DUKE ENERGY®

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September 26, 2016

Serial: BSEP 16-0078

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
Request for License Amendment – Revision to Technical Specification 2.1.1.2
Minimum Critical Power Ratio Safety Limit

Ladies and Gentlemen:

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR Part 50.90), *Applications for Amendment of License, Construction Permit, or Early Site Permit*, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendment modifies the Minimum Critical Power Ratio Safety Limit values in TS 2.1.1.2 for both two recirculation loop and single recirculation loop operation.

Enclosure 1 provides a technical evaluation of the proposed change. Enclosures 2 and 3 provide the existing TS pages, marked up to show the proposed change. Enclosures 4 and 5 provide revised (i.e., typed) TS pages.

Enclosure 6 provides a copy of AREVA Document FS1-0028338 Revision 1.0, *Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Proprietary Version)*. The AREVA document contains information considered to be proprietary to AREVA; therefore, an affidavit supporting withholding the document from public disclosure is provided in Enclosure 7. A non-proprietary version of the document (i.e., AREVA Document FS1-0028339 Revision 1.0, *Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Non-Proprietary Version)* is provided in Enclosure 8.

Duke Energy has evaluated the proposed license amendment in accordance with 10 CFR 50/91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change involves no significant hazards consideration.

In accordance with 10 CFR 50.91, Duke Energy is providing a copy of the proposed license amendment to the designated representative for the State of North Carolina.

There are no new regulatory commitments contained in this submittal.

These changes are needed to support the next cycle of operation for BSEP, Unit 2 (i.e., Cycle 23), which is scheduled to begin March 18, 2017. Accordingly, Duke Energy requests

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approval of the proposed license amendments no later than March 31, 2017. Once approved, the Unit 2 amendment will be implemented prior to start-up from the 2017 Unit 2 refueling outage. For the Unit 1 changes, Duke Energy requests that implementation be deferred until prior to start-up from the 2018 Unit 1 refueling outage, which is scheduled to begin March 2018).

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager - Regulatory Affairs, at (910) 457-2487.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on September 26, 2016.

Sincerely,

William R. Gideon

Enclosures:

1. Evaluation of Proposed License Amendment Request
2. Technical Specification Page Mark-ups – Brunswick Steam Electric Plant, Unit 1
3. Technical Specification Page Mark-ups – Brunswick Steam Electric Plant, Unit 2
4. Typed Technical Specification Page – Brunswick Steam Electric Plant, Unit 1
5. Typed Technical Specification Page – Brunswick Steam Electric Plant, Unit 2
6. AREVA Document FS1-0028338 Revision 1.0, *Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Proprietary Version)* **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**
7. Affidavit Regarding AREVA Document FS1-0028338 Revision 1.0, *Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Proprietary Version)*
8. AREVA Document FS1-0028339 Revision 1.0, *Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Non-Proprietary Version)*

cc (with all enclosures):

U.S. Nuclear Regulatory Commission, Region II
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cc (with Enclosures 1 through 5, 7, and 8):

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Affidavit Regarding AREVA Document FS1-0028338 Revision 1.0,
Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23
MELLLA SLMCPR Analyses with SAFLIM3D Methodology (Proprietary Version)

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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

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

3. I am familiar with the AREVA information contained in the report FS1-0028338 Revision 1, "Brunswick Unit 1 Cycle 21 and Unit 2 Cycle 23 MELLLA SLMCPR Analyses with SAFLIM3D Methodology," dated August 2016 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document 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 AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA'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 AREVA.

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

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

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

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

8. AREVA 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.

SUBSCRIBED before me this 30th day of August, 2016.

SUSAN K MCCOY
NOTARY PUBLIC - WASHINGTON
MY COMMISSION EXPIRES 01-14-2020

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2020

Evaluation of Proposed License Amendment Request

Subject: Request for License Amendment – Revision to Technical Specification 2.1.1.2 Minimum Critical Power Ratio Safety Limit

1.0 Description

Duke Energy Progress, LLC (Duke Energy), is requesting Nuclear Regulatory Commission (NRC) approval of this proposed revision to the Technical Specifications (TS) of Renewed Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

The proposed change revises TS 2.1.1, *Reactor Core SLs*, by incorporating revised Safety Limit Minimum Critical Power Ratio (SLMCPR) values.

These changes are needed to support the next cycle of operation for BSEP, Unit 2 (i.e., Cycle 23), which is scheduled to begin March 18, 2017. Accordingly, Duke Energy requests approval of the proposed license amendments no later than March 31, 2017. Once approved, the Unit 2 amendment will be implemented prior to start-up from the 2017 Unit 2 refueling outage. For the Unit 1 changes, Duke Energy requests that implementation be deferred until prior to start-up from the 2018 Unit 1 refueling outage, which is scheduled to begin March 2018).

2.0 Assessment

2.1 Proposed Change:

TS 2.1.1.2 specifies the values for the SLMCPR. The current BSEP Unit 1 and Unit 2 TS state:

> MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

The proposed amendments will revise the SLMCPR values in TS 2.1.1.2 for two loop operation and single loop operation to state:

> MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.2 Technical Evaluation:

Methodologies or computer codes used to support licensing basis analyses are documented in topical reports which are reviewed by the NRC on a generic basis. The NRC, in its safety evaluation for the approved topical report, defines the basis for acceptance in conjunction with any limitations and conditions on use of the topical report, as appropriate. In situations where a plant-specific license amendment request references a generic topical report, plant-specific applicability of the material presented in the topical report is reviewed.

<u>System Description/Applicable Safety Analysis</u>

The BSEP, Unit 1 and 2 cores consist of ATRIUM 10XM fuel assemblies channeled with Zircaloy-4 AREVA fuel channels, with the Unit 1 Cycle 21 core also including Zircaloy-BWR

Beta-Quenched lead use channels as described in Reference 5.1, and the Unit 2 Cycle 23 core designed to continue irradiating eight ATRIUM 11 lead test fuel assemblies with Z4B-BQ lead use channels as described in Reference 5.2. Z4B-BQ is the same material as Zircaloy-BWR Beta-Quenched material.

Plant-Specific Methodology Applicability Evaluation

In Reference 5.4, the NRC approved plant-specific application of the methodology described in ANP-10307PA, Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors* to BSEP, as augmented by the following TS Appendix B Additional Condition:

> The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors* (i.e., TS 5.6.5.b.11) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.

The methodology described in ANP-10307PA, as augmented by this TS Appendix B Additional Condition, is used to determine the SLMCPR such that at least 99.9 percent of the fuel rods in the core will not experience dry-out during normal operation and anticipated operational occurrences if the core MCPR is greater than or equal to the SLMCPR. Enclosure 6 summarizes the methodology, inputs, and results supporting the BSEP Unit 1 Cycle 21 and Unit 2 Cycle 23 SLMCPR values calculated using the ANP-10307PA methodology as augmented by the associated TS Appendix B Additional Condition related to channel bow model uncertainty. The SLMCPR is determined using a statistical analysis that employs a Monte Carlo process that perturbs key input parameters used in the MCPR calculation based on their uncertainties. Table 1 in Enclosure 6 identifies these uncertainty inputs.

The fuel-related power distribution uncertainty inputs used by the ANP-10307PA methodology are calculated from separately determined uncertainty components described in EMF-2158(P)(A), *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*. The NRC-approved uncertainty components from EMF-2158(P)(A) are shown to be applicable to BSEP, Unit 1 and Unit 2, in Enclosure 3 of Reference 5.3. Plant related uncertainty inputs (i.e., feedwater flow rate, feedwater temperature, core pressure and total core flow uncertainties) are identified in Table 1 of Enclosure 6, and are consistent with the NRC-approved plant uncertainties reported from Topical Report NEDC-32601P-A, *Methodology and Uncertainties for Safety Limit MCPR Evaluations.*

ANP-10307PA incorporates a realistic fuel channel bow model. Model uncertainty is based on AREVA fuel channel measurements. Channel fluence is an input to the channel bow model, and is calculated based on BSEP-specific reactor core operating conditions. BSEP has not experienced indications of abnormal channel bow with AREVA channels. Should indications of abnormal channel bow be experienced by BSEP, the channel bow model will be applied in a conservative manner as described in ANP-10307PA. The ANP-10307PA channel bow model is applicable to BSEP because the model accounts for channel fluence calculated specific to the BSEP reactor core operating conditions. The channel bow model uncertainty is applicable to

BSEP because BSEP has not experienced abnormal channel bow with AREVA fuel channels, and because the uncertainty is augmented as required by the associated TS Appendix B Additional Condition related to channel bow model uncertainty.

Conformance with Methodology and Safety Evaluation Limitations

The NRC letter approving Topical Report ANP-10307PA states that the report is acceptable for referencing in licensing applications for boiling water reactors to the extent specified and under the limitations delineated in the topical report and in the final NRC safety evaluation. The final NRC safety evaluation concluded the topical report is acceptable for use for plant-specific licensing actions, without listing any limitations or conditions.

With the NRC's approval documented in Reference 5.4, Topical Report ANP-10307PA, Revision 0, was incorporated into the BSEP Unit 1 and 2 Technical Specifications, and Duke Energy has implemented the analytical methods described in the report and conforms with the methodology described in the report and the associated TS Appendix B Additional Condition related to channel bow model uncertainty.

The SLMCPR results determined using the ANP-10307PA methodology, as augmented by the associated TS Appendix B Additional Condition related to channel bow model uncertainty, are provided in Enclosure 6. These results support the requested TS SLMCPR two loop operation values of 1.07, and single loop operation SLMCPR values of 1.09, for BSEP Unit 1 Cycle 21 and BSEP Unit 2 Cycle 23. More restrictive values than those provided in Enclosure 6 are being requested to accommodate small cycle-to-cycle variations.

No plant hardware or operational changes are required with the proposed license amendment.

3.0 Regulatory Analysis

3.1 No Significant Hazards Consideration Analysis

The proposed license amendment involves revising the Safety Limit Minimum Critical Power Ratio (SLMCPR) values contained in Technical Specification (TS) 2.1.1.2 for two recirculation loop operation from ≥ 1.08 to ≥ 1.07, and from ≥ 1.11 to ≥ 1.09 for single loop recirculation operation, for both Brunswick Steam Electric Plant, Units 1 and 2. Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, *Issuance of amendment*, as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed SLMCPR values have been determined using NRC-approved methods discussed in AREVA Topical Report ANP-10307PA, Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors*, June 2011, as augmented by the associated TS Appendix B Additional Condition related to channel bow model uncertainty. Establishing a two recirculation loop SLMCPR value of ≥ 1.07 and a single recirculation

loop SLMCPR value of ≥ 1.09 ensures that the acceptance criteria continues to be met (i.e., at least 99.9 percent of all fuel rods in the core do not experience transition boiling).

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed license amendments do not involve any plant modifications or operational changes that could affect system reliability or performance, or that could affect the probability of operator error. As such, the proposed changes do not affect any postulated accident precursors. Since no individual precursors of an accident are affected, the proposed license amendments do not involve a significant increase in the probability of a previously analyzed event.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The basis for the SLMCPR calculation is to ensure that during normal operation and during anticipated operational occurrences, at least 99.9 percent of all fuel rods in the core do not experience transition boiling if the safety limit is not exceeded.

Based on these considerations, the proposed changes do not involve a significant increase in the consequences of a previously analyzed accident.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The SLMCPR is a TS numerical value calculated for two recirculation loop operation and single recirculation loop operation to ensure at least 99.9 percent of all fuel rods in the core do not experience transition boiling if the safety limit is not exceeded. SLMCPR values are calculated using NRC-approved methodology identified in the TS. The proposed SLMCPR values do not involve any new modes of plant operation or any plant modifications and do not directly or indirectly affect the failure modes of any plant systems or components. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The SLMCPR provides a margin of safety by ensuring that at least 99.9 percent of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences if the MCPR Safety Limit is not exceeded. Revision of the SLMCPR values in Technical Specification 2.1.1.2, using NRC-approved methodology, will ensure that the current level of fuel protection is maintained by continuing to ensure that the fuel design safety criterion is met (i.e., that no more than 0.1 percent of the rods are expected to be in boiling transition if the MCPR Safety Limit is not exceeded). Therefore, the proposed amendments do not result in a significant reduction in the margin of safety.

Based on the above, Duke Energy concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Commitments

There are no new regulatory commitments contained in this submittal.

3.3 Applicable Regulatory Requirements

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Duke Energy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and do not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR).

As stated in the NRC's "Safety Evaluation of the Brunswick Steam Electric Station Units 1 and 2," dated November 1973, BSEP meets the intent of the General Design Criteria (GDC), published in the *Federal Register* on May 21, 1971, as Appendix A to 10 CFR Part 50. The proposed changes do not affect compliance with the intent of the GDCs. In particular, the intent of GDC 10, *Reactor design*, continues to be met.

GDC 10 states:

> 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.

10 CFR 50, Appendix A, GDC 10 requires that the reactor core and associated coolant, control, and protection systems 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 (AOOs). To ensure compliance with GDC 10, Duke Energy has performed plant-specific critical power limit analyses using NRC-approved methodologies. The MCPR Safety Limit ensures that sufficient conservatism exists in the operating limit MCPR such that, in the event of an anticipated operational occurrence, there is a reasonable expectation that at least 99.9 percent of the fuel rods in the core will avoid boiling transition for the power distribution within the core including uncertainties.

10 CFR 50.36, *Technical specifications*, paragraph (c)(1), requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The fuel cladding integrity SLMCPR is established to assure that at least 99.9% of the fuel rods in the core do not experience boiling transition during normal operation and AOOs. Thus, the SLMCPR is required to be contained in TS. The proposed amendments to the BSEP, Unit 1 and 2 TS do not remove the SLMCPR from the TS.

10 CFR 50.36(c)(5) states that the Technical Specifications will include administrative controls that address the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe

manner. The Core Operating Limits Report (COLR) is required as a part of the reporting requirements specified in the Brunswick Technical Specifications Administrative Controls section. The Technical Specifications require the core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and to be documented in the COLR. In addition, it requires the analytical methods used to determine the core operating limits to be those that have been previously reviewed and approved by the NRC, and specifically to be those described in Technical Specification 5.6.5.b. The proposed amendments ensure that these requirements are met.

3.4 Precedents

By letter dated June 14, 2011 (i.e., Reference 5.5), Topical Report ANP-10307P, Revision 0, was approved by the NRC. The NRC letter approving Topical Report ANP-10307P states that the report is acceptable for referencing in licensing applications for boiling water reactors to the extent specified and under the limitations delineated in the topical report and in the final NRC safety evaluation. The final NRC safety evaluation concluded the topical report is acceptable for use for plant-specific licensing actions, without listing any limitations or conditions.

By letter dated March 1, 2013 (i.e., Reference 5.4), Topical Report ANP-10307PA, Revision 0, was approved by the NRC as an analytical method used to determine the core operating limits for BSEP. Subsequent to that approval, ANP-10307PA has been used for determining core operating limits for Unit 1 Cycle 20, Unit 2 Cycle 21, and Unit 2 Cycle 22.

License amendments to revise SLMCPR values for other facilities that have been recently issued by the NRC include the Hatch Nuclear Plant, Unit 1 (i.e., References 5.6 and 5.7), the Browns Ferry Plant, Unit 1 (i.e., References 5.8 and 5.9), and the Nine Mile Point Nuclear Station, Unit 2 (i.e., References 5.10 and 5.11).

3.5 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

4.0 Environmental Consideration

A review has determined that the proposed amendments do not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, *Standards for Protection Against Radiation*, and do not change an inspection or surveillance requirement. The proposed amendments do not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Because the proposed amendment meets the eligibility criterion for categorical exclusion and does not involve a special circumstance, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

5.0 References

5.1 Letter from Pedro Salas (AREVA Inc.) to the U.S. Nuclear Regulatory Commission Document Control Desk, "Informational Transmittal Regarding the Zircaloy-BWR Beta-Quenched Lead Use Channel Program at Brunswick Steam Electric Plant, Unit 1," dated April 10, 2014, ADAMS Accession Number ML14104B552.

5.2 Letter from Pedro Salas (AREVA Inc.) to the U.S. Nuclear Regulatory Commission Document Control Desk, "Informational Transmittal Regarding the ATRIUM 11 Lead Test Assembly Program at Brunswick Nuclear Plant, Unit 2," dated September 30, 2015, ADAMS Accession Number ML15275A289.

5.3 Letter from William Jefferson, Jr. (Duke Energy) to U.S. Nuclear Regulatory Commission Document Control Desk, "Response to Additional Information Request Supporting License Amendment Requests for Addition of Analytical Methodology Topical Reports to Technical Specification 5.6.5 (NRC TAC Nos. ME3856, ME3857, ME3858, and ME3859)," dated November 18, 2010, ADAMS Accession Number ML103330242.

5.4 Letter from Christopher Gratton (NRC) to Michael J. Annacone (Duke Energy), "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments Regarding Addition of Analytical Methodology Topical Reports to Technical Specification 5.6.5 and Revision to Minimum Critical Power Ratio Safety Limit (TAC Nos. ME8135 and ME8136)", dated March 1, 2013, ADAMS Accession Number ML13037A551.

5.5 Letter from Robert A. Nelson (NRC) to Pedro Salas (AREVA NP Inc.), "Final Safety Evaluation for AREVA NP, Inc. Topical Report ANP-10307P, Revision 0, 'AREVA MCPR [Minimum Critical Power Ratio] Safety Limit Methodology for Boiling Water Reactors' (TAC No. ME2914)," dated June 14, 2011, ADAMS Accession Number ML11140A125.

5.6 Letter from C. R. Pierce (Southern Nuclear) to U.S. Nuclear Regulatory Commission Document Control Desk, *License Amendment Request Concerning Safety Limit Minimum Critical Power Ratio*, dated September 1, 2015, ADAMS Accession Number ML15252A186.

5.7 Letter from Bob Martin (NRC) to C. R. Pierce (Southern Nuclear), *Issuance of Amendment Regarding Minimum Critical Power Ratio (CAC No. MF6681)*, dated January 29, 2016, ADAMS Accession Number ML15342A398.

5.8 Letter from J. W. Shea (TVA) to U.S. Nuclear Regulatory Commission Document Control Desk, *Application to Modify Technical Specification 2.1.1.2, Reactor Core Minimum Critical Power Ratio Safety Limits (TS-506)*, dated September 25, 2015, ADAMS Accession Number ML15268A566.

5.9 Letter from Farideh E. Saba (NRC) to J. W. Shea (TVA), *Issuance of Amendment to Revise Technical Specifications Related to Cycle 12 Safety Limit Minimum Critical Power Ratio (CAC No. MF6760)*, dated April 26, 2016, ADAMS Accession Number ML16028A414.

5.10 Letter from James Barstow (Exelon Generating Company, LLC) to U.S. Nuclear Regulatory Commission Document Control Desk, *License Amendment Request – Safety*

Limit Minimum Critical Power Ratio Change, dated September 3, 2015, ADAMS Accession Number ML15252A204.

5.11 Letter from Brenda L. Mozafari (NRC) to Bryan C. Hanson (Exelon Nuclear), *Issuance of Amendment Re: Technical Specifications for Safety Limit Minimum Critical Power Ratio (CAC No. MF6714)*, dated January 8, 2016, ADAMS Accession Number ML15341A336.

Technical Specification Page Mark-ups – Brunswick Steam Electric Plant, Unit 1

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

1.07 1.09

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Technical Specification Page Mark-ups – Brunswick Steam Electric Plant, Unit 2

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

1.07 1.09

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ $\boxed{1.08}$ for two recirculation loop operation or ≥ $\boxed{1.11}$ for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Typed Technical Specification Page – Brunswick Steam Electric Plant, Unit 1

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Typed Technical Specification Page – Brunswick Steam Electric Plant, Unit 2

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.