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

January 8, 2020

Mr. John A. Krakuszeski
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. 297 AND 325 TO MODIFY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS 3.4.3.2 AND 3.5.1.11 REGARDING SAFETY RELIEF VALVES (EPID L-2019-LLA-0043)

Dear Mr. Krakuszeski:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 297 to Renewed Facility Operating License No. DPR-71 and Amendment No. 325 to Renewed Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2, respectively. The amendments are in response to your application dated March 4, 2019.

The amendments modify Technical Specification (TS) 3.4.3, "Safety/Relief Valves (SRVs)," Surveillance Requirement (SR) 3.4.3.2 and TS 3.5.1, "ECCS [Emergency Core Cooling System]–Operating," SR 3.5.1.11. The amendments replace the current requirement in these TS SRs to verify the SRVs open when manually actuated with an alternate requirement that verifies that the SRVs are capable of being opened.

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. 297 to DPR-71
2. Amendment No. 325 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. 297
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 March 4, 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. 297, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: January 8, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 297

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 Pages	Insert Pages
3.4-6	3.4-6
3.5-7	3.5-7

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

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.3.2	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. -----	
	Verify each required SRV is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.9	-------------------------------NOTE--------------------------- Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.10	-------------------------------NOTE--------------------------- Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.11	-------------------------------NOTE--------------------------- Not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. --- Verify each required ADS valve is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.12	-------------------------------NOTE--------------------------- Instrumentation response time may be assumed to be the design instrumentation response time. --- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within the limit.	In accordance with the Surveillance Frequency Control Program

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DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 325
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 March 4, 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. 325, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: January 8, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 325

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 Pages	Insert Pages
3.4-6	3.4-6
3.5-7	3.5-7

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

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.3.2	-----------------------------NOTE----------------------------- Not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. --	
	Verify each required SRV is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.9	------------------------------NOTE------------------------------ Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.10	------------------------------NOTE------------------------------ Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.11	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. --- Verify each required ADS valve is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.12	------------------------------NOTE------------------------------ Instrumentation response time may be assumed to be the design instrumentation response time. --- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within the limit.	In accordance with the Surveillance Frequency Control Program

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 297 AND 325

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 letter dated March 4, 2019 (Agencywide Documents Access and Management System Accession No. ML19063B740), Duke Energy Progress, LLC (the licensee) requested U.S. Nuclear Regulatory Commission (NRC, the Commission) approval of proposed amendments to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (Brunswick), Units 1 and 2. The amendments would modify TS 3.4.3, "Safety/Relief Valves (SRVs)," Surveillance Requirement (SR) 3.4.3.2 and TS 3.5.1, "ECCS [Emergency Core Cooling System]–Operating," SR 3.5.1.11. The amendments would replace the current requirement in these SRs to verify the SRVs open when manually actuated with an alternate requirement that verifies that the SRVs are capable of being opened. The method to verify the SRVs open when manually actuated would be available as an alternate test method.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(c)(2)(ii)(C) requires, in part, that a TS limiting condition for operation be established for a component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

The regulation at 10 CFR 50.36(c)(3) requires, in part, that SRs be established to ensure that the necessary quality of components is maintained, and that facility operation will be within safety limits.

The regulation at 10 CFR 50.55a(f) requires, in part, that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with the specified ASME Operation and Maintenance Code (OM Code) and applicable addenda incorporated by reference in the regulations.

3.0 TECHNICAL EVALUATION

3.1 Background

Currently, Brunswick, Units 1 and 2, SR 3.4.3.2 and SR 3.5.1.11 state that each required SRV must be manually actuated with a frequency in accordance with the surveillance frequency control program. This frequency is currently every 24 months during a startup, 12 hours after reactor steam pressure is adequate to perform the test. In its amendment request, the licensee proposed to replace the manual actuation SRV test method in SR 3.4.3.2 and SR 3.5.1.11 with an alternative requirement to verify that the valves are capable of being opened as determined through a series of overlapping tests performed during refueling outages. The test frequency is proposed to be in accordance with the IST Program.

The SRVs at Brunswick, Units 1 and 2, are Target Rock model 7567F two-stage, pilot-operated SRVs. Eleven SRVs per unit are located on the four main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either the safety mode or the relief mode. The SRVs that provide the relief mode are the automatic depressurization system (ADS) valves. The ADS consists of 7 of the 11 SRVs in each unit and is designed to automatically depressurize the reactor vessel, allowing injection by low pressure emergency cooling sources.

3.2 Licensee's Basis for Proposed TS Changes

The licensee proposes the following changes to the Brunswick, Units 1 and 2, TSs.

Current SR 3.4.3.2 states, "Verify each required SRV opens when manually actuated." Current SR 3.5.1.11 states, "Verify each required ADS valve opens when manually actuated." The proposed amendments would change these SRs to require verifying that each required valve "is capable of being opened." Manual actuation testing would be available as an alternative to the proposed SRs.

The licensee states that the verification of the capability to open would be satisfied by a series of overlapping tests that demonstrate the required function of the SRV components. Specifically:

- The simulated automatic actuation test specified in SR 3.5.1.10 and additional surveillances associated with SR 3.3.5.1 demonstrate the ability of various logics and controls to actuate the SRVs up to the point of energizing the solenoids. These tests are performed once per operating cycle (24 months).

- A solenoid valve (SOV) functional test will be performed in-situ for each SRV solenoid valve once per operating cycle (24 months). In the SOV functional test, a test rig with a pressure gauge will be connected downstream of the SOV pneumatic manifold in place of the SRV actuator. Each SOV will be energized, and pneumatic pressure at the downstream connection will be recorded and compared with pneumatic header pressure.

- An SRV actuator functional test will be performed at an offsite test facility as part of certification testing for each SRV pilot assembly. This procedure tests SRV manual mode actuation. The procedure requires applying steam pressure to the SRV at approximately 1,000 pounds per square inch gauge (psig), pressurizing the SRV solenoid to approximately 100 psig, and energizing the SRV solenoid valve with approximately 125 volts direct current. Parameters such as steam inlet pressure, pilot disc motion, main disc motion, solenoid actuation signal, and valve response time are recorded. The current practice of replacing all 11 SRV pilot assemblies each operating cycle (24 months) will be maintained.

- SRV setpoint testing is performed using steam at the offsite test facility as part of certification testing for each SRV pilot assembly at intervals determined in accordance with the IST program. This test is the existing test required by SR 3.4.3.1. In addition to demonstrating that the SRV pilot stage will actuate on high steam pressure in the safety mode, this test overlaps with the pilot assembly actuator functional test to demonstrate that the pilot stage will actuate in the relief mode.

- Currently, the licensee removes and tests all 11 pilot valves every 24-month refueling cycle and actuates each SRV main stage during plant startup to comply with the ASME OM Code 5-year test requirement. Coincident with this license amendment request, the licensee will adopt ASME OM Code Case OMN-17, "Alternative Rules for testing ASME Class 1 Pressure Relief/Safety Valves." OMN-17 extends the frequency for 100 percent removal/refurbishment and as-left certification testing to three refueling outages, or 6 years plus 6 months grace period. Extending the test interval to 6 years would reduce the number of SRVs removed during an individual outage to four, such that the full scope of 11 SRVs would be tested over three refueling cycles while still complying with the other ASME OM Code requirement to test 20 percent of the valves within any 24-month period. Additionally, the 6-month grace period would allow flexibility in the scheduling of certification testing to account for the variability of refueling outage dates.

The TS Bases associated with SR 3.4.3.2 and SR 3.5.1.11 would also be revised to describe these new testing methods.

The licensee states:

> The Boiling Water Reactor Owners' Group (BWROG) Evaluation of NUREG-0737, "Clarification of TMI Action Plan Requirements, Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves," recommends that the number of SRV openings be reduced as much as possible and that unnecessary challenges to the SRVs be avoided.
>
> Experience in the industry has shown that manual actuation of SRVs during plant operation may create a potential for SRV seat leakage. SRV leakage is routed to the suppression pool. The increased heat and fluid additions to the suppression pool requires more frequent suppression pool cooling and more frequent pump-down operations to control suppression pool temperature and level. Main stage SRV seat leakage also tends to mask the indications of SRV pilot stage seat-leakage. Pilot stage leakage could cause spurious SRV actuation and/or SRV failure to reclose after actuation. Excessive leakage would require plant shutdown to replace the leaking SRV.
>
> Reducing or eliminating the number of manual actuations of the SRVs during plant startup minimizes the potential depressurization and cooldown events due to failure-to-close SRV events as well as minimizing the potential for pilot or main

stage leakage of the SRVs. Implementing this change would still maintain the capability to manually open and close SRVs, as necessary, for the IST Program or as corrective action for SRVs with excessive leakage.

The proposed change revises the Frequency for performing SR 3.4.3.2 and SR 3.5.1.11 to be in accordance with the Inservice Testing Program. Specifying the required frequency through the IST Program is not a new technique and occurs throughout the TS (e.g., SR 3.4.3.1 for verifying the safety function lift setpoints of the SRVs). Performing testing in accordance with the IST Program retains appropriate legal control over the testing methodology and specified frequency, since performance is required and is governed by a code adopted into the regulation, i.e., 10 CFR 50.55a. Also, future OM Code changes could then be adopted without requiring a corresponding TS change, allowing NRC endorsed code changes to be more rapidly put in place. Additionally, this will allow crediting IST Program tests performed at frequencies other than the current 24-months as established by the [Brunswick Surveillance Frequency Control Program].

3.3 <u>NRC Staff Evaluation of Proposed TS Changes</u>

The regulation at 10 CFR 50.55a(f) requires that the licensee's IST program meet the requirements of the ASME OM Code. The Brunswick, Units 1 and 2, fifth 10-year interval IST program complies with the 2004 Edition through 2006 Addenda of the ASME OM Code. Specifically, the following ASME OM Code, Mandatory Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," test requirements apply to the Brunswick, Units 1 and 2, IST program for testing the main steam SRVs:

- Section I-1320, "Test Frequencies, Class 1 Pressure Relief Valves," (a) requires that Class 1 pressure relief valves be set pressure tested at least once every 5 years. The licensee will adopt ASME OM Code Case OMN-17, which extends the frequency for 100 percent removal/refurbishment and as-left certification testing to three refueling outages, or 6 years plus 6 months grace period.

- Section I-1320(a) and Code Case OMN-17 further require that a minimum of 20 percent of the SRVs be tested in any 24-month period.

- Section I-1320(c) and Code Case OMN-17 require additional SRVs to be tested if test failures occur within the original test sample of valves.

- Section I-3310, "Periodic Testing/Class 1 Main Steam Pressure Relief Valves With Auxiliary Actuating Devices," (c) requires that the set pressure of each SRV be determined at the frequency specified in I-1320(a).

- Section I-3310(d) requires that the electrical characteristics and pressure integrity be determined for each SOV at the frequency specified in I-1320(a).

- Section I-3310(e) requires that the pressure integrity and stoke capability of each SRV air actuator be determined at the frequency specified in I-1320(a).

- Section I-3310(h) requires that the actuating pressure and electrical continuity of auxiliary actuating device sensing elements of each SRV (as applicable) be determined at the frequency specified in I-1320(a).

- Section I-3400, "Disposition After Testing or Maintenance/Class 1 Main Steam Pressure Relief Valves With Auxiliary Actuating Devices," I-3410, "Class 1 Main Steam Pressure Relief Valves With Auxiliary Actuating Devices," (d) requires that SRVs with auxiliary actuating devices that have been removed for maintenance or testing and reinstalled after meeting the requirements of I-3310 shall have the electrical and pneumatic connections verified either through mechanical/electrical inspection or test prior to the resumption of electrical power generation.

The NRC staff reviewed the licensee's proposed TS changes and determined that the current TS requirement to perform in-situ manual actuation of the SRVs on reactor steam can cause undesirable SRV leakage. On the other hand, the test methods and frequencies proposed by the licensee fully meet the requirements of the ASME OM Code, 2004 Edition through 2006 Addenda, and ASME OM Code Case OMN-17 for safety and relief valves, which the NRC staff has previously found to be acceptable. Additionally, the current manual actuation testing, which is allowed by the ASME OM Code, remains available as an alternative.

Another difference between the current TS-required manual actuation requirements and the licensee's proposal is that when performing the testing in-situ as required by the current TSs, the testing verifies that the SRV discharge lines are not blocked. However, the licensee stated that its maintenance procedures and its foreign material exclusion procedures and practices provide assurance that the discharge piping will remain free of obstructions, and Brunswick, Units 1 and 2, has had no previous instances of test failures due to loss of foreign material exclusion controls. The NRC staff finds that the licensee has acceptably addressed this concern.

In addition, the NRC staff has received other requests for TS changes related to the testing requirements for boiling-water reactor dual-function main steam SRVs. Licensees have determined that in-situ testing of the SRVs on reactor steam can contribute to undesirable seat leakage of the valves during subsequent plant operation and have received NRC approval to perform testing at a test facility coupled with in-situ tests and other verifications of component performance.

3.4 Technical Review Summary

As described above, the licensee has proposed changes to the plant TSs that replace the current requirement to verify the SRVs open when manually actuated with an alternate requirement that verifies the SRVs are capable of being opened. Based on the above evaluation, the NRC staff concludes that the licensee has demonstrated the adequacy of the proposed changes to the Brunswick, Units 1 and 2, TSs. The proposed changes demonstrate proper SRV operation without the need for in-situ testing with reactor steam. Therefore, the proposed changes to SR 3.4.3.2 and SR 3.5.1.11 are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments on November 19, 2019. After discussion with the NRC staff, the State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (84 FR 19968, dated May 7, 2019). 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.

Principal Contributor: Robert Wolfgang

Date: January 8, 2020

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 297 AND 325 TO MODIFY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS 3.4.3.2 AND 3.5.1.11 REGARDING SAFETY RELIEF VALVES (EPID L-2019-LLA-0043) DATED JANUARY 8, 2020

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