range. The resultant steady-state correlation comparison to measured data is presented in Figure B-5 of ANP-3108P. **[**

]

Case 2: Adjustment of Case 1 **[** **]** that would correspond to a **[** **]** offset for the ATRIUM-10 benchmark comparisons. For the steady-state simulator the nodal void fractions calculated from the modified correlation in Case 1 are adjusted by **[**

]

Case 3: Adjustment of Case 1 **[** **]** that would correspond to a **[** **]** offset for the ATRIUM-10 benchmark comparisons. For the steady-state simulator the nodal void fractions calculated from the modified correlation in Case 1 are adjusted by **[**

]

For each of the cases identified above, a multi-cycle depletion with MICROBURN-B2 was performed to establish the end-of-cycle core conditions (exposure, power flow and reactivity parameters) in equilibrium with the revised void quality correlation. The end-of-cycle condition was then used in the downstream transient analyses to quantify the impact on the OLMCPR.

Table SNPB-RAI-3-1 Summary of Void-Quality Correlation Statistics

	FRIGG	**ATRIUM-10 (KATHY)**	**ATRIUM 10XM (KATHY)**
Core Simulator Mean	**[ ]**	**[ ]**	**[ ]**
Core Simulator Standard Deviation	**[ ]**	**[ ]**	**[ ]**
Transient Method Mean	**[ ]**	**[ ]**	**[ ]**
Transient Method Standard Deviation	**[ ]**	**[ ]**	**[ ]**

NRC SNPB-RAI-4

It has been stated in Section 9.3.3 *ATWS with Core Instability* of DUKE-0B21-1104-000(P) (M+ LTR) that sensitivity studies are performed with GEXL by varying from **[[ ]]**, and conservative GEXL parameters have been developed by AREVA for Application to ATRIUM 10XM fuel. The staff could not find any reference to this GEXL correlation and could not determine which version of GEXL correlation has been utilized and which sensitivity analysis has been performed.

a. Please provide details of the formulation of the above GEXL correlation formulation.

b. Do the biases and uncertainties associated with this GEXL correlation comply with the requirements for the ATRIUM 10XM fuel design?

c. Is this GEXL correlation compatible with TRACG code?

d. Are the R-factors (K-factors) and additive constants specifically derived for the ATRIUM 10XM fuel design?

Response SNPB-RAI-4a

The GEXL97 correlation for ATRIUM-10 fuel was transmitted by GEH to Framatome. This transmittal included both the correlation form and the coefficients for ATRIUM-10 fuel. Framatome then modified the coefficients as necessary to conservatively model the ATRIUM 10XM fuel. The data used to benchmark this correlation is the same data set used to benchmark the ACE/ATRIUM 10XM correlation. No changes to the correlation form were permitted, so only changes to the correlation coefficients were allowed. The range of applicability for this correlation was chosen to be within the range of the data available.

Response SNPB-RAI-4b

The GEXL correlation for ATRIUM 10XM was not constructed as a general use correlation. Instead, it was biased to provide a conservative CPR prediction specifically for the ATWS-I event. As such, this correlation is not valid for all applications. The correlation was designed to provide conservative estimates for critical power during ATWS-I scenario. This correlation was created to be closer to best estimate, but still conservatively low, at the low flow conditions of interest for the ATWS-I scenario. For other assembly flows the correlation is more conservative. To illustrate this trend, the ECPR versus mass flux for the benchmarking data set is shown in Figure SNPB-4-1. This figure shows the trend of increasing conservatism with increasing flow, and also shows the overall conservatism of the correlation.

Figure SNPB-4-1 ECPR Versus Mass Flux for ATRIUM 10XM GEXL Correlation

In order to assess the sensitivity of the critical power calculation, a range of critical power adjustments was provided to conservatively bound the benchmarking uncertainty. Since the correlation itself is not a best estimate correlation where ECPR is approximately 1.0, the uncertainty ranges were adjusted to account for this. Ranges were calculated separately for both the high flow range and the low flow range. As an example, the mean of the low flow ECPR data was found to be **[ ]** with a standard deviation of **[ ]**. Using a **[ ]** sigma multiplier, the adjusted high end ECPR was found to be ECPR = **[ ]**. In order to bring this ECPR back to the best estimate value of 1.0, a **[ ]** decrease in the critical power would be required. For purposes of sensitivity study, a **[ ]** conservative adjustment in critical power was recommended. For the reduced conservatism case, the adjusted ECPR was found to be ECPR = **[ ]**. For the purposes of sensitivity study, a **[ ]** increase (less conservative direction) in critical power was recommended.

The correlation coefficients, a general description of the conservatism applied, the range of applicability, and the range of CPR sensitivity to be examined were transferred from Framatome to GEH.

Response SNPB-RAI-4c

Yes, this GEXL correlation is compatible with the TRACG code.

Response SNPB-RAI-4d

Framatome utilized the ACE K-factor in place of the GEXL R-Factor in the CPR calculation. The subsequent benchmarking to the data described above showed this to be a reasonable assumption. The assembly specific R-factors were calculated by Framatome for all assemblies at each core exposure analyzed for ATWS-I using the K-factor method. In order to ensure that the K-factors indeed produced conservative results with the GEXL correlation, the GEXL correlation was implemented in MICROBURN-B2. Critical power calculations were performed for the limiting assemblies at two exposures (BOC and EOC) to demonstrate that the K-factors and the provided GEXL correlation produced a conservative prediction of critical power when compared to the approved ACE/ATRIUM 10XM correlation. These calculations were performed at both high flow and low flow statepoints and confirmed the overall conservatism of the correlation, as well as the trend in increasing conservatism with increasing flow.

NRC SNPB-RAI-5

Appendix A page A-2 indicates that "*At the time of the creation of this document, Reference 7 (ANP_10298PA Revision 0 Supplement 1P Revision 0 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, December 2011) had not been generically approved."* The staff notices that Revision 1 of ANP-10298P-A was published as the accepted version (-A) incorporating Reference 7 in to Reference 2 (ANP-10298PA Revision 0) in March 2014. However, the Brunswick EPFOD analysis was performed using the Revision 0 when Revision 1 of the topical report was available. Please explain why the Revision 1 was not used for the Brunswick analysis. What is the impact on results if the Revision 1 is used in the analysis?

Response SNPB-RAI-5

As stated in Section 2.2 of the M+SAR (ADAMS Accession No. ML16257A411), the analyses for the representative cycle (BSEP Unit 1 Cycle 19) were performed using the critical power correlation consistent with the Technical Specification (TS) at the time of BSEP Unit 1 Cycle 19. Since that time, the BSEP Unit 1 TS has been updated (i.e., Amendment 269) to reference Revision 1 of ANP-10298PA (ADAMS Accession No. ML16019A029). In Section 4.1 of the supporting safety evaluation for Amendment 269 of the BSEP Unit 1 TS, a comparison of ΔCPRs between Revision 0 and Revision 1 of ANP-10298PA for BSEP limiting events is provided.

NRC RAI SNPB-RAI-7

For ATWS (Licensing basis) calculations (Section 9.3.1 of DUKE-0B21-1104-000(P) (M+ LTR)) and for ATWS with core instability (Section 9.3.3 of M+LTR), to address the effect of ATRIUM 10XM fuel, sensitivity analyses were performed. Sensitivity ranges are selected to include expected variation of the parameters; direct energy deposition, gap conductance as

applied to ATRIUM 10XM as supplied by PRIME, thermal and hydraulic channel losses. Please respond to the following questions.

a. Discuss the suitability of using ODYN code for ATRIUM 10XM fuel design.

b. It is stated that gap conductance data as applied to ATRIUM 10XM fuel data is supplied by the PRIME code. However, Section 4.0 of safety evaluation for PRIME *limits the use of PRIME to "approved GNF fuel rods designs clad in RXA Zircaloy-2.*" This limitation further states that *"In case core transition from one vendor to another, PRIME may be applied to generate inputs for the downstream safety analyses and overpower limit compliance for the non-GNF BWR fuel, provided that the design and operating parameters for the non-GNF fuel must be within the range approved for the PRIME models;*

    1. Since the cladding type for the ATRIUM 10XM is different from the cladding type of GNF fuel designs (RXA), please justify how PRIME can be used for the ATRIUM 10XM analysis.

    2. To satisfy the limitation for PRIME stated above, show that the ATRIUM 10XM design and operating parameters are within the ranges of the parameters approved for PRIME

In order to use PRIME for the ATRIUM 10XM analyses, has any modifications been performed in the PRIME code to enable it to perform ATRIUM 10XM analyses?

Response SNPB-RAI-7a

The GEH transient analysis model, ODYN, had been used with and is qualified to apply to different fuels. The same method for GNF fuel application is applied for ATRIUM fuel. ATRIUM-10 fuel has been used with ODYN for LaSalle, Columbia, River Bend, and Grand Gulf. ATRIUM-10 ATWS calculations have also been performed using ODYN to support Susquehanna. ATRIUM-10 and ATRIUM 10XM are very similar except for the inventory of full and part-length rods. For the BSEP ATWS analyses, the ATRIUM 10XM fuel geometry was modeled explicitly. The core parameters were introduced into ODYN in the same manner via PANAC wrap-ups as would be done for GNF and ATRIUM-10 fuel designs. ODYN is suitable for ATRIUM 10, and as a result remains suitable for ATRIUM 10XM.

Response SNPB-RAI-7b

The PRIME computer model provides best-estimate predictions of the thermal mechanical performance of nuclear fuel rods. For BSEP ATWS analyses, the PRIME code was not used for any thermal mechanical evaluation. The PRIME code was only used to create the fuel files needed to determine the gap conductance.

3. Any differences in heat transfer from differences in the cladding thermal properties are insignificant compared to the gap conductance heat transfer. Gap conductivity sensitivities cover a large range of uncertainty and cover any variation in cladding heat

transfer properties. The impact of the large variability in gap conductance had a small impact on PCT.

4. The PRIME code was only used to create the fuel files needed to determine the gap conductance; therefore, many of the operating parameters are not applicable. The fuel parameter sensitivity study accounts for uncertainty in the fuel design.

No modifications were made to the PRIME code.

NRC RAI EMIB-RAI-1

As per the LTR, NEDC-33006P-A, Rev 3, the relief and safety valves and the reactor protection system provide overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, -34, and -35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 Table of RS-001, "Review Standard for Extended Power Uprates," The pressure relief systems provide reactor overpressure protection for the NSSS to prevent failure of the nuclear system pressure boundary and uncontrolled release of fission products, during abnormal operational transients, the ASME Upset overpressure protection event, and postulated ATWS events. Section 9.3.1 of the M+SAR evaluates the ATWS response for operation at the MELLLA+ operating domain.

Since the evaluation of the steam separator and dryer performance at MELLLA+ conditions indicates an increase in moisture carry over (MCO) of < 0.20 wt% where the original MCO performance specification was 0.10 wt. Please discuss the impact of the higher moisture concentration on components in the main steam lines, including MSIVs and flow restrictors.

Response EMIB-RAI-1

Any increase in the steam moisture content has the potential to increase erosion/corrosion rates in the main steam line (MSL) piping and components. It can be reasonably assumed that the rate of flow accelerated corrosion (FAC) in the MSLs, moisture separator, and steam drains is proportional to the local steam moisture content. The increase in the steam dryer outlet moisture from 0.1 wt.% to 0.2 wt.% is insignificant with respect to pipe erosion/corrosion. The 0.1 wt.% increase in MCO will cause a commensurate increase in steam moisture content at any location in the MSL. Nonetheless, the small increase in steam moisture content (or corresponding reduction in steam quality) will have only a negligible effect on MSL pipe erosion/corrosion. The plant’s FAC program, which includes the MSL, monitors susceptible areas for corrosion and factors the results into the piping replacement program at the plant site so that any adverse impact on the MSL piping will be monitored by the plant. As discussed in Section 10.7.2 of Enclosure 5 of the maximum extended load line limit analysis plus (MELLLA+) submittal

(Reference EMIB-1-1), "The BSEP FAC implementing documents…consider the FAC susceptibility of lines with steam quality above 99.5% to be low." In addition, Section 3.3.4 of Enclosure 5 of the MELLLA+ submittal (i.e., Reference EMIB-1-1) concludes that "The amount of time BSEP is operated with higher than the original design moisture content (0.10 wt.%) is expected to be minimal" since the increased MCO is only expected to occur when the plant is operating at or near the MELLLA+ minimum core flow. Therefore, the slight increase in steam moisture (i.e., an increase of 0.1 wt.% at any location) is not expected to have a significant impact on the MSL.

The principal MSL components that are potentially susceptible to higher steam moisture content are the main steam isolation valves (MSIVs), the safety relief valves (SRVs), the flow elements (or flow restrictors), and the high-pressure turbine. For BWR/4 plants equipped with GE-manufactured turbines such as BSEP, the MSIV is typically the limiting MSL component with respect to increased MCO.

MSIVs

The local steam moisture content at the MSIVs is expected to increase slightly from approximately 0.25 wt.% to 0.26 wt.%. Because all critical surfaces of the MSIVs are hard-faced, erosion of these surfaces due to the slightly higher moisture levels is not a concern. Further, since the MSIVs are leak tested at every outage, potential degradation will be monitored and corrected if observed, as required by plant operating procedures. Therefore, no adverse impacts to the MSIVs are expected under MELLLA+ operating conditions with the steam dryer outlet moisture of 0.2 wt.%.

SRVs

The SRVs are required to operate under high-quality two-phase flow conditions and have been demonstrated to have the capability to function with water flows at low-pressure conditions. The purchase specification for the SRVs stipulate a steam moisture content of < 1.0 wt.% under normal operating conditions. Since the increase in MCO from 0.1 wt.% to 0.2 wt.% does not cause the moisture content at the SRVs to exceed this value, operation at the higher moisture content is acceptable.

Flow Elements (Flow Restrictors)

Increasing MCO from 0.1 wt.% to 0.2 wt.% produces a negligible (i.e., < 0.1%) change in the specific volume of the steam leaving the reactor pressure vessel. The resulting difference in specific volume at the steam flow restrictors does not affect steam flow measurement. Similarly, the increase in MCO produces a negligible (i.e., < 0.1%) increase in the steam density, which causes a slight reduction in the margin to choke flow (i.e., ~0.4%) through the flow restrictors. In addition, the increase in MCO to 0.2 wt.% reduces the local steam quality at the flow elements from approximately 99.75 wt.% to 99.65 wt.%, but remains above the 99.5 wt.% steam quality criterion for low FAC susceptibility. Therefore, the slight increase in MCO has no impact on the flow restrictors.

High-Pressure Turbine

The higher MCO increases the steam moisture content entering the high-pressure turbine. At 0.20 wt.% MCO, the moisture content of steam entering the high-pressure turbine is expected to increase from 0.51 wt.% to 0.61 wt.%. Nonetheless, this increase remains well below the maximum inlet steam moisture content for the turbine **[[ ]]**. Therefore, the slight increase in steam moisture has no impact on the high-pressure turbine.

In summary, there are no adverse impacts to the MSL piping and components associated with continuous reactor operation with MCO less than or equal to 0.2 wt.%.

References

EMIB-1-1 Duke Energy, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," DUKE-0B21-1104-000, July 2016. (Enclosure 5 of Letter, William R. Gideon (Duke Energy) to NRC Document Control Desk, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Request for License Amendment Regarding Core Flow Operating Range Expansion," BSEP 16-0056, September 6, 2016.)