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

October 31, 2019

Mr. John A. Krakuszeski
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. 295 AND 323 TO REVISE TECHNICAL SPECIFICATION 3.3.8.1, "LOSS OF POWER (LOP) INSTRUMENTATION" (EPID L-2018-LLA-0281)

Dear Mr. Krakuszeski:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 295 and 323 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2, respectively. These amendments are in response to your license amendment request dated October 18, 2018, as supplemented by letter dated April 3, 2019.

The amendments revise Technical Specification 3.3.8.1, "Loss of Power (LOP) Instrumentation," with regard to Function 1.b (i.e., 4.16 kV [kilovolt] Emergency Bus Undervoltage (Loss of Voltage) – Time Delay) of Table 3.3.8.1-1, "Loss of Power Instrumentation." Specifically, the change modifies the allowable values for the time delay associated with the loss of voltage relays to resolve a design vulnerability impacting the emergency diesel generator output breaker logic.

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,

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. 295 to DPR-71
2. Amendment No. 323 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. 295
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 October 18, 2018, as supplemented by letter dated April 3, 2019, 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. 295, 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 prior to the end of the 2023 Unit 2 refueling outage.

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: October 31, 2019

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 295</u>

<u>BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1</u>

<u>RENEWED FACILITY OPERATING LICENSE NO. DPR-71</u>

<u>DOCKET NO. 50-325</u>

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.

<u>Remove Page</u>	<u>Insert Page</u>
3.3-71	3.3-71

(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. 295, 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

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)			
a. Bus Undervoltage	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3115 V and ≤ 3400 V
b. Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 1.35 seconds and ≤ 3.0 seconds
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3706 V and ≤ 3748 V
b. Time Delay	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 9.0 seconds and ≤ 11.0 seconds

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 323
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 October 18, 2018, as supplemented by letter dated April 3, 2019, 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. 323, 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 prior to the end of the 2023 Unit 2 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop

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: October 31, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 323

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
3.3-71	3.3-71

(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. 323, 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,

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)			
a. Bus Undervoltage	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3115 V and ≤ 3400 V
b. Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 1.35 seconds and ≤ 3.0 seconds
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3706 V and ≤ 3748 V
b. Time Delay	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 9.0 seconds and ≤ 11.0 seconds

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 295 AND 323

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 October 18, 2018 (Reference 1), as supplemented by letter dated April 3, 2019 (Reference 2), Duke Energy Progress, LLC (Duke Energy or the licensee) requested changes to the Technical Specifications (TSs) for Brunswick Steam Electric Plant (Brunswick or BSEP), Units 1 and 2.

The license amendment request (LAR) proposed to revise TS 3.3.8.1, "Loss of Power (LOP) Instrumentation," for Function 1.b (i.e., 4.16 kV [kilovolt] Emergency Bus Undervoltage (Loss of Voltage) – Time Delay) of Table 3.3.8.1-1, "Loss of Power Instrumentation." Specifically, the proposed changes would modify the allowable values for the time delay associated with the loss of voltage (LOV) relays to resolve a design vulnerability impacting the emergency diesel generator (EDG) output breaker logic.

The supplement dated April 3, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 31, 2019 (84 FR 811).

2.0 REGULATORY EVALUATION

2.1 System Description

As described in Section 8, "Electric Power," of the Brunswick Updated Final Safety Analysis Report (UFSAR) (Reference 3), and Section 2, "Detailed Description," of the LAR, the Brunswick Class 1E Electrical Power Distribution System alternating current (AC) sources consist of the offsite power sources (i.e., preferred and alternate power sources), and the onsite standby power sources (i.e., EDGs 1, 2, 3, and 4). Each EDG is dedicated to its associated 4.16 kV emergency bus E1, E2, E3, or E4. An EDG starts automatically on a loss-of-coolant accident (LOCA) signal from either Unit 1 or Unit 2 or on undervoltage (LOV or degraded voltage (DV) conditions on the emergency bus. After the EDG has started, it automatically ties to its respective bus after offsite power is tripped because of emergency bus LOV or DV, independent of or coincident with a LOCA signal. Following the trip of offsite power, all loads are stripped from the 4.16 kV emergency bus except the 480 volt (V) emergency bus. When the EDG is tied to the emergency bus, select safety-related loads are then sequentially connected to their respective emergency bus by individual timers associated with each auto-connected load, following a permissive from a voltage relay monitoring each emergency bus.

An LOV on a 4.16 kV emergency bus indicates a loss of offsite power (LOOP) to the emergency bus. During an LOV condition on an emergency bus, the 27/59E LOV relay strips the incoming line breakers and the 4 kV motor load breakers from the emergency bus to provide a permissive for the EDG output breaker to close to allow the EDG to connect to the emergency bus.

Currently, during testing of an EDG, when an EDG is running at rated voltage and frequency and in manual mode (i.e., either from the control room or locally), there is a potential that the EDG output breaker logic scheme could result in a failure of EDG output breakers to remain closed to the emergency bus following an LOV signal. The condition is associated with the timing between the 27/59E LOV relay and the RCR-X relay. If an EDG automatic start signal is generated while the EDG is being tested (i.e., in manual mode), the RCR-X relay will send a trip signal to the EDG output breaker. This will allow the EDG to revert to automatic mode, and the EDG output breaker will then automatically reclose onto the emergency bus if the following closure permissives are met: (1) the EDG is at rated voltage and frequency, (2) an undervoltage condition is sensed on the emergency bus and an emergency bus breaker (i.e., CL-A or CL-B) is opened, and (3) the load sequencing relays are reset.

The 27/59E operates the 27EX auxiliary relay, which trips the emergency bus incoming line and 4 kV motor load breakers. Auxiliary contacts on these breakers operate the CL-A and CL-B relays. With the current TS allowable value (i.e., ≥ 0.5 seconds) for the time delay of the 27/59E LOV relay, it is possible that the EDG output breaker close permissive would be made prior to the RCR-X relay operation. If the CL-A or CL-B relay operates before the RCR-X relay, the EDG output breaker would close and then trip, once the RCR-X relay would operate. After the RCR-X relay would deenergize, the EDG output breaker anti-pump circuit would energize since the closing springs would still be charging. This would prevent the EDG output breaker from closing until the close signal would be removed.

To resolve the above concern with the design vulnerability impacting the EDG output breaker logic, the licensee proposed to revise the TS allowable values for the time delay of the 27/59E LOV relays so that upon an LOV condition on the 4.16 kV emergency bus, the CL-A/CL-B relay logic does not operate before the RCR-X relay changes state and cause the EDG output breaker lockout.

2.2 Proposed TSs Changes

The allowable value associated with Table 3.3.8.1-1, Function 1.b (i.e., 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) – Time Delay) would be revised from ≥ 0.5 seconds and ≤ 2.0 seconds to ≥ 1.35 seconds and ≤ 3.0 seconds.

In Section 3, "Technical Evaluation," of the LAR, the licensee stated that the proposed new allowable values for the time delay of the 27/59E LOV relay are based on nominal trip setpoints (NTSPs), which were conservatively chosen to account for total device uncertainties and include margin from the lower design limit (LDL) and upper design limit (UDL). The proposed lower allowable value (i.e., 1.35 seconds) and upper allowable value (i.e., 3.0 seconds) considered both the respective LDL, UDL, and the as-found tolerance for the time delay function. Adequate margin was built into the lower and upper allowable values to ensure margin is maintained from their respective LDL and UDL, and the time delay nominal trip setpoints will operate within the lower and upper allowable values between calibration.

2.3 Regulatory Requirements and Guidance

The regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. More specifically, 10 CFR 50.36(c)(3) states, in part, that SRs 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."

The Brunswick design was reviewed prior to construction under the General Design Criteria (GDC) for Nuclear Power Plant Construction issued for comment by the Atomic Energy Commission in July 1967. Brunswick is committed to meet the intent of the GDC published in the *Federal Register* on May 21, 1971, as Appendix A to 10 CFR Part 50. The following GDC are applicable to this LAR.

GDC 17, "Electric Power Systems," requires, in part, that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of Proposed Time Delay Changes

In Section 3, "Technical Evaluation," of the LAR, the licensee stated that the proposed new allowable values (≥ 1.35 seconds and ≤ 3.0 seconds) for the 27/59E LOV relay time delay were established in Brunswick Calculation 04kV-0001, "4.16 kV Emergency Bus (Loss of Voltage) Undervoltage and Time Delay Uncertainty and Setpoint Calculation (for 4kV 1-E1(E2)-AE7(AG5)-27/59E and 2-E3(E4)-AI3(AK0)-27/59E)," Revision 2, which was provided in Attachment 5 of the LAR. The calculation determined the allowable values for the time delay based on the UDL, LDL, uncertainties, and margin.

Calculation 04kV-0001 selected the LDL for the 27/59E LOV relay's time delay so that following an LOV condition on the 4.16 kV emergency bus, at the time the EDG is running unloaded in manual mode, the CL-A/CL-B relay logic may not operate before the RCR-X relay changes state and cause the EDG output breaker lockout. According to the calculation, the 27/59E LOV relay must not operate before 1 second to ensure the CL-A/CL-B relays do not change state before the RCR-X relay changes state. The calculation conservatively used 1.2 seconds as the LDL for the time delay of the 27/59E LOV relay. The NRC staff finds the 1.2-second LDL for the time delay acceptable since the 1.2-second LDL for the time delay resolves the design vulnerability of the EDG output breaker logic.

Calculation 04kV-0001 determined that the UDL for the 4.16 kV emergency bus 27/59E LOV time delay must be less than the time assumed for the EDG to restore the bus voltage and frequency. The time assumed for the 4.16 kV emergency bus restoration is 13 seconds, which includes the EDG start time (10 seconds) plus a 3-second delay for instrument lag. The calculation used 3.5 seconds as the UDL.

The Brunswick UFSAR Section 8.3.1.1.6.5.1, "Automatic Starting," states in part,

> The diesels are capable of achieving rated speed and voltage within ten [10] seconds of receipt of a start signal. The analytical limit for restoring bus voltage and frequency is based on supporting accident analyses and design basis calculations. The 13 seconds for bus restoration assumed in the Reactor Building Environmental Report [RBER] [...] is more limiting than the 15 seconds assumed in the LOCA analyses [....].

The NRC staff noted that the 13 seconds for bus restoration would be exceeded if the EDG start time (i.e., time to start and achieve rated speed and voltage) is 10 seconds, and the LOV relay's time delay to initiate the EDG start signal is 3.5 seconds, as assumed in the calculation 04kV-0001. The NRC staff requested the licensee to discuss the purpose of the 13 seconds for bus restoration assumed in the RBER and the impact of the 3.5-second time delay for the 27/59E LOV relay on the bus restoration times assumed in the RBER (13 seconds) and the LOCA analyses (15 seconds).

In its response letter dated April 3, 2019, the licensee clarified that the 13 seconds for bus restoration assumed in the RBER is the time for the EDG to reenergize the 4.16 kV emergency bus after a LOOP event (assumed for a high energy line break (HELB) scenario) to allow the closure of isolation valves in response to the HELB. The 13 seconds is the sum of the 10-second EDG start time and 3 seconds for EDG output breaker closure, which occurs after the EDG reaches rated speed and voltage. The licensee also explained that the 27/59E LOV relay does not initiate the EDG start signal, but it strips the emergency bus loads in parallel with, not in sequence with, the EDG start signal. This stripping function of the 27/59E LOV relay must be completed before the EDG reaches rated speed and voltage (i.e., between 3.5 seconds time delay for the 27/59E LOV relay and 10 seconds for the EDG start sequence) to avoid delaying the EDG output breaker closure. This case assumed that the EDG is not initially running at the time of the LOOP or LOOP/LOCA event. If the EDG is initially running in manual or test mode at the time of the LOOP or LOOP/LOCA event, the 27/59E LOV relay time delay would determine the EDG output breaker closure. For this case, the EDG is already running and does not require the 10-second start time. Hence, this case is bounded by the case when the EDG is in standby mode.

The NRC staff reviewed the licensee's response and finds that the 3.5-second UDL for the 27/59E LOV relay's time delay will not impact the 4.16 kV emergency bus restoration times assumed in the RBER (13 seconds) and the LOCA analyses (15 seconds) since the 3.5-second time delay occurs in parallel with the EDG start sequence time (10 seconds) assumed in the RBER and the LOCA analyses.

The NRC staff requested the licensee to discuss the impacts of the 3.5-second time delay for the 27/59E LOV relay on the existing licensing basis for the response times of equipment (pumps and valves) as assumed in accident analyses, sequenced after the 4.16 kV bus is reenergized. In its April 3, 2019, letter, the licensee clarified that after the emergency bus is reenergized, the automatic emergency bus load sequence is initiated by relays, which are independent from the 27/59E LOV relay. In addition, the licensee stated that the 27/LOV relay is reset after the EDG breaker closes, and failure of the 27/59E LOV relay to reset cannot prevent automatic loading of the emergency bus, nor can it trip the loads that are sequenced on the associated EDG during a LOOP or LOCA event. Therefore, the timing of the load sequence for the various pumps and valves is not affected by the additional time required for the 27/59E LOV relay to reset.

The NRC staff reviewed the licensee's response and finds that the 3.5-second UDL time delay for the 27/59E LOV relay will not impact the existing equipment (pump and valves) response times assumed in the accident analyses after the 4.16 kV bus is reenergized, since the function of the 27/59E LOV relay does not impact the load sequencing after EDG output breaker closure.

3.2 Evaluation of Calculation and Methodology

The NRC staff reviewed the methodology, setpoint change, and allowable value changes for acceptance. Calculation 04kV-0001 references Regulatory Guide (RG) 1.105, Revision 2, "Instrument Setpoints for Safety-Related Systems," dated February 1986 (Reference 4); International Society of Automation (ISA) Standard S67-04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation"; and ISA Standard S67-04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Calculation 04kV-0001 also listed procedure EGR-NGGC-0153, "Engineering Instrument Setpoint," Revision 12, as one of the references; however, this procedure was not provided. To determine how this procedure met the intent of RG 1.105, the licensee provided the following clarification with respect to RG 1.105 in its April 3, 2019, letter:

> Calculation 04KV-0001 is prepared in accordance with Duke Energy procedure EGR-NGGC-0153 *Engineering Instrument Setpoints*. EGR-NGGC-0153 is based upon ISA Standard S67-04.01-2000, *Setpoints for Nuclear Safety-Related Instrumentation*; and ISA Standard S604-02-2000 [S67-04.02-2000], *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation*. BSEP is not committed to Regulatory Guide (RG) 1.105; however, RG 1.105 was consulted to ensure the calculated instrument setpoints are, and will remain, within their technical specification limits. The as-found tolerance band established in Calculation 04KV-0001 accounts for instrument setting tolerance, drift, and calibration instrument uncertainty. This method is consistent with NRC guidance provided in RIS [Regulatory Issue Summary] 2006-17.

Calculation 04kV-0001 also addressed the errors and the setpoint for the 27/59E LOV relay time delay. Section 5.2.2 of calculation 04kV-0001 documented the errors for the time delay function for instrument uncertainties, including reference accuracy, drift, temperature effects etc. The calculation used the square root of the sum of the squares method for combining the random errors. In addition, the bias errors were added algebraically.

During the review of the drift errors in Section 5.2.2.2 of the calculation, the NRC staff noted that the drift error calculation is based on the current 1-year calibration interval. Even though the

NRC approved the 24-month surveillance frequencies to support a 24-month fuel cycle as part of the Brunswick improved Standard Technical Specification amendments issued on June 5, 1998 (Reference 5), the licensee confirmed in its April 3, 2019, letter that the current calibration interval of 1 year remains unchanged.

The setpoint for the 27/59E LOV relay time delay is calculated based on the UDL, LDL, and the errors. Margins have been added for conservatism. A summary of the setpoint is presented in Section 7.2 of calculation 04kV-0001 as noted in Table 1 below.

Table 1: Emergency Bus Over / Undervoltage Relay – Time Delay
(Reference Section 7.2.2 of Calculation 04kV-0001 in Attachment 5 of the LAR)

Parameter	Value	Equation
Upper Design Limit (UDL)	3.50 seconds	N/A
Margin[1] (M^1)	1.20 seconds	N/A
Margin[2] (M^2)	0.04 seconds	N/A
Margin[3] (M^3)	0.39 seconds	N/A
Total Loop Uncertainty (TLU)	0.43 seconds	N/A
Other Uncertainties (OU)	0.11 seconds	OU = TLU - LAFT
Loop As-Found Tolerance (LAFT)	0.32 seconds	LAFT = AFT
Loop As-Left Tolerance (LALT)	0.28 seconds	LALT = ALT
Upper Allowable Value (UAV)	≤ 3.00 seconds	UAV = UDL - OU - M^3
Setpoint (SP)	1.87 seconds	SP = UDL - TLU - M^1
Lower Allowable Value (LAV)	≥ 1.35 seconds	LAV = LDL + OU + M^2
Reset Value	N/A	N/A
Lower Design Limit (LDL)	≥ 1.20 seconds	N/A

Based on its review of the calculation 04kV-0001, the NRC staff finds the calculation for the setpoint for the time delay acceptable because: (1) different errors for the time delay have been appropriately combined per the guidance of RG 1.105, Revision 2, (2) the setpoints have been selected based on consideration of all applicable errors with margins as needed, and (3) the determination of the as-left tolerances and as-found tolerances is consistent with the staff position in RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels" (Reference 6).

3.3 Technical Evaluation Conclusion

The NRC staff reviewed the licensee's proposed changes to Brunswick TS Table 3.3.8.1-1, Function 1.b, associated with 4.16 kV Emergency Bus LOV time delay allowable values. Based on the above technical evaluation, the NRC staff concludes that the proposed TS changes provide reasonable assurance of the availability of required safety equipment needed to shut down the reactor and keep it in a safe condition following an LOV condition or an accident. Furthermore, the staff concludes that the proposed TS change does not impact the licensee's continued compliance with 10 CFR 50.36(c) and GDC 17 for availability of offsite or onsite power systems in a timely manner, as assumed in the accident analyses. Therefore, the NRC staff finds the proposed changes in the LAR 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 September 24, 2019. 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, published in the *Federal Register* on January 31, 2019 (84 FR 811), and there has been no public comment on such finding. 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, Request for License Amendment - Technical Specification 3.3.8.1, 'Loss of Power (LOP) Instrumentation,'" dated October 18, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18291A628).

2. Gideon, William R., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Response to Request for Additional Information - Request for License Amendment - Technical Specification 3.3.8.1, 'Loss of Power (LOP) Instrumentation,'" dated April 3, 2019 (ADAMS Accession No. ML19093B177).

3. Gideon, William R., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Updated Final Safety Analysis Report, Revision 26," dated August 13, 2018 (ADAMS Accession Package No. ML18249A165).

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.105, Revision 2, "Instrument Setpoints for Safety-Related Systems," dated February 1986 (ADAMS Accession No. ML003740318).

5. Trimble, David C., U.S. Nuclear Regulatory Commission, letter to Hinnabt, C. S., Carolina Power & Light Company, "Issuance of Amendment No. 203 to Facility Operating License No. DPR-71 and Amendment No. 233 to Facility Operating License No. DPR-62 Regarding Conversion to Improved Standard Technical Specifications and Implementation of Reactor Stability Solution - Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. M97243 and M97244)," dated June 5, 1998 (ADAMS Accession No. ML12047A393).

6. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (ADAMS Accession No. ML051810077).

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Date: October 31, 2019

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 295 AND 323 TO REVISE TECHNICAL SPECIFICATION 3.3.8.1, "LOSS OF POWER (LOP) INSTRUMENTATION" (EPID L-2018-LLA-0281) DATED OCTOBER 31, 2019

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