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

July 12, 2017

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING REQUEST TO ADOPT TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-545 (CAC NOS. MF8359 AND MF8360)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 278 and 306 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2, respectively. The amendments are in response to your application dated August 29, 2016. The amendments allow the adoption of NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS [Technical Specification] Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (Agencywide Documents Access and Management System Accession No. ML15294A555).

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

Sincerely,

Andrew Hon, 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. 278 to DPR-71
2. Amendment No. 306 to DPR-62
3. Safety Evaluation

cc w/enclosures: Distribution via 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. 278
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, dated August 29, 2016, 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. 278, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: July 12, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 278

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 page 6.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	**Insert Pages**
1.1-3	1.1-3
3.1-22	3.1-22
3.4-5	3.4-5
3.6-12	3.6-12
3.6-13	3.6-13
5.0-9	5.0-9
5.0-10	5.0-10
5.0-11	5.0-11
5.0-12	5.0-12
5.0-13	5.0-13
5.0-14	5.0-14
5.0-15	5.0-15
5.0-16	5.0-16
5.0-17	5.0-17
5.0-17a	5.0-17a

(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. 278, 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

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1190 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.8	Verify sodium pentaborate enrichment is ≥ 47 atom percent B-10.	Prior to addition to SLC tank

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs)

LCO 3.4.3 The safety function of 10 SRVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required 10 SRVs are as follows: Number of SRVs — Setpoint (psig) 4 — 1130 ± 33.9 4 — 1140 ± 34.2 3 — 1150 ± 34.5	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.2	------------------------------NOTES------------------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions, is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.3	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.4	Verify the isolation time of each power operated and each automatic PCIV, except for MSIVs, is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.5	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.9	Verify leakage rate through each main steam line is $\leq$ 100 scfh and the combined leakage rate of all four main steam lines is $\leq$ 150 scfh when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

5.5.4 Radioactive Effluent Controls Program (continued)

h. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives > 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

i. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

k. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Table 5.3.3-2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Deleted

(continued)

5.5 Programs and Manuals (continued)

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber filter bank; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of charcoal adsorber operation; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 1, Positions C.5.a and C.5.c, and ANSI N510-1975 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
Standby Gas Treatment (SGT) System	2700 to 3300
Control Room Emergency Ventilation (CREV) System	1800 to 2200

(continued)

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 1, Positions C.5.a and C.5.d, and ANSI N510-1975 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	2700 to 3300
CREV System	1800 to 2200

c. 1) Demonstrate for the SGT System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 1, Position C.6.b, and tested in accordance with ASTM D3803-1989, at a temperature of 30°C, a face velocity of 61 fpm, and a relative humidity of 70% within the tolerances provided in Table 1 of ASTM D3803-1989, shows the methyl iodide penetration < 0.5%.

2) Demonstrate for the CREV System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 1, Position C.6.b, and tested in accordance with ASTM D3803-1989, at a temperature of 30°C and a relative humidity of 95% within the temperature and humidity tolerances provided in Table 1 of ASTM D3803-1989, meets the acceptance criteria of < 5.0% penetration of methyl iodide.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilter (SGT only), and the charcoal adsorbers is less than or equal to the value specified below when tested at the system flowrate specified as follows:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	8.5	2700 to 3300
CREV System	5.25	1800 to 2200

(continued)

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate that the heaters for each of the SGT subsystems dissipate ≥ 16.67 kW under a degraded voltage condition when tested in accordance with ANSI N510-1975.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is ≤ 10 Curies, excluding tritium and dissolved or entrained gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall establish required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil;

(continued)

5.5.9 Diesel Fuel Oil Testing Program (continued)

b. Kinematic viscosity is within limits for ASTM 2-D fuel oil when tested every 92 days; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with the applicable ASTM Standard.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.10.b.1 or 5.5.10.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

(continued)

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995, as modified by the following exceptions:

a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

(continued)

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each air lock door, leakage rate is $\leq$ 5 scfh when the gap between the door seals is pressurized to $\geq$ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

(continued)

5.5.13 Control Room Envelope Habitability Program (continued)

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the CREV System, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.14 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

(continued)

5.5 Programs and Manuals

5.5.14 Surveillance Frequency Control Program (continued)

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
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, dated August 29, 2016, 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. 306, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: July 12, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 306

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	**Insert Pages**
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5.0-10	5.0-10
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5.0-12	5.0-12
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5.0-14	5.0-14
5.0-15	5.0-15
5.0-16	5.0-16
5.0-17	5.0-17
5.0-17a	5.0-17a

(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; 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. 306, 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, 1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1190 psig	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.8	Verify sodium pentaborate enrichment is ≥ 47 atom percent B-10.	Prior to addition to SLC tank

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs)

LCO 3.4.3 The safety function of 10 SRVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required 10 SRVs are as follows:		In accordance with the INSERVICE TESTING PROGRAM
	Number of SRVs	Setpoint (psig)	
	4	1130 ± 33.9	
	4	1140 ± 34.2	
	3	1150 ± 34.5	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.2	------------------------------NOTES------------------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions, is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.3	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.4	Verify the isolation time of each power operated and each automatic PCIV, except for MSIVs, is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.5	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.9	Verify leakage rate through each main steam line is ≤ 100 scfh and the combined leakage rate of all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

5.5.4 Radioactive Effluent Controls Program (continued)

h. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives > 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

i. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

k. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Table 5.3.3-2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Deleted

(continued)

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber filter bank; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of charcoal adsorber operation; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 1, Positions C.5.a and C.5.c, and ANSI N510-1975 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
Standby Gas Treatment (SGT) System	2700 to 3300
Control Room Emergency Ventilation (CREV) System	1800 to 2200

(continued)

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 1, Positions C.5.a and C.5.d, and ANSI N510-1975 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	2700 to 3300
CREV System	1800 to 2200

c. 1) Demonstrate for the SGT System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 1, Position C.6.b, and tested in accordance with ASTM D3803-1989, at a temperature of 30°C, a face velocity of 61 fpm, and a relative humidity of 70% within the tolerances provided in Table 1 of ASTM D3803-1989, shows the methyl iodide penetration < 0.5%.

c. 2) Demonstrate for the CREV System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 1, Position C.6.b, and tested in accordance with ASTM D3803-1989, at a temperature of 30°C and a relative humidity of 95% within the temperature and humidity tolerances provided in Table 1 of ASTM D3803-1989, meets the acceptance criteria of < 5.0% penetration of methyl iodide.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilter (SGT only), and the charcoal adsorbers is less than or equal to the value specified below when tested at the system flowrate specified as follows:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	8.5	2700 to 3300
CREV System	5.25	1800 to 2200

(continued)

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate that the heaters for each of the SGT subsystems dissipate $\geq$ 16.67 kW under a degraded voltage condition when tested in accordance with ANSI N510-1975.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is $\leq$ 10 Curies, excluding tritium and dissolved or entrained gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall establish required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil;

(continued)

5.5 Programs and Manuals

5.5.9 Diesel Fuel Oil Testing Program (continued)

b. Kinematic viscosity is within limits for ASTM 2-D fuel oil when tested every 92 days; and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days in accordance with the applicable ASTM Standard.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.10.b.1 or 5.5.10.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995, as modified by the following exceptions:

a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

(continued)

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq$ 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 La for Type B and C tests and $\leq$ 0.75 La for Type A tests.

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each air lock door, leakage rate is $\leq$ 5 scfh when the gap between the door seals is pressurized to $\geq$ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

(continued)

5.5.13 Control Room Envelope Habitability Program (continued)

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the CREV System, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.14 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

(continued)

5.5.14 Surveillance Frequency Control Program (continued)

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in 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 AMENDMENT NOS. 278 AND 306

TO RENEWED FACILITY OPERATING LICENSES 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 August 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16252A220), Duke Energy Progress, Inc. (the licensee) requested changes to the Technical Specifications (TSs) for Brunswick Steam Electric Plant (Brunswick), Units 1 and 2. Specifically, the licensee requested changes to the TSs consistent with Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications (STS) Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (ADAMS Accession No. ML15294A555). The proposed no significant hazards consideration determination was published in the *Federal Register* (FR) 81 FR 87967 on December 6, 2016.

The licensee's proposed changes delete Brunswick, Units 1 and 2, TS 5.5.6, "Inservice Testing Program," and add a new defined term, "INSERVICE TESTING PROGRAM," to the TSs. All existing references to the "Inservice Testing Program" in the Brunswick, Units 1 and 2, TS SRs are replaced with "INSERVICE TESTING PROGRAM" so that the SRs refer to the new definition in lieu of the deleted program.

The licensee's letter dated August 29, 2016, also included a request to use American Society of Mechanical Engineers (ASME) Code Case OMN-20, "Inservice Test Frequency," as an alternative to certain ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) requirements at Brunswick, Units 1 and 2. The U.S. Nuclear Regulatory Commission (NRC or the Commission) considered this request separately from the proposed license amendment and authorized the licensee's use of this alternative by letter dated February 16, 2017 (ADAMS Accession No. ML17038A552).

2.0 REGULATORY EVALUATION

2.1 Description of Inservice Testing Requirements and TSTF-545

An inservice test is a test to assess the operational readiness of a structure, system, or component after first electrical generation by nuclear heat. The ASME OM Code provides requirements for inservice testing (IST) of certain components in light-water nuclear power plants. The ASME OM Code identifies the components subject to the testing (i.e., pumps, valves, pressure relief devices, and dynamic restraints), responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and recordkeeping. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(f), "Inservice testing requirements," requires that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda. The facility's TSs also prescribe IST requirements and frequencies for ASME Code Class 1, 2, and 3 components.

The regulation in 10 CFR 50.55a(f)(5)(ii) states, in part, "[i]f a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program." TSTF-545, Revision 3, provides guidance to licensees on how to request license amendments that would eliminate conflicting requirements between 10 CFR 50.55a, "Codes and standards," and the TSs. TSTF-545, Revision 3, proposes elimination of the IST program from the Administrative Controls section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the IST program. The elimination of the IST program from the TSs could cause uncertainty regarding the correct application of these SRs. Therefore, TSTF-545, Revision 3, also proposes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs, which would be defined as "the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." TSTF-545, Revision 3, proposes replacement of existing uses of the term "Inservice Testing Program" with the defined term, as denoted by capital letters throughout the TSs.

The NRC approved TSTF-545, Revision 3, by letter dated December 11, 2015 (ADAMS Package Accession No. ML15317A071) and published a notice of availability in the *Federal Register* on March 28, 2016 (81 FR 17208).

2.2 Proposed Technical Specifications Changes

The licensee requested to delete TS 5.5.6 from the Administrative Controls section of the TSs and replace it with the word "Deleted." TS 5.5.6 currently states:

> This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.
>
> a. Testing Frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for Inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

SR 3.0.2 allows an extension of IST intervals by up to 25 percent. If it is discovered that a surveillance associated with an IST activity was not performed within the required interval, SR 3.0.3 allows the licensee to delay declaring the associated limiting condition for operation (LCO) not met in order to perform the missed surveillance. The licensee did not request changes to SR 3.0.2 or SR 3.0.3.

The licensee requested to revise the Definitions section of the TSs by adding the term "INSERVICE TESTING PROGRAM" with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "Inservice Testing Program" in the TS SRs be replaced with "INSERVICE TESTING PROGRAM" so that the SRs refer to the new definition in lieu of the deleted program.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed changes:

Technical Specifications

Paragraph 50.36(c) of 10 CFR requires TSs to include the following categories: (1) safety limits, limiting safety systems settings, and control settings; (2) LCOs; (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Section 50.36(c)(3) of 10 CFR states that "[s]urveillance requirements 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 limiting conditions for operation will be met." Section 50.36(c)(5) of 10 CFR states that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

The NRC staff's guidance for review of the TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 16, "Technical Specifications," Revision 3, March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the staff has prepared improved STS for each of the LWR nuclear steam supply systems and associated balance-of-plant equipment systems. The licensee's proposed amendments are based on TSTF-545, Revision 3, which is an NRC-approved change to the improved STS. The NRC staff's review includes consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. Special attention is given to TS provisions that depart from the improved STS, as modified by NRC-approved TSTF travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met. In addition, the guidance states that comparing the change to previous STS can help clarify the intent of the TSs.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include IST of pumps and valves at nuclear power reactors in accordance with the ASME OM Code as specified in 10 CFR 50.55a(f). The regulations in 10 CFR 50.55a(f) state, in part:

> Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV [Boiler and Pressure Vessel] Code and ASME Code for Operation and Maintenance of Nuclear Power Plants as specified in this paragraph. Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions [referring to 10 CFR 50.55a(f)(1) through (f)(6)].

The ASME OM Code is a consensus standard that is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code IST program requirements were suitable for incorporation into the NRC's rules.

The regulation in 10 CFR 50.55(a)(f)(5)(ii) states, in part: "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program."

NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints

(Snubbers) at Nuclear Power Plants," Final Report, October 2013 (ADAMS Accession No. ML13295A020) provides guidance for the IST of pumps and valves.

NUREG-0800, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, March 2007 (ADAMS Accession No. ML070720041), provides guidance and acceptance criteria for the NRC staff review of the IST program for pumps and valves.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and licensing information discussed in Section 2.3 of this safety evaluation. 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. Among the considerations are whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5) (i.e., provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner). In making its determination as to whether to amend the licenses, the NRC staff considered those regulatory requirements that are automatically conditions of the license through 10 CFR 50.54. Where the regulations already condition the license, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Deletion of the Inservice Testing Program from the TSs

TS 5.5.6 requires the licensee to have an IST program that provides controls for IST of ASME Code Class 1, 2, and 3 components (i.e., pumps and valves). Through 10 CFR 50.54, the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include 10 CFR 50.55a(f), which specifies the requirements for the IST of pumps and valves. Therefore, requiring the licensee to have an IST program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an IST program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained below, it is not necessary to have additional administrative controls in the TSs relating to the IST program to assure operation of the facility in a safe manner.

Consideration of TS 5.5.6.a

The ASME OM Code requires testing to normally be performed within certain time periods. TS 5.5.6.a sets IST frequencies more precisely than those specified in the ASME OM Code and applicable addenda (e.g., "at least once per 31 days" contrasted with "monthly"). However, the NRC staff determined that the more precise IST frequencies are not necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of TS 5.5.6.a is acceptable.

Consideration of TS 5.5.6.b

TS 5.5.6.b allows the licensee to extend, by up to 25 percent, the interval between IST activities, as required by TS 5.5.6.a, and for other normal and accelerated frequencies specified as 2 years or less in the IST program. Similar to TS 5.5.6.b, the NRC authorization of ASME Code Case OMN-20 by letter dated February 16, 2017, also permits the licensee to extend the IST intervals specified in the ASME OM Code by up to 25 percent.

The NRC staff determined that the TS 5.5.6.b allowance to extend IST intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of TS 5.5.6.b is acceptable. The deletion of TS 5.5.6.b does not impact the licensee's ability to extend IST intervals using Code Case OMN-20, as authorized by the NRC.

Consideration of TS 5.5.6.c

TS 5.5.6.c allows the licensee to use SR 3.0.3 when it discovers that an SR associated with an inservice test was not performed within its specified frequency. SR 3.0.3 allows the licensee to delay declaring an LCO not met in order to perform the missed surveillance. The use of SR 3.0.3 for inservice tests is limited to those inservice tests required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. Deletion of TS 5.5.6.c does not change any of these requirements, and SR 3.0.3 will continue to apply to those inservice tests required by SRs. Based on the above, the NRC staff determined that deletion of TS 5.5.6.c is acceptable.

Consideration of TS 5.5.6.d

TS 5.5.6.d states that nothing in the ASME OM Code shall be construed to supersede the requirements of any TS. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised IST program for a facility conflicts with the TSs for the facility. The regulations require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the IST program because the regulations specify how conflicts must be resolved.

Conclusion Regarding Deletion of TS 5.5.6

The NRC staff determined that the requirements currently in TS 5.5.6 are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the staff concludes that deletion of TS 5.5.6 from the licensee's TSs is acceptable because TS 5.5.6 is not required by 10 CFR 50.36(c)(5).

3.2 <u>Definition of INSERVICE TESTING PROGRAM and Revision to SRs</u>

The licensee proposes to revise the TS Definitions section to include the term "INSERVICE TESTING PROGRAM" with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition of the INSERVICE TESTING PROGRAM is consistent with the definition in TSTF-545, Revision 3. The definition is acceptable to the NRC staff because it correctly refers to the IST requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "Inservice Testing Program" in SRs be revised to "INSERVICE TESTING PROGRAM" to reference the new TS-defined term in lieu of the deleted program. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in SRs with the new definition. The NRC staff verified that for each SR reference to the "Inservice Testing Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the SR testing is performed. However, the IST frequencies could change because the TSs will no longer include the more precise test frequencies in TS 5.5.6.a. As discussed in Section 3.1 of this SE, the staff determined that the TSs do not need to include the more precise testing frequencies currently in TS 5.5.6.a. Based on its review, the staff determined that revising the SRs to refer to the new definition is acceptable because these SRs will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The staff also determined that, with the proposed changes that allow less-precise testing frequencies, 10 CFR 50.36(c)(3) will continue to be met because the SRs will continue to ensure 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.

3.3 Deviations from TSTF-545

In its application, the licensee identified the following deviations from TSTF-545, Revision 3:

1. TSTF-545, Revision 3, completely deletes TS 5.5.6 from the TSs and renumbers the subsequent TS programs. The licensee proposes to delete the content of TS 5.5.6, but retains the TS number, and adds the word "Deleted." The licensee did not propose to renumber the subsequent TS programs.

2. Some of the numbering and wording for SRs that are modified does not match TSTF-545, Revision 3. However, the licensee stated that the SRs are equivalent.

The NRC staff finds that the proposed deviations are editorial in nature and the licensee's proposed TS changes remain consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

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 June 22, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the FR on December 6, 2016 (81 FR 87967). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Caroline Tilton

Date: July 12, 2017

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING REQUEST TO ADOPT TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-545 (CAC NOS. MF8359 AND MF8360) DATED JULY 12, 2017

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