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

April 22, 2019

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 289 AND 317 TO RELOCATE THE PRESSURE-TEMPERATURE LIMITS IN THE TECHNICAL SPECIFICATIONS TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2018-LLA-0094)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 289 and 317 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (Brunswick), Units 1 and 2, respectively. These amendments are in response to your license amendment request dated April 4, 2018, as supplemented by letters dated May 29, 2018; September 27, 2018; and December 11, 2018.

The amendments revise the Brunswick Technical Specifications (TSs) as necessary to relocate the reactor pressure vessel pressure-temperature limits to a licensee-controlled Pressure and Temperature Limits Report. Specifically, the amendments modify TS 1.1, "Definitions," and TS Section 3.4.9, "RCS [Reactor Coolant System] (P/T) Limits," to delete reference to the pressure-temperature curves, and the amendments add TS 5.6.7, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to include reference to the unit-specific Pressure and Temperature Limits Reports. The request also implements new pressure-temperature limits for both Brunswick, Units 1 and 2.

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

Sincerely,

Dennis J. Galvin

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

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 289 to DPR-71
2. Amendment No. 317 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. 289
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 April 4, 2018, as supplemented by letters dated May 29, 2018; September 27, 2018; and December 11, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

Enclosure 1

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. 289, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: April 22, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 289

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

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

Replace the following 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	Insert
1.1-6	1.1-6
3.4-19	3.4-19
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	3.4-23
3.4-24	----
3.4-25	----
3.4-26	----
3.4-27	----
3.4-28	----
5.0-22	5.0-22
----	5.0-22a

(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. 289, 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 (continued)

OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.8.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2923 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve 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.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE---------------- Required Action A.2 shall be completed if this Condition is entered. ------------------------------------ Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------------NOTE--------------- Required Action C.2 shall be completed if this Condition is entered. -------------------------------------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----------------------------NOTE----------------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	 Once within 30 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.4	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. --- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 30 minutes prior to each startup of a recirculation pump
SR 3.4.9.5	------------------------------NOTE------------------------------ Only required to be performed when tensioning the reactor vessel head bolting studs. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.6	------------------------------NOTE------------------------------ Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.7	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1045 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1045 psig.	In accordance with the Surveillance Frequency Control Program

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

21. ANP-10298P-A, ACE/ATRIUM 10XM Critical Power Correlation, Revision 1, March 2014.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

(continued)

5.6 Reporting Requirements (continued)

5.6.8 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits,"

   2. Surveillance Requirement Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

   1. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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. 317
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 April 4, 2018, as supplemented by letters dated May 29, 2018; September 27, 2018; and December 11, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 317, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: April 22, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 317

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 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	Insert
1.1-6	1.1-6
3.4-19	3.4-19
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	3.4-23
3.4-24	----
3.4-25	----
3.4-26	----
3.4-27	----
3.4-28	----
5.0-22	5.0-22
----	5.0-22a

(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. 317, 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 (continued)

OPERABLE-OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.8.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2923 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve 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.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE-------------- Required Action A.2 shall be completed if this Condition is entered. ------------------------------------ Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. --------------NOTE--------------- Required Action C.2 shall be completed if this Condition is entered. ------------------------------------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	-------------------------------NOTE------------------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 30 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.4	-----NOTE----- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 30 minutes prior to each startup of a recirculation pump
SR 3.4.9.5	-----NOTE----- Only required to be performed when tensioning the reactor vessel head bolting studs. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.6	-----NOTE----- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.7	-----NOTE----- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1045 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1045 psig.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

21. ANP-10298P-A, ACE/ATRIUM 10XM Critical Power Correlation, Revision 1, March 2014.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

(continued)

5.6 Reporting Requirements (continued)

5.6.8 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits,"

   2. Surveillance Requirement Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

   1. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 289 AND 317

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 April 4, 2018 (Reference 1), as supplemented by letters dated May 29, 2018 (Reference 2); September 27, 2018 (Reference 3); and December 11, 2018 (Reference 4), Duke Energy Progress, LLC (the licensee) requested changes to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (Brunswick or BSEP), Units 1 and 2.

The amendments would revise the Brunswick, Units 1 and 2, TSs, as necessary, to relocate the reactor pressure vessel pressure-temperature (PT, P-T, or P/T) limits to a licensee-controlled Pressure and Temperature Limits Report (PTLR). Specifically, the amendments would modify TS 1.1, "Definitions," and TS 3.4.9, "RCS [Reactor Coolant System] (P/T) Limits," to delete reference to the P-T curves, and the amendments would add TS 5.6.7, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to include reference to the unit-specific PTLRs. The licensee stated that the license amendment request (LAR) was prepared in accordance with the guidelines of U.S. Nuclear Regulatory Commission (NRC or the Commission) Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 5), and Technical Specifications Task Force (TSTF) Traveler TSTF-419, Revision 0, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (References 6), as approved by the NRC staff (Reference 7). The NRC staff clarified the use of TSTF-419 by letter dated August 4, 2011 (Reference 8).

Additionally, the licensee proposed to replace the current P-T limits for 32 effective full power years (EFPY), approved by Amendment Nos. 228 (Unit 1) and 256 (Unit 2) issued on June 18, 2003 (Reference 9), with proposed P-T limits based on the Boiling Water Reactor Owners' Group (BWROG) Topical Report, BWROG-TP-11-022-A, "Pressure Temperature Limits Report Methodology for Boiling Water Reactors" (Reference 10) (the BWROG report). The BWROG report contains the associated NRC safety evaluation (SE) (Reference 11). The proposed P-T limits in this request that would be located in the PTLR are for operation to 54 EFPY.

The supplements dated September 27, 2018 (Reference 3), and December 11, 2018 (Reference 4), provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 17, 2018 (83 FR 33266).

2.0 REGULATORY EVALUATION

2.1 System Description

All components of the RCS are designed to withstand the effects of cyclic loads resulting from system P-T changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. The reactor pressure vessel (RPV) contains the reactor core and all associated support and alignment devices. The RPV acts as part of the reactor coolant pressure boundary (RCPB), which is the second barrier to the release of fission products to the environment.

In accordance with Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, the TSs limit the P-T changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation. These limits are defined by P-T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. Therefore, acceptable operation of the RCS is defined by maintaining RCS pressure less that the P-T limits and RCS temperature greater than the P-T limits for all modes of reactor operation when the RPV closure head is tensioned to the vessel.

2.2 Proposed TSs Change

The licensee proposed the following changes in its LAR:

1. Add a definition in TS 1.1 for the Pressure and Temperature Limits Report. The wording of this definition is consistent with NUREG-1433, Revision 4, Volume 1, "Standard Technical Specifications – General Electric BWR/4 Plants: Specifications" (STS) (Reference 12).

2. Revise TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits," to refer to the PTLR.

3. Combine existing SR 3.4.9.1 and existing SR 3.4.9.2. The new SR 3.4.9.1 verifies that RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.

4. Renumber existing SR 3.4.9.3 through SR 3.4.9.8 as SR 3.4.9.2 through SR 3.4.9.7. Revise these SRs to reference the PTLR for specified limits.

5. Remove the present P-T curves, Figures 3.4.9-1, 3.4.9-2, 3.4.9-3, 3.4.9-4, and 3.4.9-5 from the TSs.

6. Add a new TS 5.6.7, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," in TS 5.0, "Administrative Controls."

The new specification is consistent in format and content with NUREG-1433 STS (Reference 12) and:

- Includes the individual TSs that address RCS P-T limits,
- References the NRC-approved topical report, which documents the PTLR methodology, and
- Requires the PTLR, and any revisions or supplements, to be submitted to the NRC.

The licensee also submitted TS Bases changes that corresponded to the proposed TS changes.

2.3 Regulatory Requirements and Guidance Applicable to P-T Limits

The "General Design Criteria for Nuclear Power Plants" listed in 10 CFR Part 50, Appendix A, as amended July 7, 1971, were used as the basis for an audit of the design features of Brunswick. The following criteria from the Brunswick Updated Final Safety Analysis Report (UFSAR) (Reference 13) are related to this LAR:[1]

- Criterion 14 - Reactor Coolant Pressure Boundary. The RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- Criterion 30 - Quality of Reactor Coolant Pressure Boundary. Components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

- Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary. The RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions:

  a. The boundary behaves in a non-brittle manner and

  b. The probability of rapidly propagating fracture is minimized.

  The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications," and require, in part, that a summary statement of the bases for such specifications shall be included by applicants for a license authorizing operation of a production or utilization facility. Specifically, the requirements for TS content in

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[1] While Section 3.1.1 of the Brunswick UFSAR indicates that the GDC, as amended July 7, 1971, were used as the basis for an audit of the design features of Brunswick, the criteria identified in Section 3.1.2 of the Brunswick UFSAR are based on the GDC published in the *Federal Register* on February 20, 1971. The July 7, 1971, amendments did not change the GDC considered in this SE, and thus, the identified inconsistency has no impact on the NRC staff's analysis for this LAR.

10 CFR 50.36(c) include the following categories related to facility operation: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

The regulation in 10 CR 50.36(c)(2), "Limiting conditions for operation," states that "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

The regulation in 10 CFR 50.36(c)(3), "Surveillance requirements," states that "[s]urveillance requirements are requirements related 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 regulation in 10 CFR 50.36(c)(5), "Administrative controls," 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."

Section 50.60 of 10 CFR requires that all light-water nuclear power reactors meet the fracture toughness and material surveillance program requirements set forth in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," in order to protect the integrity of the RCPB.

Appendix G to 10 CFR Part 50 requires, in part, that the P-T limits for an operating light-water nuclear power reactor be at least as conservative as the limits obtained by following the methods of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code) Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure." The provisions of 10 CFR Part 50, Appendix G, also require, in part, that applicable surveillance data from RPV material surveillance programs are developed in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and the effects of neutron irradiation on the material properties must be accounted for in the calculations of plant-specific P-T limits for reactor vessel beltline materials. Finally, Table 1 of Appendix G to 10 CFR Part 50 provides the P-T limits and minimum temperature requirements for the RPV during normal heatup, cooldown, and pressure test operations.

NRC Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 14), describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement on the low-alloy steels used for light-water RPVs.

NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 15), describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence with respect to meeting the regulatory requirements in 10 CFR 50.60 and Appendix G to 10 CFR Part 50.

NRC GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" (Reference 16), requested that licensees submit their plant-specific RPV data to the NRC staff for review.

NRC GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity" (Reference 17), requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 18), provides additional NRC staff expectations for evaluations of PT limits in licensing applications and PTLRs, including specific guidance on the consideration of neutron fluence and structural discontinuities in the development of PT limits.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock" (Reference 19), provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME B&PV Code, Section XI, Appendix G methodology.

2.4 Guidance Applicable to the Development of PTLRs

On January 31, 1996, the NRC staff issued GL 96-03 (Reference 5) to inform licensees that they may request a license amendment to relocate the P-T limits from the TS LCOs to a PTLR or other licensee-controlled document with reporting requirements governed by a new TS in the administrative controls section of the TSs. In order to permit relocation of the P-T limits to a PTLR, GL 96-03 states that licensees shall generate their P-T limits in accordance with an NRC-approved methodology and that the methodology used to generate the P-T limits shall comply with the requirements of 10 CFR Part 50, Appendices G and H. GL 96-03 also states that the methodology used to generate the P-T limits must be incorporated by reference into the administrative controls section of the TSs and that the PTLR must be defined in Section 1.0 of the TSs. Attachment 1 to GL 96-03 provides a list of the seven criteria that the methodology and PTLR must meet for a plant-specific PTLR license amendment.

TSTF-419, Revision 0[2] (Reference 6), amended the STS for all domestic light-water reactor designs to: (1) delete references to the TS LCOs for the P-T limits and low-temperature overpressure protection system limits in the TS definition of the PTLR, and (2) revise the standard administrative controls for the PTLR in STS Section 5.6 to allow NRC-approved topical reports for PTLR methodologies to be identified by number and title. Supplemental guidance related to TSTF-419, Revision 0, was provided in an NRC letter dated August 4, 2011 (Reference 8), that required the full topical report or methodology citation to be included in the TSs, not in the PTLR. The supplemental guidance in TSTF-419 did not change the requirement that the PTLR methodology be approved by the NRC or the TS requirement to operate the RCS within the limits specified in the PTLR. Additionally, any changes to a PTLR methodology

[2] Throughout the LAR, the licensee refers to TSTF-419-A as a basis for the requested amendments to relocate the P-T limit curves to a PTLR. The NRC staff notes that the "-A" designation added to TSTF-419 is an industry convention used to indicate that the TSTF has been approved by the NRC. TSTF-419 and TSTF-419-A are the same document. However, since TSTF-419-A is not an NRC designation, this SE refers to the traveler as TSTF-419.

continue to require NRC staff review and approval pursuant to the provisions of 10 CFR 50.90. TSTF-419, Revision 0, has since been incorporated into NUREG-1433, Revision 4, Volume 1 (Reference 12).

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee's evaluation of the relocation of P-T limits to a PTLR is provided in the enclosure to the LAR. In Section 3.2 of the enclosure, the licensee addressed the seven criteria in Attachment 1 to GL 96-03 for P-T limits relocation under three headings: Neutron Fluence Calculations, BSEP Surveillance Capsule Results and Adjusted Reference Temperature, and Pressure-Temperature Curve Evaluation. Although not in a sequential order, all seven criteria were addressed. The licensee also stated that TSTF-419 was followed in development of the proposed TS changes.

The licensee's evaluation of the proposed P-T limits is contained in Attachment 6, "Pressure Temperature Limits Report (PTLR)," dated January 2018. The proposed P-T limits are based on application of the BWROG report methodology, an approved generic methodology for generating P-T limits based on the plant-specific adjusted reference temperatures (ARTs), which is consistent with NRC PTLR development guidance in GL 96-03 (Reference 5). The BWROG report methodology was implemented as documented in the proposed Brunswick PTLR.

The licensee proposed P-T limits for 54 EFPY. The proposed P-T limits consider new fluence values and the resulting new reference temperature shift (ΔRT_{NDT}) for the RPV materials. The proposed P-T limits were developed for three regions: beltline, bottom head, and non-beltline (upper vessel). The composite curves are the limiting segments of the three sets of P-T limits. The licensee used finite element analyses to develop the stress distributions for the beltline and non-beltline regions, and the instrument nozzles. For the bottom head region, the licensee used the closed-form stress solution permitted by the BWROG report. These stress distributions were used in determining the stress intensity factor due to pressure (K_{IP}), thermal (K_{IT}), or both. For the beltline region, the licensee identified Plate B8496-1 with an ART of 129.1 degrees Fahrenheit (°F) as the limiting material for Unit 1, and N16A/B instrument nozzles with an ART of 123.4.1 °F as the limiting material for Unit 2. For the bottom head region, the limiting RT_{NDT} remains the same at 10 °F (Heat No. C4654) for Unit 1 and 40 °F (Heat No. C4890) for Unit 2. For the non-beltline region, the limiting RT_{NDT} is 60 °F for both units. P-T limits were then developed for these regions using the K_{IP} and K_{IT}, and the material toughness of the limiting materials based on ART in accordance with ASME B&PV Code, Section XI, Appendix G, with details supplemented by the BWROG report methodology.

3.2 NRC Staff Evaluation

The NRC staff evaluated the Brunswick implementation of the PTLR against the seven methodological criteria in Attachment 1 to GL 96-03 for P-T limits relocation. The NRC staff evaluated the neutron fluence methodology, since a specific methodology is not included in the BWROG report methodology. The NRC staff evaluated the proposed P-T limits in the PTLR. Finally, the NRC staff evaluated the conformance of the proposed TS changes to TSTF-419.

3.2.1 PTLR Implementation

As stated in the LAR, the licensee utilized the BWROG report methodology to generate the P-T limits. The BWROG report methodology is an NRC-approved method for use in generating PTLRs (Reference 11). The licensee proposes to incorporate the BWROG report into the new TS 5.6.7 to reflect the above.

As noted in Section 3.1 of this SE, the licensee demonstrated the acceptability of its PTLR by evaluating the PTLR methodology against the seven methodological criteria in Attachment 1 to GL 96-03 for P-T limits relocation. The NRC staff examined the proposed PTLR and determined that it was developed appropriately based on the BWROG report methodology and meets the seven methodological criteria in Attachment 1 to GL 96-03 as described below.

(1) The PTLR methodology describes the transport calculation methods, including computer codes and formula used to calculate neutron fluences.

Section 3 of the PTLR indicates that the fluence was determined in accordance with RG 1.190, as documented in Westinghouse Report WCAP-17660-NP, Revision 0, "Neutron Exposure Evaluations for Core Shroud and Pressure Vessel Brunswick Units 1 and 2." As noted in Section 3.2.2 of this SE, this report was submitted by the licensee to the NRC by letter dated May 29, 2018 (Reference 2). The NRC staff's evaluation and acceptance of the fluence methodology and its predicted neutron fluence values for RPV materials are discussed fully in Section 3.2.2 of this SE. Therefore, the NRC staff concludes that Criterion 1 is met.

(2) The PTLR methodology describes the surveillance program.

Appendix A of the PTLR provides a description of the Brunswick reactor vessel materials surveillance program. Appendix A of the PTLR states that Brunswick has replaced the original RPV material surveillance program with the BWR Vessel and Internals Project Integrated Surveillance Program (BWRVIP ISP) and has made a licensing commitment to use the ISP during the period of extended operation.

The NRC staff reviewed Appendix A of the PTLR and confirmed that in an SE dated January 14, 2004 (Reference 20), the NRC approved Brunswick's replacement of the original plant-specific RPV surveillance program with the BWRVIP ISP. The BWRVIP ISP is described in BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan" (Reference 21), and BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan" (Reference 22 and Reference 23). The NRC approved the BWRVIP ISP by letter dated February 1, 2002 (Reference 24).

Appendix A of the PTLR states that one surveillance capsule has been removed and tested from each of the Brunswick reactor vessels, but that no further capsules are scheduled for removal. Instead, representative surveillance capsule material for Brunswick, Units 1 and 2, are contained in the River Bend Station and Duane Arnold Energy Center. Specifically, Appendix A of the PTLR states:

> Representative surveillance capsule materials for the BSEP Unit 1 and 2 limiting beltline plate are contained in the River Bend and Supplemental Surveillance Program (SSP) Capsules C, F, and H.

> Representative materials for the BSEP Unit 1 and 2 limiting beltline weld are in the Duane Arnold and SSP-F surveillance capsules.

By letter dated September 27, 2018 (Reference 3), the licensee clarified that the words "plate" and "weld" were incorrectly transposed in Appendix A of the PTLR, and the correct statement is:

> Representative surveillance capsule materials for the BSEP Unit 1 and 2 limiting beltline weld are contained in the River Bend and Supplemental Surveillance Program (SSP) Capsules C, F, and H. Representative materials for the BSEP Unit 1 and 2 limiting beltline plate are in the Duane Arnold and SSP-F surveillance capsules.

The NRC staff reviewed BWRVIP-86, Revision 1-A, "Updated BWR Integrated Surveillance Program (ISP) Implementation Plan" (Reference 25), as approved by the NRC by letter dated October 20, 2011 (Reference 26), and confirmed the information in Appendix A of the Brunswick PTLR that no further capsules are scheduled for removal from the Brunswick, Units 1 and 2, vessels. Additionally, the staff confirmed that the updates to Appendix A regarding the capsules hosting the representative materials for the Brunswick, Unit 1 and 2, limiting beltline materials is correct.

The NRC staff concludes that Criterion 2 is met because the description of the surveillance program in the PTLR contains the necessary and correct information for Brunswick's ISP program.

(3) The PTLR methodology describes how the low temperature overpressure protection (LTOP) system limits are calculated applying system/thermal hydraulics and fracture mechanics.

LTOP system limits are for pressurized water reactors only. Therefore, Criterion 3 does not apply to the Brunswick PTLR because Brunswick, Units 1 and 2, are BWR units.

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2.

Section 3 of the PTLR indicates that the ART values for the limiting beltline materials are calculated in accordance with RG 1.99, Revision 2 (Reference 14). The NRC staff reviewed the ART summaries in Tables 7 and 8 of the PTLR, independently verified the calculations, and confirmed that all ARTs for the beltline materials are calculated in accordance with RG 1.99, Revision 2. Therefore, the NRC staff concludes that Criterion 4 is met.

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on the ASME B&PV Code, Section XI, Appendix G, and SRP Section 5.3.2.

Section 3 of the PTLR states that the P-T limits are calculated in accordance with the BWROG report methodology, which is an approved P-T limit methodology based on ASME B&PV Code, Section XI, Appendix G, and SRP Section 5.3.2. The NRC staff reviewed the PTLR and confirmed that the fracture mechanics applied in the construction of the P-T limits in the PTLR is in accordance with ASME B&PV Code, Section XI, Appendix G. Therefore, the NRC staff concludes that Criterion 5 is met.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T curves.

Section 3 of the PTLR states that the P-T limits were calculated in accordance with the BWROG report methodology, which is an approved P-T limit methodology based on ASME B&PV Code, Section XI, Appendix G, and SRP Section 5.3.2. Section 4 of the PTLR further describes how the minimum temperature limits are set in accordance with Appendix G to 10 CFR Part 50. The NRC staff reviewed the PTLR and confirmed that the PTLR describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to the P-T curves. Therefore, the NRC staff concludes that Criterion 6 is met.

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

The PTLR indicates that the representative heat of the plate and weld material for both Brunswick, Units 1 and 2, in the ISP is not the same as the target plate and weld material in Brunswick, Units 1 or 2. Additionally, the PTLR states that the representative heat of the plate and weld material do not exist in Brunswick, Units 1 or 2, beltlines. Therefore, the PTLR notes that for all beltline materials, the chemistry factor values are calculated using table values from RG 1.99, Revision 2.

Based on a review of the PTLR, the NRC staff verified that the method described in the PTLR is consistent with the first part of the procedure described in BWRVIP-86, Revision 1-A, Section 5.6, "Data Utilization," Option 1. By letter dated September 27, 2018 (Reference 3), the licensee provided an evaluation of the surveillance data for the target limiting plate and weld of the Brunswick units to demonstrate consistency with the second part of Option 1. This evaluation confirmed that the measured Charpy ΔT_{30} shift is within the normally expected scatter in the predicted shift of RG 1.99, Revision 2. Based on a review of the supplemental information provided by the licensee, the NRC staff finds that the PTLR is consistent with the BWRVIP-86, Revision 1-A procedure in its entirety, and therefore concludes that Criterion 7 is met.

In summary, the NRC staff finds that all seven criteria of Attachment 1 to GL 96-03 are met, and that implementation of the Brunswick, Units 1 and 2, PTLR is acceptable.

3.2.2 Fluence Evaluation

As noted in Section 3.2.1 of this SE, the licensee demonstrated the acceptability of its PTLR by evaluating the PTLR methodology against the seven methodological criteria in Attachment 1 to GL 96-03 for P-T limits relocation. Since a specific neutron fluence methodology is not included in the BWROG report methodology, the NRC staff reviewed each of the three topics identified for Criterion 1, which include: (a) describe how the neutron fluence is calculated, (b) describe the transport calculation methods, including computer codes and formula used to calculate neutron fluence, and (c) provide the neutron fluence values that are used in the ART calculation.

Regarding the reactor vessel neutron fluence, Table 1-1 of the BWROG report states, "[f]luence methods and results must comply with RG 1.190, and have NRC approval for use with this [licensing topical report]." In LAR Section 3.2, "Technical Analysis," under the heading "Neutron Fluence Calculations," the licensee explains that the neutron fluence calculations were updated using an NRC-approved methodology in accordance with RG 1.190. Further, Section 3 of the PTLR indicates that the neutron fluence is calculated in accordance with RG 1.190, as documented in Westinghouse Report WCAP-17660-NP (Reference 2).

The licensee did not originally include WCAP-17660-NP in the LAR submittal. By letter dated May 15, 2018 (Reference 27), the NRC staff requested supplemental information from the licensee to address the first criterion in Attachment 1 to GL 96-03 to describe how the neutron fluence is calculated. By letter dated May 29, 2018 (Reference 2), the licensee submitted WCAP-17660-NP, Revision 0 to the NRC.

WCAP-17660-NP, Section 1.0, states that the neutron fluence calculational methodology used in the PTLR fluence analysis has been applied to the Brunswick reactors in the past and was previously reviewed and accepted by the NRC staff in Amendment Nos. 228 and 256 to Brunswick, Units 1 and 2, respectively, issued on June 18, 2003 (Reference 9). These amendments contain the NRC staff's evaluation of the Brunswick neutron fluence calculational methodology used to determine the fluence for the existing P-T limits. WCAP-17660-NP, Section 1.0, further states that the neutron fluence calculational methodology complies with RG 1.190.

The same neutron fluence calculational methodology was also used as part of the Brunswick license renewal in 2006. The NRC's SE report related to the Brunswick, Units 1 and 2, license renewal (Reference 28) confirmed that the methodology used to determine 54 EFPY fluence values conforms to the recommendations in RG 1.190.

WCAP-17660-NP, Section 3.0, states that the neutron fluence calculational methodology has been enhanced from the previously reviewed version, as follows:

> [S]everal enhancement[s] to the analytical model were included to better describe the [boiling water reactor] fuel and bypass coolant features for the outermost row of peripheral assemblies, which are the most influential to the neutron exposures at the core shroud and the reactor vessel. The results of this evaluation indicate that, in general, the projected neutron exposures of critical components are less than those reported in [the 2003 license amendment request].

By letter dated September 27, 2018 (Reference 3), the licensee clarified that the neutron fluence calculational methodology in WCAP-17660-NP is the same as the previously approved methodology. The licensee stated that the enhancements described in WCAP-17660-NP, Section 3.0, refer to modeling of the core geometry. In WCAP-17660-NP, the interior regions of the core and the peripheral regions of the core are modeled as separate regions, whereas previously the entire core was modeled as a single region. The NRC staff has confirmed that the underlying fluence calculational methodology is unchanged, except for certain geometrical modeling enhancements, including the use of a higher fidelity fluence calculation by increasing the anisotropic scattering order (P_5 versus P_3), and by using a finer angular mesh (S_{16} versus S_8). The geometrical modeling enhancements described by the licensee in the September 27, 2018, letter can be seen in WCAP-17660-NP, Figure 2.1-1. These enhancements are an acceptable change because they are expected to increase the accuracy of the fluence calculation based on a more accurate representation of the heterogeneities of the peripheral fuel assemblies that contribute the most to reactor pressure vessel fluence.

The NRC staff observed that fluence values were provided in WCAP-17660-NP (and the LAR) for the limiting N16 instrumentation nozzle that exists within the core active height; no other nozzles were identified within the beltline. The NRC staff notes that there is plant-specific validation data available (i.e., ex-vessel neutron dosimetry) spanning the active core height. This is documented in the responses to requests for additional information (RAIs) received as part of the 2003 fluence calculational method review and supports fluence method qualification specific to Brunswick (Reference 29).

The NRC staff also confirmed the following with respect to the fluence values reported in WCAP-17660-NP and summarized in the LAR:

- Fluence values are appropriately projected to 54 EFPY based on using actual operational data for all completed cycles through Cycle 18 (25 EFPY) for Unit 1 and Cycle 19 (24.5 EFPY) for Unit 2.[3]
- Cycle-specific fuel designs are used and cycle-specific EFPY is tracked.
- Fluence projections are based on an average of the most recent 4-5 cycles of fluence data (as of 2012) at nominal Maximum Extended Load Line Limit Analysis Plus (MELLLA+) conditions.
- The assumed power level is based on the highest approved power level of 2,923 megawatts thermal (MWt).
- The assumed capacity factor of 0.92 is greater than the recent historical average of 0.89-0.90 based on the last 6-7 cycles of data from WCAP-17660-NP, Table 2.1-1, "Brunswick Units 1 and 2 Operating History."
- The latest approved fuel design in use at Brunswick (i.e., ATRIUM 10XM) is used for the fluence projection.
- Neutron fluence will be reevaluated on an as-needed basis to include operating data and new fuel projections.[4]

Based on the above, the NRC staff finds that (1) the WCAP-17660-NP fluence calculational method inputs are representative of past operating conditions, (2) the licensee has demonstrated that current and future operating conditions will be appropriately accounted for and will result in updated fluence projections when necessary, and (3) use of the WCAP-17660-NP fluence calculational method is expected to continue to produce best-estimate fluence values within the 20 percent allowance for uncertainty at the 1-sigma level recommended in RG 1.190. This ensures that the margins provided for fluence in the temperature shift calculations

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[3] The NRC staff confirmed that the peak surface fluence and N16 nozzle peak fluence values reported in the LAR match those from the respective tables in WCAP-17660-NP.

[4] This is stated in the licensee's letter dated September 27, 2018 (Reference 3).

required by 10 CFR Part 50, Appendix G are bounding of the uncertainties associated with the calculated, best-estimate fluence values.

The NRC staff concludes that the fluence calculational method described in WCAP-17660-NP is acceptable for use with the PTLR methodology based on appropriate 54 EFPY fluence projections and conforms to the guidance provided in RG 1.190. Consequently, the NRC staff has reasonable assurance that the proposed PTLR and subsequent updates will use appropriate fluence calculational method inputs with a fluence calculational methodology that adheres to RG 1.190 and is, therefore, acceptable.

3.2.3 P-T Limits

The proposed P-T limits in the PTLR were generated using the BWROG report methodology based on ASME B&PV Code, Section XI, Appendix G. The NRC staff evaluated the proposed P-T limits with respect to three RCPB regions, including: (1) the RPV beltline shell region, (2) components outside the RPV beltline shell region, and (3) ferritic RCPB components outside of the RPV. The staff's evaluation is provided below.

3.2.3.1 RPV Beltline Shell Region (including instrument nozzles)

Table 1, "Pressure and Temperature Requirements for the Reactor Pressure Vessel," of Appendix G to 10 CFR Part 50," provides the requirements for P-T limits and minimum temperature requirements for the RPV. Appendix G to 10 CFR Part 50 states, in part, that the temperatures provided in Table 1 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature.

To evaluate the proposed RPV beltline P-T limits, the NRC staff first reviewed the licensee's selection of limiting materials: Plate B8496-1 with a 54 EFPY ART of 129.1 °F for Unit 1 and N16A/B instrument nozzles with a 54 EFPY ART of 123.4.1 °F for Unit 2. Information on the limiting material (the initial RT_{NDT}, copper (Cu), and nickel (Ni) values) and the fluence information are key inputs to the ART calculation. The NRC staff found that the information in Tables 7 and 8 of the PTLR for the RPV materials is identical to the information in the license renewal application approved in 2006 (i.e., the current licensing basis). The RPV beltline information was originally requested by GL 92-01, Revision 1 (Reference 16), and its supplement (Reference 17). The licensee updated the information through various LARs over the years, considering NRC Regulatory Issue Summary 2014-11 (Reference 18) for RPV nozzles, penetrations, and other discontinuities. The NRC staff also evaluated the fluence methodology and fluence values in Section 3.2.2 of this SE and found them acceptable. Using this verified material and fluence information, the NRC staff independently calculated and confirmed the ART values reported in the LAR. Therefore, the ART values for the limiting beltline material are acceptable for the P-T limit calculation. It should be noted that although the ART is 2 degrees higher for the N16A/B instrument nozzles as compared to Plate B8496-1 for Unit 1, Plate B8496-1 is identified as the limiting beltline material for Unit 1. The NRC staff finds this reasonable because, in addition to the ART, the component geometries and assumed flaws (beltline plate versus instrument nozzle) that affect the subsequent stress and fracture mechanics analyses in determining the P-T limits also contribute to the limiting material determination.

By letter dated September 27, 2018 (Reference 3), the licensee provided responses to the NRC staff's RAIs, which included information on the pressure stress intensity factor (K_{Ip}) and the thermal stress intensity factor (K_{IT}) for beltline, bottom head, and non-beltline regions. Based on this information and the licensee's ART values for the RPV 1/4T location, the NRC staff performed confirmatory P-T limit calculations on selected points on the P-T limit curves to verify the licensee's proposed P-T limits. The difference between the NRC staff's and the licensee's results is around 1 °F for the beltline and 7.5 °F for the N16 instrument nozzle, which is within the discrepancies caused by use of different calculation tools by the licensee and the NRC staff. Further, the NRC staff notes that the licensee's proposed P-T limits for the N16 instrument nozzle are more conservative than the NRC staff's confirmatory calculations of the proposed P-T limits. It should be noted that Enclosure 5 of the licensee's letter dated September 27, 2018, clarified that there are two limiting materials for the beltline region of Unit 2, as indicated by Figures 5 and 6 of the PTLR, rather than one limiting material, as stated in Section 3 of the PTLR. Enclosure 5 of the licensee's letter dated September 27, 2018, also clarified that both the single relief or safety valve (SRV) blowdown thermal transient event and the shutdown transient were used for developing P-T limits, confirming that the Brunswick, Units 1 and 2, unique operation requires consideration of the SRV blowdown transient. Based on calculations performed by the staff to verify the proposed P-T limits, the NRC staff found that the proposed Curve A (pressure test), Curve B (normal operation – core not critical), and Curve C (normal operation – core critical) for each unit was generated correctly and consistent with the requirements of Appendix G to 10 CFR Part 50 and ASME B&PV Code, Section XI, Appendix G.

Further, the NRC staff verified that the proposed P-T limits are consistent with the requirements in Appendix G to 10 CFR Part 50 for the minimum temperature of the closure flange regions. For all proposed P-T limit curves, the far left straight line corresponds to the minimum boltup temperature of 70 °F. This is higher than the reference temperature (RT_{NDT}) of the closure flange regions (16 °F for Unit 1 and 10 °F for Unit 2), as required by Appendix G. Table 1 of Appendix G to 10 CFR Part 50 requires different minimum temperatures for P-T limits depending on whether the pressure is less than or greater than 20 percent of the preservice hydrostatic test pressure. This requirement creates a notch (or bend) in the P-T curves. In the proposed P-T curves, a notch is observed only for the non-beltline P-T limits around 312.6 pounds per square inch gauge (psig) (20 percent of 1,563 psig) because only this region contains the closure flange. For the non-beltline P-T limits of Curve A, the NRC staff verified that the requirements of Table 1 of Appendix G to 10 CFR Part 50 are met by confirming that the notch temperature is 106 °F (16 °F + 90 °F) for Unit 1 and 100 °F (10 °F + 90 °F) for Unit 2. For the non-beltline P-T limits of Curve B for each unit, the NRC staff verified that since Curve B is more limiting than the notch temperature of 136 °F (16 °F + 120 °F) for Unit 1 and 130 °F (10 °F + 120 °F) for Unit 2, no notch exists. For the non-beltline P-T limits of Curve C for each unit, the NRC staff verified that the requirements of Table 1 of Appendix G to 10 CFR Part 50 are met by confirming that the notch temperature of 195 °F for Unit 1 and 188 °F for Unit 2 is greater than the temperature for Curve A at 1,100 psig and 16 °F + 160 °F per Reference 3. Based on the above review of the P-T curves, the NRC staff determined that the proposed P-T limits meet the minimum temperature requirements listed in Table 1 of Appendix G to 10 CFR Part 50.

3.2.3.2 Components Outside of the RPV Beltline Shell Region

Appendix G to 10 CFR Part 50 requires that P-T limits be developed for the ferritic materials in the RPV beltline, as well as ferritic materials not in the RPV beltline. Further, 10 CFR Part 50, Appendix G, requires that all ferritic RCPB components must meet the applicable ASME B&PV Code, Section III requirements. The relevant ASME B&PV Code, Section III requirements that affect P-T limits are the lowest service temperature requirement of subparagraph NB-2332(b) for piping, pumps, and valves, and the fracture toughness requirements of subparagraph NB-3211(d) for vessels.

In NRC Regulatory Issue Summary 2014-11 (Reference 18), the NRC staff noted that P-T limit calculations for ferritic RCPB components that are not RPV beltline shell materials may define P-T curves that are more limiting than those calculated for the RPV beltline shell materials because:

1. RPV nozzles, penetrations, and other discontinuities may exhibit significantly higher stresses than those for the RPV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the RT_{NDT} for these components is lower than the RPV beltline shell materials.

2. Ferritic components that are not part of the RPV may have initial RT_{NDT} values that may define a more restrictive lowest operating temperature in the P-T limits than those for the RPV beltline shell materials.

The proposed P-T limits are based on the BWROG report methodology. The NRC staff note that the P-T limits for the non-beltline region are not part of the existing P-T limits, but appear in the proposed P-T limits. To address this discrepancy, the licensee indicated in its September 27, 2018, letter (Reference 3), that the feedwater nozzle is assumed to be the bounding non-beltline component for development of the proposed Brunswick P-T limits because the combined stresses from the applied thermal and pressure loads are considered to bound all other non-beltline discontinuities. This is consistent with the BWROG report methodology and is, therefore, acceptable. As indicated in the P-T limit curves in Enclosure 5 of the September 27, 2018, letter (Reference 3), the feedwater nozzle curves define part of the composite P-T limits for the entire RPV, demonstrating that the licensee's proposed methodology to include feedwater nozzle curves is more conservative than the current methodology and is acceptable. The September 27, 2018, letter also indicated that the bottom head curve has been updated from the curve in the current P-T limits to bound the SRV blowdown transient. Likewise, this update is conservative and acceptable because the SRV blowdown transient is more severe than the shutdown transient.

Similar to the P-T limits for the beltline region discussed in Section 3.2.3.1 of this SE, the NRC staff performed P-T limit calculations on selected points on the proposed P-T limit curves for the non-beltline and bottom head regions and verified that the differences between the NRC staff's calculated values and the licensee's proposed P-T limits are around 1 °F, which is negligible considering that different calculation tools have been used by the licensee and the NRC staff. Therefore, the NRC staff concludes that the proposed P-T limits based on the feedwater nozzle for the non-beltline region and the proposed P-T limits for the bottom head are acceptable.

The PTLR also states that separate heatup curves based on fluence at 3/4T are not necessary. This is because the BWROG approach conservatively applied the maximum tensile stresses and the irradiated fracture toughness at the 1/4T location for both heatup and cooldown. The NRC staff finds that this approach is consistent with the BWROG report methodology and is acceptable.

3.2.3.3 Ferritic RCPB Components Outside of the RPV

The SE for the BWROG report requires licensees to confirm that all ferritic RCPB components that are not part of the RPV will not define a more restrictive operating temperature than the proposed P-T limits. The PTLR states that the minimum temperature of 70 °F in the proposed P-T limits bounds the lowest service temperature for ferritic non-RPV components of the RCPB per the piping design specifications. Because the licensee properly applied the BWROG report methodology to generate the proposed P-T limits in the PTLR, the staff finds that the licensee has appropriately addressed the condition in the SE for the BWROG report.

Based on the NRC staff's evaluation presented above, the NRC staff determined that the licensee adequately demonstrated that the proposed composite P-T limits for 54 EFPY in the Brunswick PTLR are bounding for all ferritic RPV materials and ferritic RCPB materials, consistent with the requirements of Appendix G to Section XI of the ASME B&PV Code and to 10 CFR Part 50, Appendix G.

3.2.4 Conformance with TSTF-419

TSTF-419, Revision 0 (Reference 6), as approved by the NRC by letter dated March 21, 2002 (Reference 7), and clarified by the NRC staff's letter dated August 4, 2011 (Reference 8), requires that reference to NRC-approved topical reports used in the PTLR methodology must be cited in the TSs using the full citation, including revision number and date of the topical report. The NRC staff reviewed the proposed changes to Brunswick TS 5.6.7 and determined that the revisions properly reference BWROG-TP-11-022-A, Revision 1, dated August 2013. As such, the NRC staff finds the proposed changes to TS 5.6.7 are acceptable and appropriately adopt TSTF-419.

3.3 Technical Conclusion

The NRC staff has reviewed the information provided in the licensee's April 4, 2018; May 29, 2018; September 27, 2018; and December 11, 2018, submittals. The NRC staff concludes that the proposed Brunswick, Units 1 and 2, PTLR is consistent with GL 96-03 with respect to PTLR implementation and, therefore, is approved as part of the Brunswick, Units 1 and 2, licensing bases.

The NRC staff further concludes that the proposed P-T limits valid for 54 EFPY are based on an acceptable methodology documented in BWROG-TP-11-022-A. The NRC staff performed independent evaluations and verified that the ART values and P-T limits are developed appropriately using the BWROG report methodology and satisfy the requirements of Appendix G to Section XI of the ASME B&PV Code and Appendix G to 10 CFR Part 50.

The NRC staff finds that plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50 and that the P-T limits in the TSs are established using a methodology approved by the NRC. Therefore, the NRC staff concludes that the proposed changes continue to meet the requirements of 10 CFR 50.36.

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 February 4, 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 changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on July 17, 2018 (83 FR 33266). 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

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[5] BWRVIP-78NP is the non-proprietary, publicly available version of BWRVIP-78.

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Date: April 22, 2019

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 289 AND 317 TO RELOCATE THE PRESSURE-TEMPERATURE LIMITS IN THE TECHNICAL SPECIFICATIONS TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2018-LLA-0094) DATED APRIL 22, 2019

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