William R. Gideon
Vice President
Brunswick Nuclear Plant
P.O. Box 10429
Southport, NC 28461

o: 910.832.3698

November 1, 2017

Serial: BSEP 17-0093

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)

Reference:

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 Containment Accident Pressure in the MELLLA+ LAR (CACs MF8864 and MF8865) (Non-Proprietary)*¸ dated October 2, 2017, ADAMS Accession Number ML17275A277

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 October 2, 2017, 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 enclosed. The questions and/or responses to RAIs SRXB-C-RAI 2, RAI 3, and RAI 4 involve proprietary information. Enclosure 1 provides responses to RAIs SRXB-C-RAI 2, RAI 3, and RAI 4 which contain proprietary information, as defined by 10 CFR 2.390.

General Electric Hitachi (GEH), as owner of the proprietary information, has executed the affidavit provided in Enclosure 2, which identifies 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 a GEH transmittal that is referenced by the affidavit. The proprietary information has been

faithfully reproduced in Enclosure 1 such that the affidavit remains applicable. The GEH proprietary information 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 the affidavit provided in Enclosure 2 which provides the basis for the proprietary determination.

A non-proprietary version of the responses to RAIs SRXB-C-RAI 2, RAI 3, and RAI 4, as well as responses to RAIs SRXB-C-RAI 5, RAI 6, and RAI 7, is provided in Enclosure 3.

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 November 1, 2017.

Sincerely,

William R. Gideon

WRM/wrm

Enclosures:

1. Response to Request for Additional Information SRXB-C-RAI 2, RAI 3, and RAI 4 **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**
2. GEH Affidavit Regarding Withholding Contained in Attachment 1 of GEH Letter GEH-PGN-MPLUS-148, "GEH Responses to NRC MELLLA+ Requests for Additional Information SRXB-C-RAIs 2, 3, and 4," dated October 6, 2017
3. Response to Request for Additional Information

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. Andrew Hon (Mail Stop OWFN 8G9A) **(Electronic Copy Only)**
11555 Rockville Pike
Rockville, MD 20852-2738
Andrew.Hon@nrc.gov

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

cc (with enclosures 2 and 3):

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
lee.cox@dhhs.nc.gov

GEH Affidavit Regarding Withholding
Contained in Attachment 1 of GEH Letter GEH-PGN-MPLUS-148,
"GEH Responses to NRC MELLLA+ Requests for
Additional Information SRXB-C-RAIs 2, 3, and 4,"
dated October 6, 2017

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-148, "GEH Responses to NRC MELLLA+ Requests for Additional Information SRXB-C-RAIs 2, 3, and 4," dated October 6, 2017. The GEH proprietary information in Attachment 1, which is entitled "Response to SRXB-C-RAIs 2, 3, and 4 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 6th day of October 2017.

Lisa K. Schichlein

Lisa K. Schichlein
Senior Project Manager, NPP/Services Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
Wilmington, NC 28401
Lisa.Schichlein@ge.com

Response to Request for Additional Information

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, add a new TS 5.6.7, and add 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 October 2, 2017, by electronic mail, the NRC provided a request for additional information (RAI) regarding the LAR. The response to the RAI is provided below.

NRC Background

By application dated September 6, 2016 (**A**gencywide **D**ocuments **A**ccess and **M**anagement **S**ystem (ADAMS) Accession No. ML16257A410) (Reference 1), pursuant to Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.90), Duke Energy Progress, Inc. (Duke Energy, or the licensee) submitted a License Amendment Request (LAR) proposing revisions to the Brunswick Steam Electric Plant (BSEP) Units 1 and 2 operating license. The proposed request would allow a change in the BSEP Units 1 and 2 Technical Specifications from operating in the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to operating in the expanded MELLLA Plus (MELLLA+) operating domain at the currently licensed thermal power.

The Nuclear Regulatory Commission (NRC) staff has reviewed the containment related portions in Section 4.0, “Engineered Safety Features” in Enclosure 5 or M+SAR (MELLLA+ Safety Evaluation Report) (Reference 2), and Enclosure 11 (Reference 3) of the licensee’s letter dated September 6, 2016 (Reference 1). In order to complete its review, the staff requests responses to the following Requests for Additional Information (RAIs). Note that the response to SRXB-C-RAI 1 was received by the NRC in licensee’s letter dated April 6, 2017 (ADAMS Accession No. ML17096A482). The proprietary information, pursuant to 10 CFR Section 2.390, in the RAIs is identified by underlined text (in red font) enclosed within double brackets as shown here [[example proprietary text]].

NRC RAI SRXB-C-RAI 2

Regulatory Basis: Title 10 of the U.S. *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 16 (i.e., GDC 16) states: "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment

and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

To assure that the containment design conditions, i.e., its design pressure and temperature are not exceeded during a Loss-of-Coolant Accident (LOCA) in the MELLLA+ operating domain, it is necessary to determine their peak values of these conditions for a bounding case.

Refer to Section 4.1.1 of the M+SAR (Reference 2); provide the list of the analyzed cases for the recirculation and main steam line break Loss-of-Coolant Accidents (LOCAs) that formed the basis for the limiting primary containment response due to a postulated LOCA as initiated from 102% power / 85% core flow) (Figure 1-1 in M+SAR (Reference 2), MELLLA+ state point N). Include the calculated primary containment pressure and temperature results corresponding to each case analyzed in the list. Provide justification that the list is complete and no further cases are necessary to be analyzed.

If the MELLLA+ state point N in Figure 1-1 of M+SAR (Reference 2) does not generate the limiting primary containment temperature and pressure responses, please include the analysis cases and results from the other state points that determined the limiting case.

NRC RAI References

1. Letter from Duke Energy to NRC dated September 6, 2016, "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," (ADAMS Accession Number ML16257A410)

2. Enclosure 5 of Reference 1, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," Proprietary (ADAMS Accession Number ML16257A413)

Response

For Boiling Water Reactor (BWR) plants with a Mark I containment, the limiting break (or Design Basis Accident (DBA)) for short-term containment pressure and temperature is the double-ended guillotine break of a recirculation suction line. Therefore, [[

]]

For Brunswick Steam Electric Plant (BSEP) MELLLA+, sensitivity studies were performed to determine the limiting power/flow point. [[
]] Table 2-1 shows the sensitivity results. [[

]]

Table 2-1 Sensitivity Study Results of Containment Analysis[1]

MELLLA+ State Point in Figure 1-1 of the BSEP M+SAR	Power (%)	Flow (%)	Recirculation Suction Line Break (RSLB) Short-Term Containment Pressure (psia)	RSLB Short-Term Containment Temperature (°F)
N	[[			
E[4]				]]

Notes:

1: [[

]]

4: This point, together with Point D, belongs to the CLTP (EPU) domain.

[[

]]

In summary, sufficient runs and/or evaluations were made for the BSEP MELLLA+ containment short term analysis. No further cases are necessary to be analyzed for the BSEP MELLLA+ domain.

As discussed above, [[

]]

GEH Response References

2-1. 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.)
2-2. GE Nuclear Energy, "Safety Analysis Report For Brunswick Units 1 and 2 Extended Power Uprate," NEDC-33039P, August 2001.

NRC RAI SRXB-C-RAI 3

Regulatory Basis: 10 CFR, Part 50, Appendix A, GDC 4 requires in part, Structures, Systems, and Components (SSCs) important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects of discharging fluids that may result from equipment failures.

In order to meet the above requirement of GDC 4, it is necessary to assure that the Condensation Oscillation (CO) load, which is one of the dynamic load imposed on the containment and its internal SSCs, during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.1 of M+SAR (Reference 2) under heading "Condensation Oscillation Loads" states:

> The Mark I CO [Condensation Oscillation] load definition was developed from test data from Full Scale Test Facility (FSTF) tests (Reference 33 [GE Nuclear Energy, "Mark I Containment Program, Full Scale Test Program Final Report, Task Number 5.11," NEDE-24539-P, April 1979]) to simulate LOCA thermal-hydraulic conditions (i.e., [[ ]]. The tests are bounding for all US Mark I plants, including the BSEP, considering MELLLA+ conditions.

Explain why the FSTF tests results are bounding for the BSEP Units 1 and 2 LOCA CO loads in the MELLLA+ operating domain.

NRC RAI References

1. Letter from Duke Energy to NRC dated September 6, 2016, "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," (ADAMS Accession Number ML16257A410)

2. Enclosure 5 of Reference 1, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," Proprietary (ADAMS Accession Number ML16257A413)

Response

Note that the Condensation Oscillation (CO) evaluation is located in Section 4.1.3.1 of the BSEP M+SAR.

CO loads result from oscillation of the steam-water interface that forms at the vent exit during the region of vent high steam mass flow rate. The CO loads occur after pool swell. The basis for the Mark I CO load definition is the Load Definition Report (LDR, Reference 3-1). [[

]]

Table 3-1 Results Summary For BSEP CO Load

[[			
			]]

Note:

1: [[ ]]

GEH Response References

3-1. GE Nuclear Energy, "Mark I Containment Program Load Definition Report," NEDO-21888, Revision 2, November 1981.

3-2. GE Company, "Mark I Containment Program, Full Scale Test Program Final Report, Task Number 5.11," NEDE-24539-P, April 1979.

NRC RAI SRXB-C-RAI 4

Regulatory Basis: 10 CFR, Part 50, Appendix A, GDC 4 requires in part, SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects of discharging fluids that may result from equipment failures.

In order to meet the above requirement of GDC 4, it is necessary to assure that the chugging load, which is one of the dynamic load imposed on the containment and its internal SSCs during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.1 of M+SAR (Reference 2) under heading "Chugging Loads" states:

> The thermal-hydraulic conditions for these tests [[
>]] were selected to produce maximum chugging amplitudes so that it bound all Mark I plants. Therefore, the current chugging load definitions remain applicable at MELLLA+ conditions for BSEP.

Explain why the FSTF tests results are bounding for the BSEP Units 1 and 2 LOCA chugging loads in the MELLLA+ operating domain.

NRC RAI References

1. Letter from Duke Energy to NRC dated September 6, 2016, "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," (ADAMS Accession Number ML16257A410)

2. Enclosure 5 of Reference 1, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," Proprietary (ADAMS Accession Number ML16257A413)

Response

Note that the chugging loads evaluation is located in Section 4.1.3.1 of the BSEP M+SAR.

Chugging loads include loads on the suppression pool boundary and submerged structures and vent (downcomer) lateral loads. Chugging loads result from the collapse of steam bubbles that form at the vent exit (Reference 4-1). [[

]] The design loads for Brunswick are in accordance with the LDR (Reference 4-2) load definition, which in turn is in accordance with Reference 4-1. [[

]]

GEH Response References

4-1. GE Company, "Mark I Containment Program, Full Scale Test Program Final Report, Task Number 5.11," NEDE-24539-P, April 1979.
4-2. GE Nuclear Energy, "Mark I Containment Program Load Definition Report," NEDO-21888, Revision 2, November 1981.

NRC RAI SRXB-C-RAI 5

Regulatory Basis: 10 CFR, Part 50, Appendix A, GDC 38 states in part, the containment heat removal system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

To assure that the containment heat removal function is adequately performed during a design basis LOCA, the pumps that draw water from suppression pool during the LOCA should have a positive margin for the Net Positive Suction Head (NPSH). In order to determine the minimum margin, the limiting (maximum) LOCA suppression pool temperature response should be analyzed with biased inputs for calculating the available NPSH at the pump inlet using the SHEX and GOTHIC computer codes for the conservative and realistic analyses.

Refer to Enclosure 11 (Reference 3), response to SECY-11-0014 (Reference 4) Criteria 1.

(a) Confirm that the same inputs and assumptions, including containment heat sinks and their associated heat transfer coefficients, were used for the conservative suppression pool temperature response (that maximizes the temperature) analysis using GOTHIC 8.0 and SHEX. Provide justification for the differences in those cases where any of the inputs, assumptions, heat sinks and the associated heat transfer coefficients were different in the two analyses.

(b) Please identify which of the input parameters and assumptions were assumed to be different in the GOTHIC 8.0 realistic analysis from the GOTHIC 8.0 and SHEX conservative analyses.

(c) Provide the basis of the input values selected for the GOTHIC 8.0 realistic analysis.

NRC RAI References

3. Enclosure 11 of Reference 1, "SECY-11-0014 Discussion - Use of Containment Accident Pressure (CAP) in Analyzing ECCS and Containment Heat Removal System Pump Performance," (ADAMS Accession Number ML16257A411)

4. SECY 11-0014, Enclosure 1, "The Use of Containment Accident Pressure in Reactor Safety Analysis," (ADAMS Accession Number ML102110167)

Response

The conservative GOTHIC model used the same inputs that formed the design basis (SHEX) code that generated the Extended Power Uprate (EPU) containment response. These results were used for the DBA-LOCA NPSH analysis at BSEP. Being different codes, the SHEX and GOTHIC cases are not identical. Regardless, the conservative responses are nearly equal as shown in the figure presented with the Criterion 1 response.

The following table shows the key initial conditions and containment input parameters that affect NPSHA used in both conservative analyses (i.e., GOTHIC 8.0 and SHEX) and the GOTHIC 8.0 realistic analysis. The basis for each parameter is also shown.

Comparison of Conservative and Realistic Input Values for
BSEP DBA-LOCA Containment Analyses

Input Parameter	Unit	Conservative Input Values	Realistic Input Values	Basis Conservative/Realistic
Initial Reactor Thermal Power	MWT	2981	2923	EPU Rated + 2% / EPU Rated
Decay Heat		Nominal + 2 sigma uncertainty	Nominal +0.11 sigma uncertainty	ANSI/ANS 5.1-1979 decay heat: Licensing Basis / Realistic estimate
Initial Suppression Pool Temperature	°F	95	95	Tech. Spec. maximum normal operating temperature
Service Water Temperature	°F	92	92	Maximum ultimate heat sink temperature
RHR Heat Exchanger K value	btu/sec-°F	235	Varies with Suppression Pool Inlet Temperature	UFSAR Table 6-4 / Realistic case: K = (SP Inlet Temp - 120) * 0.1 + 229.2
Initial Suppression Pool Volume	ft^3	86450	88100	TS Minimum / Nominal
Initial Drywell Temperature	°F	150	135	TS Maximum / Nominal (Drywell average air temp.)
Initial Drywell Pressure	psia	14.196	14.196	0.5 psi between DW and Torus Wet-well
Initial Wetwell Pressure	psia	14.696	14.696	Ambient
Relative Humidity for Drywell and Wetwell	%	100	100	Maximum
Ambient Pressure	psia	14.696	14.696	Sea Level Bar. Press.
Containment Leakage Rate (1 La)	wt. % / day	0.5	0.5	Tech. Spec. Limit

Note 1: The conservative case input parameters result in maximizing suppression pool temperature and minimizing the torus wet-well pressure. The impact is to minimize NPSHA and NPSH margin.

Note 2: The realistic case parameters result in conservative but a more realistic suppression pool temperature and wet-well pressure. Since some of the parameters were the same, the results are still conservative for NPSH.

NRC RAI SRXB-C-RAI 6

Regulatory Basis: 10 CFR, Part 50, Appendix A, GDC 38 states in part, the containment heat removal system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

To assure that the containment heat removal function is adequately performed during a design basis LOCA, the pumps that draw water from the suppression pool during LOCA should have a positive margin for the Net Positive Suction Head (NPSH). SECY-11-0014, Enclosure 1 (Reference 4), Section 6.6.6 states it is possible that the available NPSH may be less than the required NPSH (NPSHreff). It further states that the operation in this mode is acceptable if appropriate tests are done to demonstrate that the pump will continue to perform its safety functions under the applicable conditions given in Section 6.6.6 of Reference 4.

Refer to the following statement in Enclosure 11 (Reference 3) under heading “Design Basis LOCA” in response to SECY-11-0014 (Reference 4) Criteria 2:

> However, it was found that the limiting short term (<600 sec) RHR flow (i.e. two RHR pumps delivering a total of 21,100 gpm into a broken recirculation line) could not be maintained due to degraded NPSH margin. Input from the RHR pump manufacturer was obtained, which showed that the RHR pumps could operate at a reduced flow rate until the pumps could be throttled at 600 seconds. Since cavitation would occur during the initial 600 seconds, there is a concern of related damage. The manufacturer evaluated this condition and provided qualitative assurance the pumps could operate for this short time without damage.

Provide the pump manufacturer’s input, such as test reports, including the basis for qualifying the RHR pumps to operate satisfactorily while cavitating without any damage with the short term (<600 sec from LOCA initiation) flow rate of 21,100 gpm.

NRC RAI References

3. Enclosure 11 of Reference 1, "SECY-11-0014 Discussion - Use of Containment Accident Pressure (CAP) in Analyzing ECCS and Containment Heat Removal System Pump Performance," (ADAMS Accession Number ML16257A411)

4. SECY 11-0014, Enclosure 1, "The Use of Containment Accident Pressure in Reactor Safety Analysis," (ADAMS Accession Number ML102110167)

Response

In the NPSH analysis prepared for the SECY 11-0014, the limiting NPSH condition occurs with two RHR pumps delivering flow at maximum flow conditions during the short term (< 600 seconds) phase of the DBA-LOCA. No operator action or Containment Accident Pressure (CAP) is credited during this period. In this limiting scenario, in the short term (< 600 seconds), the flow from both pumps is lost out of the break (piping failure) and no credit for core or containment cooling is assumed. The analysis results indicate that NPSHA is less than $NPSHR_{eff}$ for approximately 400 seconds (from 213 seconds to 600 seconds).

The as-tested BSEP RHR pump curves were extrapolated through runout flow to generate new pump curves. The pump vendor (Flowserve) then generated NPSHR performance curves for the BSEP RHR pumps with an assumed 12% drop in total dynamic head (TDH). The 12% drop curve associated with an NPSHR of 19.90 ft. was used to assess pump performance.

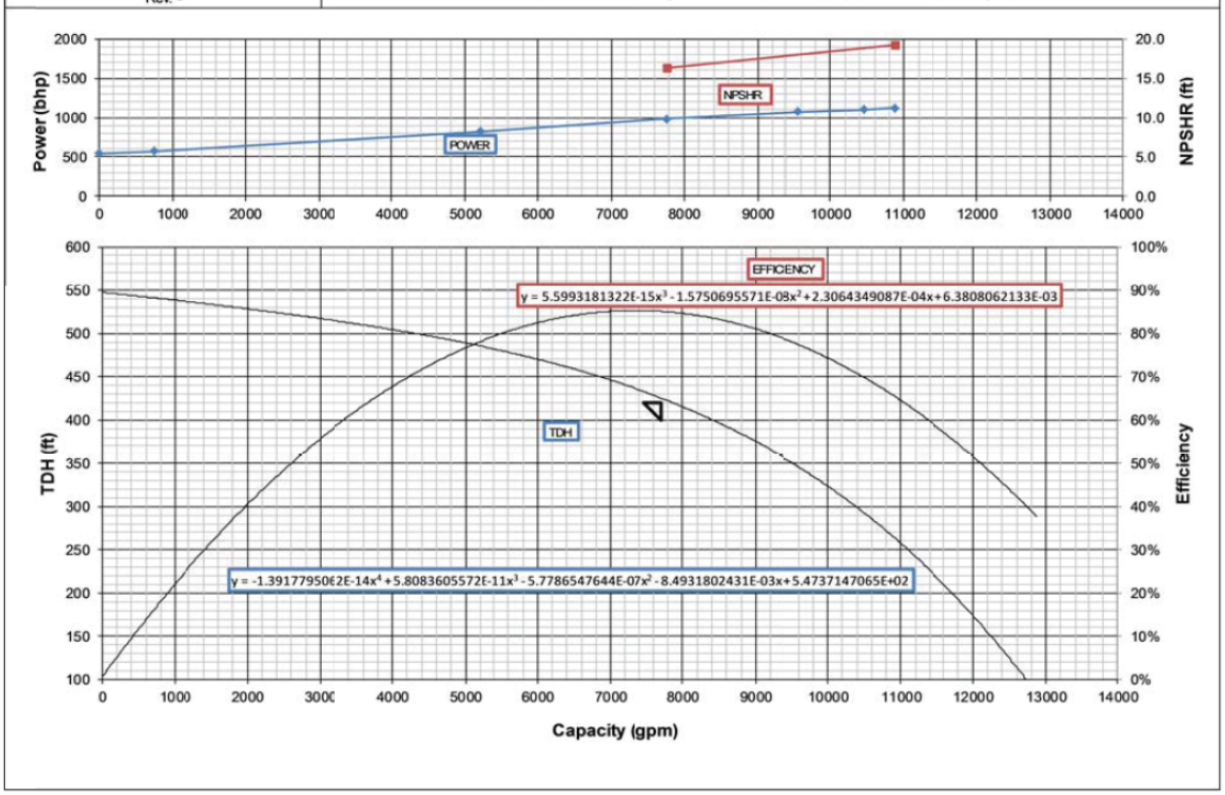
One BSEP RHR extrapolated pump curve and the NPSHR performance curves associated with various drops in TDH created by the vendor are shown below.

BSEP RHR Extrapolated Pump Curve

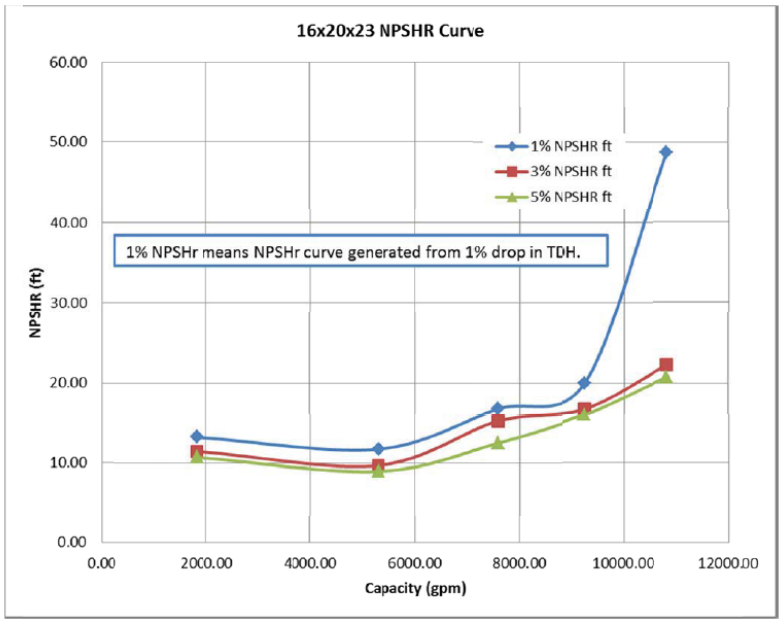
FLOWSERVE

Test Data

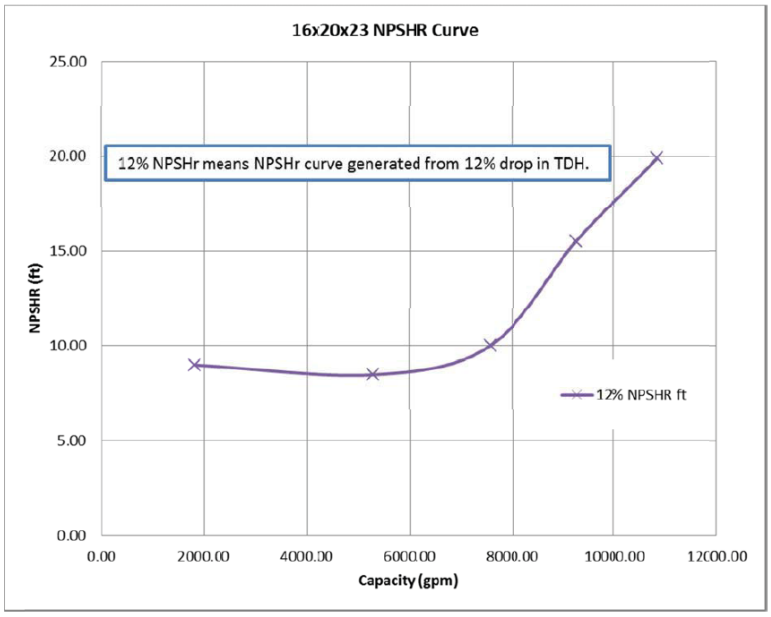
Order Number		Liquid	Water	Speed (RPM)	1780	Capacity (gpm)	7700
Customer		Temperature (°F)	82	Pump type	16X20X23 DVDS	TDH (ft)	420
Service		Viscosity (cP)	N/A	Based on curve no.	TC-3670	Power (bhp)	
Date	12/15/1971	Specific Gravity	1.0	Number of stages	1	NPSHR (ft)	
Item Number		Created by:				Diameter (in)	21.5 V-Cut, 3/8X2 underfiled
Rev. -		Curves are approximate and is based on Flowserve clearance. Pump is guaranteed for one set of conditions. Capacity, head, and efficiency guarantees are based on shop test and when handling clear, cold, fresh water at a temperature of not over 85 degrees.					



Byron Jackson 16x20x23 DVDS RHR Pump NPSHR
(1%, 3% and 5% THD Drop - 10,710 gpm)



Byron Jackson 16x20x23 DVDS RHR Pump NPSHR
(12% THD Drop - 10,710 gpm)



The following observations were extracted from the Flowserve report containing the plots above:

- "There was no report of surging or system instability during actual testing. The phenomenon like cavitation surging or vapor stalling of a pump during very low NPSHA is a very complex phenomenon and depends heavily on the system characteristics and pump system interaction."
- "Flowserve's experience during pump testing is that on many occasions pump NPSHR drops fall below 10% and the pumps operate at high cavitation noise, vibration and sometime flow fluctuation. Many a times pumps can be held relatively stable over these conditions. On other occasions the pump flow drops to a lower value."
- "During abnormal operating condition when the NPSHA falls below such that the head drop is 12% of head at rated or run out flow, the pump may exhibit high vibration, high noise, and fluctuation of flow and head. There is a small possibility that the pump system interaction during extreme cavitation condition may cause wider fluctuations in head and flow, called surging."
- "It is my opinion that the pumps should mechanically survive for approximately 7-8 minutes in such conditions."

The Flowserve assessment provides qualitative assurance that the pumps would survive this short period (from 213 seconds to 600 seconds) without credit for CAP thereby ensuring that the pumps are able to support long term containment cooling.

NRC RAI SRXB-C-RAI 7

Regulatory Basis: 10 CFR, Part 50, Appendix A, GDC 38 states in part, the containment heat removal system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. To assure that the containment heat removal function is adequately performed during a design basis LOCA, the pumps that draw water from the suppression pool during LOCA should have a positive margin for the Net Positive Suction Head (NPSH). SECY-11-0014, Enclosure 1 (Reference 4), Section 6.6.9 states:

> A realistic calculation of NPSHa [available NPSH] should be performed to compare with the NPSHa determined from the Monte Carlo 95/95 calculation.

Refer to the following statement in Enclosure 11 (Reference 3) under heading “Design Basis LOCA” in response to SECY-11-0014 (Reference 4) Criteria 3:

> The 95/95 analysis is performed to quantify uncertainties in the containment response evaluation by randomly selecting values of critical parameters within a probable range of values.

Please justify the validity/applicability of the results of 95/95 analysis that: (a) quantified uncertainties in the containment response; and (b) the critical input parameters that were randomly selected with the basis of selection and their range of values.

NRC RAI References

3. Enclosure 11 of Reference 1, "SECY-11-0014 Discussion - Use of Containment Accident Pressure (CAP) in Analyzing ECCS and Containment Heat Removal System Pump Performance," (ADAMS Accession Number ML16257A411)

4. SECY 11-0014, Enclosure 1, "The Use of Containment Accident Pressure in Reactor Safety Analysis," (ADAMS Accession Number ML102110167)

Response

A 95/95 analysis was not performed. In hindsight, this response could have been written more clearly. The intent of this statement is to describe the purpose of the 95/95 analysis as requested by the SECY. The next statement in the same paragraph contrasted this by stating that deterministic evaluations, versus statistical, using conservative and bounding inputs were performed. The purpose was to describe what was done for BSEP in lieu of the 95/95 analysis to accomplish the same goal. Further information that justified this approach as well as additional analysis details were provided to the NRC in response to SRXB-C-RAI-1 (Reference 7-1).

Duke Response Reference

7-1. Letter from William R. Gideon (Duke Energy) to the U.S. Nuclear Regulatory Commission Document Control Desk, *Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion*, dated April 6, 2017, ADAMS Accession Number ML17096A482