From:	Hon, Andrew
To:	Murray, William R. (Bill) (Bill.Murray@duke-energy.com) (Bill.Murray@duke-energy.com)
Subject:	Brunswick Unit 1 and Unit 2 Request for Additional Information related Containment Accident Pressure in the MELLLA+ LAR (CACs MF8864 and MF8865) (Non-Proprietary)
Date:	Monday, October 02, 2017 10:05:00 AM
Attachments:	BRUNSWICK 12 MELLLA PLUS - SRXB RAIs on CONTAINMENT Public.pdf

In a letter dated November 18, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16343A521), Duke Energy Progress (the licensee) requested the subject amendment to Operating Licenses OLs DPR-71 and DPR-62.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is attached. The proposed questions related to containment review were discussed by telephone with your staff on September 14, 2017. Your staff confirmed that the attached request for additional information (RAI):

1. was understood,
2. the NRC staff proposed proprietary information redaction is appropriate, and
3. you will provide a response in 30 days after receiving this request.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. Please note that if you do not respond to this request by the agreed-upon date or provide an acceptable alternate date, we may deny your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Section 2.108. If circumstances result in the need to revise the agreed upon response date, please contact me.

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REQUEST FOR ADDITIONAL INFORMATION

BY REACTOR SYSTEMS BRANCH

MAXIMUM EXTENDED LOAD LINE LIMIT PLUS ANALYSIS

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

By application dated September 6, 2016 (Agencywide Documents Access and Management System (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**]]**.

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.

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.

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 bounds 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.

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.

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 the 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.

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.

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).

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).