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

May 9, 2018

Mr. Bryan B. Wooten
Director – Organizational Effectiveness
Brunswick Steam Electric Plant
8470 River Rd., SE
M/C BNP001
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 & 2 – ALTERNATIVE FOR ISI-09 REGARDING REACTOR PRESSURE VESSEL SHELL WELDS FOURTH TEN-YEAR INSERVICE INSPECTION INTERVAL (EPID L-2018-LLR-0001)

Dear Mr. Wooten:

By letter dated January 23, 2018, as supplemented by letters dated April 11, and 24, 2018, Duke Energy Progress, LLC (Duke Energy, the licensee), submitted to the United States Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the reactor pressure vessel (RPV) shell welds at the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1).

Specifically, the licensee proposes to permanently eliminate the volumetric examination requirements for RPV circumferential welds of the ASME Code, Section XI (Examination Category B-A "Pressure Retaining Welds in Reactor Vessel," Item No. B1.11), for the remainder of the BSEP, Units 1 and 2, fourth ISI interval and through the period of extended operation (PEO). 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

The NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) by meeting the conditions of NRC Generic Letter 98-05, *"Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998*. Therefore, the NRC staff authorizes request for alternative number ISI-09 for the remainder of the fourth ISI intervals of BSEP, Units 1 and 2, and through the PEO which ends on September 8, 2036, for BSEP, Unit 1, and December 27, 2034, for BSEP, Unit 2.

All other requirements of Section XI of the ASME Code for which relief was not specifically requested and approved in the subject relief requests remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Andy Hon, at 301-415-8480 or Andrew.Hon@nrc.gov

Sincerely,

Brian Tindell

Brian Tindell, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-324

Enclosure:
Safety Evaluation

cc: Listserv

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# REQUEST FOR ALTERNATIVE NO. ISI-09

# REACTOR PRESSURE VESSEL SHELL WELDS

# FOURTH TEN-YEAR INSERVICE INSPECTION INTERVAL AND

# PERIOD OF EXTENDED OPERATION

# DUKE ENERGY PROGRESS, LLC

# BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

# DOCKET NOS. 50-325 AND 50-324

# EPID NO. L-2018-LLR-0001

## 1.0 INTRODUCTION

By letter dated January 23, 2018 (Reference 1), as supplemented by letter dated April 11, and 24, 2018 (Reference 2 and 10), Duke Energy Progress, LLC (Duke Energy, the licensee), submitted to the U.S Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the reactor pressure vessel (RPV) shell welds at the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1). Specifically, the licensee proposes to permanently eliminate the volumetric examination requirements for RPV circumferential welds of the ASME Code, Section XI (Examination Category B-A "Pressure Retaining Welds in Reactor Vessel," Item No. B1.11), for the remainder of the BSEP, Units 1 and 2, fourth ISI interval and through the period of extended operation (PEO). Details of the licensee's proposed alternative are in Section 3.3 of this safety evaluation (SE). 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The NRC staff discussed the attendant regulations and requirements in Section 2.0 of this SE. Electric Power Research Institute proprietary report TR-105697 "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)" (Reference 3), contains the technical basis for the proposed alternative. The licensee followed the guidance for meeting the conditions of NRC Generic Letter 98-05, *"Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998* (Reference 4 and 9).

Enclosure

## 2.0 REGULATORY EVALUATION

### 2.1 Applicable Requirements of 10 CFR Part 50 and Licensing Bases

10 CFR 50, Appendix A, Criterion 31, Fracture prevention of reactor coolant pressure boundary

Insofar as it requires that the reactor coolant system (RCS) boundary design shall have sufficient margin to assure that when stressed under maintenance and testing conditions: (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. To accomplish this, one of the items is for the design to consider service temperatures of the boundary material under maintenance and testing conditions.

10 CFR 50, Appendix G, Fracture Toughness Requirements

Insofar as it requires to meet the fracture toughness for the ferritic materials of the RPV with adequate safety margins during normal operations and hydrostatic test with core not critical, the Pressure-Temperature (P-T) limits of the RCS as defined in this Appendix should be met.

10 CFR 50.55a(z)(1), Request for Alternatives

Licensees may use and pursue alternatives to the requirements in 10 CFR 50.55a(g)(4). One method of pursuing alternatives is defined in 10 CFR 50.55a(z)(1), which states that licensees shall demonstrate that the alternative would provide an acceptable level of quality and safety.

### 2.2 BWRVIP-05 and NRC Generic Letter 98-05

The technical basis for the licensee's proposed alternative is BWRVIP-05, which calculates conservative conditional probabilities of failures for RPV welds. The basic principle for justifying the proposed alternative is to demonstrate that the conditional probabilities of failures of the BSEP, Units 1 and 2, RPV welds are lower than the conservative values determined in the July 28, 1998, SE BWRVIP-05. Section 3, "Conclusions" of the July 28, 1998, SE of BWRVIP-05 states that since the failure frequency for the limiting circumferential weld could significantly increase through the PEO, the NRC staff will be requesting plants to perform plant-specific assessments that consider weld chemistry and neutron fluence at the end of the PEO. The July 28, 1998 SE of BWRVIP-05 states that licensees may also request relief from the requirements of ASME Code, Section XI, Examination Category B-A, Item No. B1.11, by demonstrating the following[1] conditions:

1) At the expiration of the license[2], the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, SE of BWRVIP-05.

2) Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's

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[1] These two conditions are also stated in NRC Generic Letter 98-05 (ADAMS Accession No. ML082460066), which the licensee referenced.

[2] "License" in this case refers to the renewed license (i.e., applicable through the PEO) since the licensee received NRC approval of relief for the same welds for the remaining term of operation by letter dated September 14, 2000 (Reference 5).

July 28, 1998 SE of BWRVIP-05. Licensees will still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Section 4 "Implementation" of the July 28, 1998, SE of BWRVIP-05 states that if the axial weld examinations reveal an active mode of degradation, the examination of the circumferential welds shall be performed.

On November 10, 1998, the NRC issued Generic Letter 98-05 (Reference 9), which informed the industry the completion of NRC staff evaluation of BWRVIP-05. Thus, the boiling-water reactor licensees could request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds by demonstrating the above two conditions in BWRVIP-05 are met.

Brunswick Current Licensing Bases

The RPV shell welds at BSEP, Units 1 and 2, are ASME Code, Class 1 components, whose ISI are performed in accordance with Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," of the ASME Code and applicable edition and addenda, as required by 10 CFR 50.55a(g). Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(a)(1)(ii) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications in 10 CFR 50.55a(b)(2). The Code of Record for BSEP, Units 1 and 2, for the fourth 10-year ISI interval is the 2003 Addenda to the 2001 Edition of the ASME Code, Section XI.

Technical specification (TS ) Section 3.4.9, Figure 3.4.9-1, provide the RCS P-T limit curves for heatup/cooldown and maximum rate of change of RCS temperature with core not critical under normal operating conditions. Surveillance requirement (SR) 3.4.9-1 verifies that the RCS heatup and cooldown operations are within the limits specified in Figure 3.4.9-1. In the same Section, Figures 3.4.9-3, 3.4.9-4, and 3.4.9-5 provide the RCS P-T limit curves for hydrostatic and in-service leak tests. SR 3.4.9-2 verifies the RCS P-T limits are within those specified in Figures 3.4.9-3, 3.4.9-4, and 3.4.9-5 for these tests.

By letter dated September 14, 2000 (Reference 5), the NRC staff issued its SE that approved the proposed alternative for the remaining term of operation under the existing license of BSEP, Units 1 and 2. Subsequently, the licensee performed a time-limited aging analysis (TLAA) of the RPV circumferential welds through the end of the PEO in its 2004 license renewal application (LRA) (Reference 6) for BSEP, Units 1 and 2. The NRC staff approved the LRA in 2006, as documented in SE report NUREG-1856 "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2" (Reference 7). The NRC staff's evaluation of the TLAA of the RPV circumferential welds (circumferential weld TLAA) through the end of the PEO is in Section 4.2.5 "RPV Circumferential Weld Examination Relief" of NUREG-1856.

## 3.0 TECHNICAL EVALUATION

### 3.1 ASME Code Requirements

The specific examination requirement for RPV shell welds is volumetric examination of essentially 100 percent of the weld length of the volume defined in Figure IWB-2500-1 "Vessel Shell Circumferential Weld Joints" of the ASME Code, Section XI, as specified in Table IWB-2500-1, "Examination Categories" of the ASME Code, Section XI, Examination Category B-A, Item No. B1.11.

### 3.2 ASME Code Components Affected

The licensee proposes the alternative for following ASME Code components (as stated in the submittal):

| | |
|---|---|
| Unit(s) Affected: | Brunswick Steam Electric Plant (BSEP), Units 1 and 2 |
| Code Class: | ASME Code, Section XI, Class 1 |
| References: | Subarticle IWB-2500, Table IWB-2500-1 |
| Examination Categories: | B-A, "Pressure Retaining Welds in Reactor Vessel" |
| Item Numbers: | B1.11, "Circumferential Shell Welds" |
| Component Numbers: | 1B11-RPV-DA, 1B11-RPV-DB, 1B11-RPV-DC, 1B11-RPV-K, 2B11-RPV-DA, 2B11-RPV-DB, 2B11-RPV-DC, 2B11-RPV-K |
| Description: | Volumetric Examination Coverage |

### 3.3 Licensee's Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), in lieu of the ASME Code requirements stated in Section 2.2 of this SE, the licensee proposes the following alternative (as stated in the submittal):

> The alternative plan will require performance of RPV vertical weld examinations and incidental examination of 2 to 3 percent of the intersecting circumferential shell welds to the maximum extent possible based on accessibility. The RPV circumferential welds will be permanently deferred until facility operating license expiration. This alternative aligns with BWRVIP-05.
>
> The axial weld seams (i.e., Examination Category B-A, Item No. B1.12) and their intersection with the associated RPV circumferential weld seams will be examined in accordance with ASME Section XI except where specific relief is granted when essentially 100 percent (i.e., greater than 90 percent) coverage cannot be obtained.

### 3.4 Licensee's Justifications to address the two conditions of BWRVIP-05

To evaluate Condition 1, the licensee performed a TLAA of the RPV circumferential welds through the end of the PEO in its 2004 LRA (Reference 6) for BSEP, Units 1 and 2. The NRC renewed the operating licenses in 2006, as documented in SE report NUREG-1856 "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2" (Reference 7). The NRC staff's evaluation of the TLAA of the RPV circumferential welds (circumferential weld TLAA) through the end of the PEO is in Section 4.2.5 "RPV Circumferential Weld Examination Relief" of NUREG-1856.

In Section 6 "Basis for Use" of the enclosure to the submittal, the licensee included Table 4.2.5-1 "Comparison of NRC and CP&L 54 EFPY Mean $\Delta RT_{NDT}$ Calculations to the 64 EFPY (effective full-power year) Mean $\Delta RT_{NDT}$ Calculations for the Limiting CB&I Case Study on BWRVIP-05" of NUREG-1856. Table 4.2.5-1 of NUREG-1856 compares conditional probabilities of failures for the RPV circumferential welds, computed for a bounding 64 EFPY case study on BWRVIP-05, with those from the NRC staff's and licensee's calculations for 54 EFPY for BSEP, Units 1 and 2. The 54 EFPY represents the operational condition of BSEP, Units 1 and 2, at the end of the PEO. Note 2 of Table 4.2.5-1 of NUREG-1856 includes the acceptance criterion for the conditional probability of failure the NRC staff approved: if the values of mean $\Delta RT_{NDT}$ for the limiting RPV circumferential welds at BSEP, Units 1 and 2, are less than the mean $\Delta RT_{NDT}$ of the bounding case study, then the conditional probability of failure for the limiting RPV circumferential welds is less than that for the bounding case study. Table 4.2.5-1 of NUREG-1856 shows that the mean $\Delta RT_{NDT}$ for the limiting RPV circumferential welds is 6.6 °F (degree Fahrenheit) for BSEP, Unit 1, and -31.4 °F for BSEP, Unit 2. The mean $\Delta RT_{NDT}$ for the bounding case study is 70.6 °F. Since 6.6 °F and -31.4 °F are less than 70.6 °F, the conditional probability of failure for the limiting RPV circumferential welds at BSEP, Units 1 and 2, is less than that for the bounding case study, therefore, satisfying Condition 1. Note 2 of Table 4.2.5-1 of NUREG-1856 further states that plants that meet the above acceptance criterion may conclude that the conditional of probability of failure for the limiting RPV circumferential weld is low enough to justify elimination of the required ASME Code volumetric examinations for RPV circumferential welds.

The licensee stated that in November 2012 the 54 EFPY fluence values for BSEP, Units 1 and 2, were updated and that the updated values are lower than the fluence values in Table 4.2.5-1 of NUREG-1856. The licensee, therefore, concluded that the fluence values in Table 4.2.5-1 of NUREG-1856 are still bounding.

The July 28, 1998, SE of BWRVIP-05 Section 2.3 states if the axial weld examinations reveal an active mode of degradation, the examination of the circumferential welds shall be performed. The licensee stated in Section 6 of the enclosure to the submittal, in the section titled "Inspection of Axial Welds" that the recent examinations to date of the RPV axial weld seams of the BSEP, Units 1 and 2, revealed no unacceptable indications.

To evaluate Condition 2, the licensee stated that the operating procedures for BSEP are sufficient to prevent a cold over-pressure event from occurring during activities such as the system leak test performed at the conclusion of a refueling outage. Therefore, a challenge to the BSEP reactor pressure vessel from a non-design basis cold over-pressure transient is unlikely.

## 3.5 NRC Staff Evaluation

### 3.5.1 Condition 1: Satisfying the Conditional Probability of Failure for Circumferential Welds through the Period of Extended Operation

#### Circumferential Welds

As discussed in Section 3.4 of this SE, the licensee included Table 4.2.5-1 of NUREG-1856 in the submittal as its basis for satisfying condition 1 of the July 28, 1998, SE of BWRVIP-05. Table 4.2.5-1 of NUREG-1856 shows that the mean $\Delta RT_{NDT}$ for the limiting RPV circumferential welds for BSEP, Units 1 and 2, are lower than the mean $\Delta RT_{NDT}$ for the bounding case study, therefore, satisfying condition 1. The NRC staff's review of the licensee's basis is discussed below.

The NRC staff noted that the licensee's circumferential weld TLAA summarized in Table 4.2.5-1 of NUREG-1856 in 2006 was approved as part of NUREG-1856. However, surveillance capsule test data that are withdrawn and/or tested after 2006 can potentially impact the values in Table 4.2.5-1 of NUREG-1856, and therefore invalidate them, especially values of mean $\Delta RT_{NDT}$. The NRC staff, therefore, reviewed the capsule withdrawal schedule in the current licensing basis of BSEP, Units 1 and 2. This capsule withdrawal schedule is contained in the integrated surveillance program (ISP) in BWRVIP-86, Revision 1-A (Reference 8), as indicated in Section 5.3.1.6 "Material Surveillance" of the updated final safety analysis report (UFSAR). The NRC staff noted that the licensee updated the fluence values in November 2012, as stated in the submittal. In a supplement dated April 11, 2018 (Reference 2), in response to a request for additional information, the licensee stated that the updated fluence values in November 2012 were not related to capsule tests for BSEP, Units 1 and 2, and confirmed that no further capsules are scheduled for removal. The licensee also confirmed in the supplement that the calculation of fluence values performed in November 2012 was consistent with NRC-approved methodology. The NRC staff finds the information the licensee provided in the supplement acceptable.

The NRC staff determined that since the fluence values the licensee calculated in November 2012 are lower than the fluence values presented in Table 4.2.5-1 of NUREG-1856, the mean $\Delta RT_{NDT}$ values in Table 4.2.5-1 of NUREG-1856 are still bounding. Furthermore, surveillance capsule tests do not impact the mean $\Delta RT_{NDT}$ values in Table 4.2.5-1 of NUREG-1856 since no more surveillance capsules are scheduled for removal. Therefore, the NRC staff determined that the condition probability of failure evaluation for the RPV circumferential welds at BSEP, Units 1 and 2, presented in in Table 4.2.5-1 of NUREG-1856 is still valid.

Based on the discussion above, the NRC staff determined that the licensee has adequately shown that the conditional of probability of failure for the limiting RPV circumferential weld applicable to BSEP, Units 1 and 2, through the PEO is bounded by the analysis in the July 28, 1998, SE of BWRVIP-05. Therefore, the NRC staff determined that the licensee has adequately satisfied Condition 1 of the July 28, 1998, SE of BWRVIP-05 and that the elimination of the required ASME Code volumetric examinations of the circumferential welds listed in Section 3.2 of this SE PEO is justified.

Axial Welds

As stated in the proposed alternative in Section 3.3 of this SE, the licensee will examine the RPV axial welds of the BSEP, Units 1 and 2 in accordance with the requirements of the ASME Code, Section XI. Additionally, the July 28, 1998, SE of BWRVIP-05 (see Section 2.3 of this SE) states if the axial weld examinations reveal an active mode of degradation, the examination of the circumferential welds shall be performed. The licensee stated in Section 6 of the enclosure to the submittal, in the section titled "Inspection of Axial Welds," that the recent examinations to date of the RPV axial weld seams of the BSEP, Units 1 and 2, revealed no unacceptable indications. The NRC staff finds this acceptable.

With respect to conditional probabilities of failure for the RPV axial welds, the licensee also performed a TLAA of the RPV axial welds (axial weld TLAA) through the PEO in its 2004 LRA, and the NRC staff evaluated and approved it in Section 4.2.6 "RPV Axial Weld Failure

Probability" of NUREG-1856. The licensee has shown in the axial weld TLAA that the conditional probability for failure for the RPV axial welds of the BSEP, Units 1 and 2, through the PEO is less than the bounding case. Similar to the circumferential weld TLAA, the NRC staff noted that the axial weld TLAA was approved in 2006 during publication of NUREG-1856. However, the NRC staff determined that the axial weld TLAA is still valid through the PEO of BSEP, Units 1 and 2, for the same reasons the circumferential weld TLAA is valid.

3.5.2 Condition 2: Implementing operator training and established procedures that limit the frequency of cold over-pressure events

Operator Training

The licensee described some of the elements of the operator training to prevent the low temperature over pressure (LTOP) events. The following are the key elements:

- The training is conducted periodically,
- The training includes training on RCS material brittle fracture limits and compliance with the TS P-T limit curves,
- It reinforces the plant operators to strict comply with the operating procedures,
- The licensee continuously reviews industry operating experience to ensure that its procedures consider the impact of LTOP events,
- The licensee implements appropriate changes to procedures and training to prevent similar events.

The NRC staff finds the above elements of the licensee's operator training adequate to reduce the possibility of LTOP events.

Procedural Controls

The licensee described several procedural controls to minimize the possibility of an LTOP event. The following are the key procedural controls that would prevent an adverse impact on the RPV water level, pressure, or temperature during the cold shutdown and refueling operations:

- Frequent monitoring and controlling of the RPV level, pressure, and temperature. The control room operator is required to provide positive control of reactor water level and pressure within the specified bands, promptly report when operating outside the band, and make sure the restoration action is taken.
- Frequent monitoring of control room alarms and indications is performed by the control room operators in order to detect abnormalities as early as possible.
- Shift turnover procedures require operators to discuss plant status and on-going activities that could affect critical plant parameters. This ensures that on-coming operators are aware of any activities which could adversely affect reactor water level, pressure, or temperature.

- A senior reactor operator (SRO) directs and/or is informed of changes that affects reactor water level, pressure, or temperature, so that any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. The SRO maintains cognizance of any activity which could potentially affect reactor level or decay heat removal during refueling outages.

- Outage oversight is performed by dedicated SROs. Outage work is coordinated through the outage command center, which provides an additional level of operations oversight. For the daily outage activities, a plan-of-the-day (POD) is developed and approved by the management. The POD lists the work activities to be performed on any particular day.

- Work activities that have the potential of affecting critical reactor parameters, are discussed in pre-job briefings attended by the cognizant individuals involved.

- An integral part of how operators are trained is the procedural controls for reactor temperature, level, and pressure, as well as responding to abnormal water level conditions outside the established limits.

The NRC staff considers the above controls acceptable and would minimize the risk of RCS LTOP events.

High Pressure Injection Sources

The licensee identified the following high pressure systems that may inadvertently inject fluid into the RPV when it is at a low temperature:

- High Pressure Coolant Injection (HPCI) system
- Reactor Core Isolation Cooling (RCIC) system
- Normal Feedwater (NFW) supply by the reactor feedwater pumps
- Control Rod Drive (CRD)
- Reactor Water Cleanup (RWCU) system
- Standby Liquid Control (SLC) system

The HPCI, RCIC, and NFW systems are steam turbine driven systems from the reactor steam. During the cold shutdown condition of the reactor, the reactor steam is not available for operation of these system pumps. Therefore, it is not possible for these systems to lead to an over-pressure event while the RPV is in a cold shutdown condition.

The CRD and RWCU systems are used to control the RPV pressure and level using a feed-and-bleed process in the cold shutdown condition. If one of these systems is not available, the operator would use the other system to control the RPV water level. The licensee stated that during the feed-and-bleed process the RPV is not filled solid with water except for the performance of hydrostatic testing. The CRD system injects water into the RPV typically with a flowrate of less than 100 gallons per minute (gpm). The low flow injection allows the operator sufficient time to react to unanticipated level changes and, thus, significantly reduces the possibility of an event that would result in a violation of the TS P-T limits.

The SLC system is a high pressure injection system that needs to be evaluated for causing an LTOP event. This system is not provided with an automatic start signal. It is designed to be manually initiated from the control room by operating a keylock switch. The SLC injection rate into the RPV is approximately 43 gpm from one SLC pump and 86 gpm from two pumps. In the event of an inadvertent operation, the low SLC flowrate would allow the operator sufficient time to react to control the RPV pressure thus eliminating the possibility of an LTOP event that violates the TS 3.4.9 P-T limits.

Low Pressure Injection Sources

The licensee identified the following low pressure sources that may inadvertently inject fluid into the RPV when it is at a low temperature:

- Condensate System
- Core Spray (CS) System
- Residual Heat Removal (RHR) System

The licensee stated that the condensate booster pumps can inject water at up to about 400 psig. However, after reactor shutdown, when the system is no longer required to control reactor level, the condensate system is secured and the pumps are placed in manual control. Also following shutdown of the condensate system, the feedwater line containment isolation valves are closed, thereby, isolating the injection path. These valves are not reopened until the condensate system is restarted and positive control of the flow rate established.

In the Enclosure of the submittal, the second paragraph under the heading “Review of Low Pressure Injection Sources” states:

> For the low pressure make-up systems, the Core Spray and Residual Heat Removal systems, these system's pumps have a shutoff head of approximately 313 psig and 250 psig, respectively. The BSEP pressure-temperature limit curves for hydrostatic testing allow pressures up to 313 psig at a temperature of 70 °F.

The NRC staff noted an inconsistency between the information in the above statement and the BSEP Units 1 and 2, TS Section 3.4.9, P-T limit curves in Figures 3.4.9-3, 3.4.9-4, and 3.4.9-5 for hydrostatic and leak tests. The RPV beltline curves in these figures allows a maximum pressure of 283 psig in the RCS temperature range of 70 °F to 110 °F. In addition, the RPV beltline curve in TS Figure 3.4.9-1 for the RPV heatup/cooldown also requires to operate below 283 psig (pounds per square inch guage) pressure in the RCS temperature range of 70 °F to 110 °F. In the supplement dated April 24, 2018 (Reference 10), the licensee corrected the above information by stating that the 313 psig stated in the relief request (RR) is incorrect; the allowed pressure limit of 283 psig in TS Figures 3.4.9-1, 3.4.9-3, 3.4.9-4 and 3.4.9-5 is correct. The licensee clarified that in the development of the proposed RR, the stated value 313 psig of the pressure limit was incorrectly transposed from the previous RR dated June 21, 2000. The P-T curves were revised to a pressure limit of 283 psig by a license amendment request dated June 26, 2002 (References 11), and approved by the NRC staff on June 18, 2003 (Reference 12). The revised pressure limit was conservatively reduced from 313 psig to 283 psig which compensates for the pressure and temperature instrument uncertainties. The NRC staff finds the licensee's response acceptable by correcting the error.

The NRC staff noted an inconsistency regarding the Core Spray (CS) pump shutoff head of 313 psig in the above statement and UFSAR Figure 6-49 which shows its shutoff head as approximately 790 ft (approximately 342 psig based on a water density of 62.4 $lb/ft^3$). In the supplement dated April 24, 2018 (Reference 10), the licensee clarified that UFSAR Figure 6-49 showing the CS pump shutoff head as 790 feet is the original and correct curve and has not been changed since plant startup. The licensee acknowledged that the CS pumps are capable of providing pressure in excess of the 283 psig allowed pressure limit.

For the evaluation of the possibility of an LTOP event due to an inadvertent operation of the Low pressure coolant injection or the CS system, the licensee referred to the July 28, 1998, SE of BWRVIP-05 (Reference 4). In Appendix C, Section C.1.4 of the SE, it is acknowledged that these systems would not represent a significant challenge to the RPV based on their pump shutoff heads. In Section C.1.8 of the safety evaluation report (SER) it is stated that high system flow rates of these systems are capable of quickly increasing the RPV water level and will affect the time available for recovery. The SER also stated that extremely unlikely actions would have to take place for an LTOP event to occur, which include: (a) operators violating the P-T curves, (b) operators ignoring RPV level instrumentation, (c) isolating the vessel, and (d) continual water injection into RPV via CRD flow for an extended period of time.

An inadvertent operation of the CS system, which has its pump shutoff head greater than the allowed pressure limit of 283 psig, during startup from cold shutdown, and during the ASME Code, Section XI, required RPV pressure test, the operator action, and compliance with plant operating procedures will prevent an LTOP event. During startup the main steam isolation valves are open to provide a greater system volume than the RPV alone. The greater system volume provides operators a longer response time to an inadvertent initiation of the CS system. Prior to performance of the ASME Code, Section XI, pressure test, operators are instructed on the requirements to maintain RCS conditions within the boundaries of the applicable TS P-T curves. The NRC staff, therefore, considers that the risk of pressurization of the RCS above the boundaries of the TS P-T curve is minimized through timely operator action, compliance with plant operating procedures, and continued training on material brittle fracture limits, for LTOP events.

Therefore, the NRC staff determined that the licensee has adequately satisfied Condition 2 of the July 28, 1998, SE of BWRVIP-05 and that the elimination of the required ASME Code volumetric examinations of the circumferential welds listed in Section 3.2 of this SE during PEO is justified.

## 4.0 CONCLUSION

As set forth above, the NRC staff determined the licensee's proposed alternative provides an acceptable level of quality and safety because both Condition 1 and Condition 2 of BWRVIP-05 and Generic Letter 98-05 are satisfied. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes request for alternative number ISI-09 for the remainder of the fourth ISI intervals of BSEP, Units 1 and 2, and through the PEO which ends on September 8, 2036, for BSEP, Unit 1, and December 27, 2034, for BSEP, Unit 2.

All other requirements of Section XI of the ASME Code for which relief was not specifically requested and approved in the subject relief requests remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

## 5.0 REFERENCES

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Nos. DPR-71 and DPR-62 – Docket Nos. 50-325 and 50-324 – Inservice Inspection Program Proposed Alternative ISI-09 In Accordance With 10 CFR 50.55a(z)(1) Regarding Reactor Pressure Vessel Circumferential Shell Examinations," January 23, 2018 ([Agencywide Documents Access and Management System] ADAMS Accession No. ML18023A134).

2. Letter from Wooten, Bryan B. (Duke Energy) to the U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 – Renewed Facility Operating License Nos. DPR-71 and DPR-62 – Docket Nos. 50-325 and 50-324 - Response to Request for Additional Information Regarding Inservice Inspection Program Proposed Alternative ISI-09 In Accordance With 10 CFR 50.55a(z)(1) Regarding Reactor Pressure Vessel Circumferential Shell Examinations," April 11, 2018 (ADAMS Accession No. ML18102A004).

3. Electric Power Research Institute Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995 (ADAMS Accession No. ML032200246).

4. Lainas, Gus C., U.S. Nuclear Regulatory Commission letter to Carl Terry, BWRVIP Chairman, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," July 28, 1998 (ADAMS Legacy Library No. 9808040037).

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9. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998 (ADAMS Accession No. ML031110082).

10. Letter from Duke Energy to NRC dated April 24, 2018, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Response to Request for Additional Information regarding Inservice Inspection Program Proposed Alternative ISI-09 In Accordance With 10 CFR 50.55a(z)(1) Regarding Reactor Pressure Vessel Circumferential Shell Weld Examinations," (ADAMS Accession No. ML18115A001)

11. Letter from Carolina Power & Light dated June 26, 2002, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62

Request for License Amendments to Revise Technical Specification Pressure-Temperature Limit Curves," (ADAMS Package Accession No. ML021890157)

12. Letter from NRC to Carolina Power & Light dated June 18, 2003, "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Re: Pressure-Temperature Limit Curves (TAC Nos. MB5579 AND MB5580)," (ADAMS Accession No. ML031690683)

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