UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 7, 2019

Mr. William R. Gideon
Site Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd., SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 287 AND 315 TO ADOPT TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-551, REVISION 3, "REVISE SECONDARY CONTAINMENT SURVEILLANCE REQUIREMENTS" (EPID L-2018-LLA-0011)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 287 and 315 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2, respectively. These amendments are in response to your license amendment request dated January 23, 2018.

The amendments revise Technical Specification 3.6.4.1, "Secondary Containment," Surveillance Requirement 3.6.4.1.2, to allow for the temporary opening of the inner and outer doors of secondary containment for the purpose of entry and exit. The changes are consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements," dated October 3, 2016.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* Notice.

Sincerely,

Dennis J Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 287 to DPR-71
2. Amendment No. 315 to DPR-62
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 287
Renewed License No. DPR-71

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated January 23, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 287, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: January 7, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 287

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace page 6 of Renewed Facility Operating License No. DPR-71 with the attached revised page 6.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3.6-29	3.6-29

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 287, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening, except when the access opening is being used for entry and exit.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	In accordance with the Surveillance Frequency Control Program

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 315
Renewed License No. DPR-62

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated January 23, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 315, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: January 7, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 315

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace page 6 of Renewed Facility Operating License No. DPR-62 with the attached revised page 6.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3.6-29	3.6-29

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180[th] day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level
The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 315, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233,

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening, except when the access opening is being used for entry and exit.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENTS NOS. 287 AND 315

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By application dated January 23, 2018 (Reference 1), Duke Energy Progress, LLC (the licensee) requested a change to the technical specifications (TSs) for Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick). Specifically, the licensee requested a change to the TSs to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements," dated October 3, 2016 (Reference 2). The U.S. Nuclear Regulatory Commission (NRC or the Commission) approved TSTF-551 on September 21, 2017 (Reference 3).

The proposed change would revise TS 3.6.4.1, "Secondary Containment," Surveillance Requirement (SR) 3.6.4.1.2. Specifically, SR 3.6.4.1.2 would be modified to permit secondary containment access openings to be open to permit entry and exit.

2.0 REGULATORY EVALUATION

2.1 System Description

The secondary containment is a structure that encloses the primary containment, including components that may contain primary system fluid. The safety function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a design-basis accident (DBA) to ensure that the control room operator and offsite doses are within the regulatory limits. There is no redundant train or system that can perform the secondary containment function should the secondary containment be inoperable.

The secondary containment boundary is the combination of walls, floor, roof, ducting, doors, hatches, penetrations, and equipment that physically form the secondary containment. Routinely used secondary containment access openings contain at least one inner and one outer door in an airlock configuration. In some cases, secondary containment access openings are shared such that there are multiple inner or outer doors. All secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit of personnel, equipment, or material.

Secondary containment operability is based on its ability to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. To prevent ground level exfiltration of radioactive material while allowing the secondary containment to be designed as a mostly conventional structure, the secondary containment requires support systems to maintain the pressure at less than atmospheric pressure. During normal operation, nonsafety-related systems are used to maintain the secondary containment at a slight negative pressure to ensure that any leakage is into the building and that any secondary containment atmosphere exiting is via a pathway monitored for radioactive material. However, during normal operation, it is possible for the secondary containment vacuum to be momentarily less than the required vacuum for a number of reasons such as during wind gusts or swapping of the normal ventilation subsystems.

During emergency conditions, the standby gas treatment (SGT) system is designed to be capable of drawing down the secondary containment to a required vacuum within a prescribed time and continue to maintain the negative pressure as assumed in the accident analysis. For Brunswick, the SGT system must be able to establish the required vacuum within 10 minutes. The leak-tightness of the secondary containment, together with the SGT system, ensure that radioactive material is either contained in the secondary containment or filtered through the SGT system filter trains before being discharged to the outside environment via the elevated release point.

2.2 Proposed TSs Change

Brunswick SR 3.6.4.1.2 requires verification that one secondary containment access door in each access opening is closed. The proposed change would modify this SR by adding the following phrase to the end of the SR statement, "...except when the access opening is being used for entry and exit." The proposed change would allow for the temporary opening of the inner and outer doors of secondary containment for the purpose of entry and exit (i.e., normal opening and prompt closure of a door for transit).

2.3 Regulatory Requirements and Guidance

The regulation at Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(a)(1) requires an applicant for an operating license to include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS bases "shall not become part of the technical specifications."

Additionally, 10 CFR 50.36(b) states:

> Each license authorizing operation of a ... utilization facility ... will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The categories of items required to be included in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection, to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The NRC staff's guidance for review of TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), dated March 2010 (Reference 4).

SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (Reference 5), provides guidance to the NRC staff for the review of alternate source term (AST) amendment requests. It states that the NRC reviewer should evaluate the proposed change against the guidance in NRC Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (Reference 6).

RG 1.183 provides an acceptable methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67, "Accident source term." RG 1.183 provides guidance to licensees on AST submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

The regulation at 10 CFR 50.67(b)(2) states:

> The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem [roentgen equivalent man])[1] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

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[1] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application in accordance with the regulations and guidance discussed in Section 2.3 of this safety evaluation and the NRC-approved TSTF-551, Revision 3. In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. In making its determination as to whether to amend the licenses, the NRC staff considered those regulatory requirements that are automatically conditions of the licenses through 10 CFR 50.54.

The regulation at 10 CFR 50.36(a)(1) states, in part: "A summary statement of the bases or reasons for such specifications ... shall also be included in the application, but shall not become part of the technical specifications." Accordingly, along with the proposed TS change, the licensee also submitted a TS Bases change that corresponds to the proposed TS change for information only.

3.1 Proposed Change to SR 3.6.4.1.2

The NRC staff review was limited to the licensee's request to provide an allowance for the brief, inadvertent, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions. Planned activities that could result in the simultaneous opening of redundant secondary containment access openings, such as maintenance of a secondary containment personnel access door or movement of large equipment through the openings that would take longer than the normal transit time, were considered outside the scope of the NRC staff's review.

The NRC staff reviewed the proposed change to SR 3.6.4.1.2. The NRC staff determined that the SR continues to provide appropriate confirmation that secondary containment boundary doors are properly positioned and capable of performing their function in preserving the secondary containment boundary. The NRC staff determined that the SR continues to appropriately verify the operability of the secondary containment and provides assurance that the necessary quality of systems and components are maintained in accordance with 10 CFR 50.36(c)(3).

Additionally, the NRC staff evaluated the impact of modifying the licensee's TSs to allow secondary containment access openings to be open for entry and exit on the licensee's design-basis radiological consequence dose analyses to ensure that the modification will not result in an increase in the radiation dose consequences and that the resulting calculated radiation doses will remain within the design criteria specified in the current radiological consequence analyses. The NRC staff's review of these DBAs determined that there is one DBA that takes credit for the secondary containment, and it is possibly impacted by the brief, inadvertent, simultaneous opening of both an inner and outer access door during normal entry and exit conditions, the loss-of-coolant accident (LOCA).

3.1.1 LOCA

Following a LOCA, the secondary containment structure is maintained at a negative pressure, ensuring that leakage from primary containment to secondary containment can be collected and filtered prior to release to the environment. The SGT system performs the function of maintaining a negative pressure within the secondary containment, as well as collecting and filtering the leakage from primary containment. The licensee credits the SGT system for mitigation of the radiological releases from the secondary containment. In the LOCA analysis, the secondary containment drawdown analysis assumes that the SGT system can draw down the secondary containment within 10 minutes. SR 3.6.4.1.3 verifies each SGT subsystem can maintain greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment for 1 hour at a flow rate of less than or equal to 3,000 cubic feet per minute.

Conservatively, the DBA LOCA radiological consequence analysis in the Brunswick Updated Final Safety Analysis Report, Section 15.6.4.3 (Reference 7), assumes that following the start of a DBA LOCA, the secondary containment pressure of 0.25 inches of vacuum water gauge is achieved at approximately 10 minutes. The license assumes that releases into the secondary containment prior to the 10-minute drawdown time leak directly to the environment as a ground level release with no filtration. After the assumed 10-minute drawdown, these releases are filtered by the SGT system and released via the SGT system exhaust vent.

Based on this information, the NRC staff concludes that the licensee's DBA LOCA analysis has sufficient conservatism by assuming a drawdown time of 10 minutes from the start of the DBA LOCA. As a result of this drawdown time, sufficient margin exists to ensure that the secondary containment can be reestablished during a brief, inadvertent, simultaneous opening of the inner and outer doors, and there is reasonable assurance that a failure of a safety system needed to control the release of radioactive material to the environment will not result. The brief, inadvertent, simultaneous opening of the secondary containment access doors does not impact the design bases and will not result in an increase in any onsite or offsite dose.

Based on the above discussion, the NRC staff finds that the licensee's proposed change to the TSs does not impact the licensee's design-basis LOCA radiological consequence analysis and will not result in an increase in any onsite or offsite dose. Therefore, the NRC staff concludes that this change is acceptable with respect to the radiological consequences of the DBAs.

The licensee was approved for AST methodology and the radiological dose consequences analyses for DBAs by License Amendment Nos. 221 and 246 for Brunswick, dated May 30, 2002 (Reference 8). The NRC staff reviewed the impact of the proposed change to the Brunswick TSs on all DBAs currently analyzed in the Brunswick Updated Final Safety Analysis Report that could have the potential for significant dose consequences. Chapter 15 of the Updated Final Safety Analysis Report describes the DBAs and their radiological consequence analysis results.

3.1.2 Conclusion

As described above, the NRC staff reviewed the technical basis provided by the licensee to assess the radiological impacts of the proposed change to SR 3.6.4.1.2. The NRC staff finds that the proposed change to SR 3.6.4.1.2 is consistent with the regulatory requirements and guidance identified in Section 2.3 of this safety evaluation. The NRC staff finds that, with the proposed change, the TSs will continue to comply with these criteria and the licensee's estimates of the dose consequences of a design-basis LOCA will continue to comply with the requirements of the current radiological consequence analyses. Therefore, the proposed change is acceptable with regard to the radiological consequences of the postulated DBAs.

3.2 Variations from the NRC-Approved TSTF-551, Revision 3

The licensee is proposing the following variations from the TS changes described in the NRC-approved TSTF-551, Revision 3, or the applicable parts of TSTF-551, or the NRC staff's safety evaluation for TSTF-551. These variations do not affect the applicability of TSTF-551 or the NRC staff's safety evaluation for TSTF-551 to the proposed license amendments.

- The Brunswick TSs do not contain an SR equivalent to SR 3.6.4.1.1 in TSTF-551. Therefore, the addition of the SR 3.6.4.1.1 note, as provided by TSTF-551, is not applicable.

- The Brunswick TSs do not contain an SR equivalent to SR 3.6.4.1.4 in TSTF-551. Therefore, the editorial change to SR 3.6.4.1.4, as provided by TSTF-551, is not applicable.

- The Brunswick TSs utilize different SR numbering than the Standard Technical Specifications on which TSTF-551 was based. Specifically, Brunswick SR 3.6.4.1.2 is equivalent to Standard Technical Specifications SR 3.6.4.1.3. This difference is administrative and does not affect the applicability of TSTF-551 to the Brunswick TSs.

- TSTF-551 and the model safety evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR Part 50, Appendix A, General Design Criteria (GDC). As stated in the U.S. Atomic Energy Commission's "Safety Evaluation of the Brunswick Steam Electric Station Units 1 and 2," dated November 1973 (Reference 9), Brunswick meets the intent of the GDC published in the *Federal Register* on May 21, 1971, as Appendix A to 10 CFR Part 50. This difference does not alter the conclusion that the proposed change is applicable to Brunswick.

- The final model safety evaluation for TSTF-551 discusses that the NRC staff's review determined that there are two DBAs that take credit for the secondary containment and are possibly impacted by the brief, inadvertent, simultaneous opening of both an inner and outer access door during normal entry and exit conditions – the LOCA and the fuel handling accident (FHA) in secondary containment. The Brunswick FHA does not credit the secondary containment or the SGT system for mitigation of fuel handling accidents. Because the Brunswick FHA radiological consequence analysis does not credit the secondary containment or the SGT system, the FHA in secondary containment analysis is not impacted by the brief, inadvertent, simultaneous opening of both an inner and outer access door during normal entry and exit conditions. This difference does not alter the conclusion that the proposed change is applicable to Brunswick.

3.3 Summary

The NRC staff reviewed the proposed change and determined that it meets the standards for TSs in accordance with 10 CFR 50.36(b). The proposed SR assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met and satisfies 10 CFR 50.36(c)(3). Additionally, the change to the TSs was reviewed for technical clarity and consistency with customary terminology and format in accordance with SRP Chapter 16.

Additionally, the NRC staff evaluated the impact of the proposed change on the design-basis radiological consequence analyses against the regulatory requirements and guidance identified in Section 2.3 of this safety evaluation. The NRC staff finds that, with the proposed change, the TSs will continue to comply with the requirements of the current radiological consequence analyses. Therefore, the proposed change is acceptable with regard to the radiological consequences of the postulated DBAs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments on September 20, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on June 19, 2018 (83 FR 28458). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

7.0 REFERENCES

1. Gideon, William R., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 - Application to Revise Technical Specifications to Adopt TSTF-551, 'Revise Secondary Containment Surveillance Requirements,'" dated January 23, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18023A896).

2. Technical Specifications Task Force Improved Standard Technical Specifications Change Traveler TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements," dated October 3, 2016 (ADAMS Accession No. ML16277A226).

3. Whitman, Jennifer M., U.S. Nuclear Regulatory Commission, letter to Technical Specifications Task Force, "Final Safety Evaluation of Technical Specifications Task Force Traveler TSTF-551, Revision 3, Revise Secondary Containment Surveillance Requirements (CAC No. MF5125)," dated September 21, 2017 (ADAMS Package Accession No. ML17236A365).

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Date: January 7, 2019

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 287 AND 315 TO ADOPT TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-551, REVISION 3, "REVISE SECONDARY CONTAINMENT SURVEILLANCE REQUIREMENTS" (EPID L-2018-LLA-0011) DATED JANUARY 7, 2019

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