Brunswick Nuclear Plant
P.O. Box 10429
Southport, NC 28461

January 23, 2018

Serial: BSEP 18-0015 10 CFR 50.55a(z)(1)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: 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
Inservice Inspection Program Proposed Alternative ISI-09 In Accordance With 10 CFR 50.55a(z)(1) Regarding Reactor Pressure Vessel Circumferential Shell Weld Examinations

Reference:

1. Letter from Warren J. Dorman (Carolina Power & Light Company) to U.S. Nuclear Regulatory Commission Document Control Desk, *Volumetric Examination of Circumferential Reactor Pressure Vessel Welds*, dated June 21, 2000, ADAMS Accession Number ML003727229.

2. Letter from Richard P. Correia (USNRC) to J. S. Keenan (Carolina Power & Light Company), *Safety Evaluation for Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i) for Reactor Pressure Vessel Circumferential Shell Weld Examinations (TAC Nos. MA9299 and MA9300)*, dated September 14, 2000, ADAMS Accession Number ML003749906.

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(z)(1) , Duke Energy Progress, LLC (Duke Energy), hereby requests approval of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2001 Edition through 2003 Addenda for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

Currently, examination of reactor pressure vessel (RPV) circumferential shell welds is required by Table IWB-2500-1, Category B-A, Item B1.11. Duke Energy requests approval of the enclosed alternative, ISI-09, to eliminate the RPV circumferential shell weld examinations as previously allowed by relief request (i.e., Reference 1) and the NRC's approval dated September 14, 2000 (i.e., Reference 2). Approval of this alternative is requested by the end of the current 10-year inservice inspection interval (i.e., by May 10, 2018) on the basis that the proposed alternative provides an acceptable level of quality and safety. This alternative will apply through the period of extended operation (i.e., September 8, 2036, for BSEP, Unit 1, and December 27, 2034, for BSEP, Unit 2).

The proposed alternative is provided in the enclosure to this submittal.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager – Regulatory Affairs, at (910) 832-2487.

Sincerely,

Bryan B. Wooten
Director – Organizational Effectiveness
Brunswick Steam Electric Plant

WRM/wrm

Enclosure: 10 CFR 50.55a Request Number ISI-09 cc (with enclosure):

U.S. Nuclear Regulatory Commission, Region II
ATTN: Ms. Catherine Haney, Regional Administrator
245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

U.S. Nuclear Regulatory Commission
ATTN: Mr. Gale Smith, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U.S. Nuclear Regulatory Commission
ATTN: Mr. Andrew Hon (Mail Stop OWFN 8G9A)
11555 Rockville Pike
Rockville, MD 20852-2738

Chair - North Carolina Utilities Commission **(Electronic Copy Only)**
4325 Mail Service Center
Raleigh, NC 27699-4300
swatson@ncuc.net

Mr. Cliff Dautrich, Bureau Chief
North Carolina Department of Labor
Boiler Safety Bureau
1101 Mail Service Center
Raleigh, NC 27699-1101

10 CFR 50.55a Request Number ISI-09

Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)

1. ASME Code Components Affected

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

2. Applicable Code Edition

The Inservice Inspection Program for the fourth 10-year inservice inspection interval is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2001 Edition with 2003 Addenda. The fourth 10-year inservice inspection interval began on May 11, 2008, and will end on May 10, 2018.

The Inservice Inspection Program for the fifth 10-year inservice inspection interval will be based on the ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition with 2008 Addenda. The fifth 10-year Inservice Inspection Interval is scheduled to start on May 11, 2018 and end on May 10, 2028. The Inservice Inspection Program for subsequent 10-year inservice inspection intervals will comply with the applicable ASME Code, Section XI Edition and Addenda required by 10 CFR 50.55a(g)(4)(ii), with applicable 10 CFR 50.55a(b)(2) conditions.

3. Applicable Code Requirement

The ASME Code, Section XI, 2001 Edition through the 2003 Addenda (i.e., and the 2007 Edition with the 2008 Addenda), Table IWB-2500-1, Examination Category B-A, Item B1.11 requires a volumetric examination of the reactor pressure vessel (RPV) circumferential shell welds each inspection interval.

4. Reason for Request

Electric Power Research Institute (EPRI) report TR-105697, *BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)* (i.e., Reference 1), concluded that the failure frequency for circumferential welds in boiling water reactor (BWR) plants is sufficiently low enough to be below the criterion specified in Regulatory Guide 1.154, *Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors* (i.e., withdrawn by the NRC on January 14, 2011, via

76 FR 2726). The NRC endorsed this position with the staff's evaluation of the BWRVIP-05 report issued July 28, 1998 (i.e., Reference 2).

On November 10, 1998, the NRC issued Generic Letter 98-05 (i.e., Reference 3) to inform BWR licensees that the NRC completed their review of BWRVIP-05 and that licensees could request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV circumferential shell welds.

On June 21, 2000 (i.e., Reference 4), Duke Energy submitted a proposed alternative for permanent relief (i.e., for the remaining term of operation under the existing, initial, license) based on BWRVIP-05, from the ASME Code, Section XI requirement to perform the volumetric examination of RPV circumferential shell welds. The NRC approved this request for permanent relief, for the remaining term of operation under the initial operating license, by letter dated September 14, 2000 (i.e., Reference 5).

Subsequently, on June 26, 2006 (i.e., Reference 6), the NRC issued the Renewed Facility Operating License for BSEP, Units 1 and 2. In NUREG-1856, "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2," (i.e., Reference 7), the NRC documented its assessment for the time-limited aging analysis (TLAA) on the RPV circumferential weld relief request in Section 4.2.5.4 of the safety evaluation for the License Renewal Application (LRA).

Based on the above, Duke Energy is now requesting Proposed Alternative ISI-09 in order to apply BWRVIP-05 to the period of extended operation ending September 8, 2036, for BSEP, Unit 1, and December 27, 2034, for BSEP, Unit 2. This request is relevant to the current (i.e., fourth) inservice inspection interval because, without relief, examination of the RPV circumferential shell welds is required by the end of the inspection interval (i.e., May 10, 2018), in accordance with the applicable ASME Code requirements.

5. <u>Proposed Alternative</u>

In accordance with 10 CFR 50.55a(z)(1), an alternative is requested to the requirement of the ASME Code, Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," Examination Item Number B1.11, "Circumferential Shell Welds," which requires a volumetric examination of 100 percent of the RPV circumferential shell welds to be performed during each inspection interval.

In lieu of the examination requirements of Category B-A, Item B1.11, the following alternative is proposed:

> 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.

6. <u>Basis for Use</u>

The technical basis supporting this proposed alternative is provided by BWRVIP-05 (i.e., Reference 1) as accepted in the NRC's final safety evaluation report enclosed in a July 28, 1998, letter from Mr. G. C. Lainas (NRC) to Mr. C. Terry (BWRVIP Chairman) (i.e., Reference 2). In this letter, the NRC concluded that "since the failure frequencies for circumferential welds in BWR plants are significantly below the criteria specified in Regulatory Guide (RG) 1.154 and the CDF of any BWR plant, and that continued future inspections would result in a negligible decrease in an already acceptably low value, elimination of ISI for RPV circumferential welds is justified."

NRC Generic Letter 98-05 documented completion of the review of the BWRVIP-05 report. The generic letter stated that BWR licensees may request permanent relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds by demonstrating the following:

1. At the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC's July 30, 1998, safety evaluation, and

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's July 30, 1998, safety evaluation.

The generic letter also stated that licensees still need to perform their required inspections of "essentially 100 percent" of all axial welds.

During the review of the Brunswick license renewal application, the NRC evaluated relief from the ASME Code, Section XI, circumferential weld examination requirements for the period of extended operation. Section 4.2.5.3 of the Safety Evaluation Report (SER) (i.e., Reference 7) for the license renewal application states in part:

Approval of the relief request on the RV circumferential weld examinations has only been granted for the current operating terms for BSEP. Therefore, should relief be desired on the applicable RV circumferential weld examinations for the period of extended operation, the applicant must submit a relief request for the period of extended operation once the renewed operating licenses for the BSEP have been issued. However, the technical evaluation section of the relief request may be simplified by referencing the TLAA discussions/calculations in LRA Section 4.2.5 and Table 4.2-7 as the basis for requesting the relief and the staff's approval of the TLAA in SER Section 4.2.5.2. To be consistent with the criteria of GL 98-05, the technical evaluation section of the relief request will also need to address CP&L's procedures and actions for mitigating the probability of a cold, overpressurization event at the facilities.

Technical evaluation of the proposed alternative is provided below. As directed by the NRC, for Condition 1 (i.e., satisfying the limiting conditional failure probability for circumferential welds), the TLAA discussions in license renewal application Section 4.2.5 are referenced and discussed below. For Condition 2 (i.e., procedures and actions for mitigating the probability of a cold, overpressurization event), similar information to that provided in the prior relief request (i.e., Reference 4) as approved by the NRC (i.e., Reference 5) is also provided below.

<u>Condition 1 – Satisfying the Limiting Conditional Failure Probability for Circumferential Welds</u>:

At the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC's July 30, 1998, safety evaluation.

<u>Condition 1 Basis for Use</u>:

During the review of the Brunswick license renewal application, the NRC evaluated relief from the ASME Code, Section XI, circumferential weld examination requirements for the period of extended operation. Section 4.2.5.1 of the SER (i.e., Reference 7) for the license renewal application states in part:

> *The NRC evaluation of BWRVIP-05 utilized a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis are:*
>
> 1. *the neutron fluence was the estimated end-of-life mean fluence*
> 2. *the chemistry values are mean values based on vessel types, and*
> 3. *the potential for beyond-design-basis events is considered.*
>
> *Table 4.2-7 provides a comparison of the BSEP Units 1 and 2 reactor vessel limiting circumferential weld parameters to those used in the NRC evaluation for the first two key assumptions. Data provided in Table 4.2-7 was supplied from Tables 2.6-4 and 2.6-5 of the previously identified, July 28, 1998 NRC Safety Evaluation Report. However, the correction of the Chemistry Factor in the table is from the supplement to the SER issued by the NRC on March 7, 2000.*
>
> *The 54 EFPY fluence values for BSEP are bounded by both the 32 EFPY and 64 EFPY fluence values in the NRC analysis. The BSEP Units 1 and 2 weld materials have lower copper and nickel values than those used in the NRC analysis. Hence, there is a smaller chemistry factor. As a result, the shifts in reference temperature for Units 1 and 2 are lower than both the 32 EFPY and 64 EFPY shift from the NRC SER analysis. The combination of unirradiated reference temperature ($RT_{NDT(u)}$) and shift (RT_{NDT}) yields adjusted reference temperatures for Units 1 and 2 that are considerably lower than the NRC mean analysis values. Therefore, the RPV shell weld embrittlement due to the additional fluence associated with the period of extended operation has a negligible effect on the probabilities of RPV shell weld failure. The Mean RT_{NDT} values for Units 1*

and 2 at 54 EFPY are bounded by the 32 EFPY and the 64 EFPY Mean RT_{NDT} provided by the NRC.

Although a conditional failure probability has not been calculated, the fact that the BSEP values for Mean RT_{NDT} at the end of the 60-year license are less than both the 32 EFPY and 64 EFPY values provided by the NRC leads to the conclusion that the BSEP RPV conditional failure probability is bounded by the NRC analysis. Therefore, the TLAA has been projected through the end of the period of extended operation.

The procedures and training used to limit cold over-pressure events during the period of extended operation will be the same as those approved by the NRC when BSEP requested that the BWRVIP-05 technical alternative be used for the current licensing term for Units 1 and 2.

Disposition: 10 CFR 54.21(c)(1)(ii) - The RPV circumferential weld analyses have been projected to the end of the period of extended operation.

Section 4.2.5.2 of the SER (i.e., Reference 7) for the license renewal application states, in part:

The staff performed an independent calculation of the mean ΔRT_{NDT} values for the limiting BSEP RV circumferential welds through 54 EFPY. Table 4.2.5-1 of this SE provides a summary of the mean RTNDT values calculated by the staff for the BSEP RVs through 54 EFPY and a comparison of the staff's mean RT_{NDT} values to both the corresponding mean RTNDT values calculated by the applicant and the mean RT_{NDT} value criterion for the limiting CB&I case study at 64 EFPY.

The results in SER Table 4.2.5-1, below, demonstrate that the mean RT_{NDT} values calculated by the licensee for the BSEP RV circumferential welds are less than that for the limiting CB&I case study and are in agreement with those calculated by the staff. Based on this analysis, the staff found that the applicant has provided a valid basis for concluding that the conditional probability of failure values for the BSEP RV circumferential welds are sufficiently low to accept the TLM and set the bases for requesting relief to eliminate the RV circumferential weld examinations for the extended period of operation once the operating licenses for BSEP have been renewed. Based on this independent assessment, the staff found that the licensee's TLAA on circumferential weld relief requests conforms to Action No. 11 on topical report BWRVIP-74-A and has been projected to 54 EFPY and is acceptable pursuant to 10 CFR 54.21(c)(1)(ii).

Table 4.2.5-1 Comparison of NRC and CP&L 54 EFPY Mean ΔRT_{NDT} Calculations to the 64 EFPY Mean ΔRT_{NDT} Calculations for the Limiting CB&I Case Study on BWRVIP-05

Parameter Description	***Limiting 64 EFPY CB&I Case Study***	***NRC 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 1 (Note 1)***	***CP&L 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 1 (Note 1)***	***NRC 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 2 (Note 1)***	***CP&L 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 2 (Note 1)***
Alloy%Cu	*0.10*	*0.06*	*0.06*	*0.02*	*0.02*

Parameter Description	***Limiting 64 EFPY CB&I Case Study***	***NRC 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 1 (Note 1)***	***CP&L 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 1 (Note 1)***	***NRC 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 2 (Note 1)***	***CP&L 54 EFPY Mean ΔRT_{NDT} Calculations for Unit 2 (Note 1)***
Alloy% Ni	*0.99*	*0.87*	*0.87*	*0.9*	*0.9*
$RT_{NDT(U)}$ (°F)	*-65*	*-50*	*-50*	*-50*	*-50*
Fluence (10^{19} n/cm^2, E > 1.0 MeV)	*1.02*	*0.324*	*0.324*	*0.322*	*0.322*
Chemistry Factor	*134.9*	*82.0*	*82.0*	*27.0*	*27.0*
ΔRT_{NDT} (°F)	*135.6*	*56.6*	*56.6*	*18.6*	*18.6*
Mean ΔRT_{NDT} (°F)	*70.6*	*6.6*	*6.6*	*-31.4*	*-31.4*
NRC Established Conditional Probability of Failure [P(F/E)] Criterion for Case I Result for Plant Specific Calculation	*1.78×10^5 (Maximum P(F/E) value to justify relief: Refer to Note 2)*	*Mean ΔRT_{NDT} is Lower than Case Study Mean ΔRT_{NDT}; Criterion is met. (Note 2)*	*Mean ΔRT_{NDT} is Lower than Case Study Mean ΔRT_{NDT}: Criterion is met. (Note 2)*	*Mean ΔRT_{NDT} is Lower than Case Study Mean ΔRT_{NDT}: Criterion is met. (Note 2)*	*Mean ΔRT_{NDT} is Lower than Case Study Mean ΔRT_{NDT}: Criterion is met Note 2)*

Notes:

1. *For the BSEP RVs, the limiting circumferential weld materials determined by the staff were equivalent to those determined by CP&L. For Unit 1, the limiting RV circumferential weld is FG, which was fabricated from weld heat No. 1P4218. For Unit 2, the limiting RV circumferential weld is FG, which was fabricated from weld heat No. 3P4000.*
2. *If the plant-specific mean ΔRT_{NDT} is less than the mean ΔRT_{NDT} associated with the limiting case study, the staff concludes that probability of failure for the plant-specific circumferential weld under review will be less than the conditional probability of failure value for the limiting circumferential weld in the limiting case study. BWR plants that meet this criterion may conclude that the probability of failure for the limiting circumferential RV welds is sufficiently low enough to justify elimination of the volumetric examinations required by Section XI of the ASME Code (Examination Category B-A, Item B1.11) and augmented volumetric examinations for the circumferential welds required by 10 CFR 50.55a(g)(6)(ii)(A)(2).*

In Brunswick's Updated Final Safety Analysis Report supplement summary description of its TLAA evaluation of RPV circumferential weld inspection relief in license renewal application Section 4.2.5.3, it states:

> *For the period of extended operation, the 54 EFPY fluence values for BSEP are bounded by both the 32 EFPY and 64 EFPY fluence values in the NRC analysis. The BSEP Units 1 and 2 weld materials have lower copper and nickel values than those used in the NRC analysis. Hence, there is a smaller chemistry factor. As a result, the shifts in reference temperature for Units 1 and 2 are lower than both the 32 EFPY and 64 EFPY shift from the NRC SER analysis. The combination of unirradiated reference temperature ($RT_{NDT(U)}$) and shift (ΔRT_{NDT}) yields adjusted reference temperatures for Units 1 and 2 that are considerably lower than the NRC Mean analysis values.*

Therefore, the RPV shell weld embrittlement due to the additional fluence associated with the period of extended operation has a negligible effect on the probabilities of RPV shell weld failure. The Mean ΔRT_{NDT} values for Units 1 and 2 at 54 EFPY are bounded by the 32 EFPY and the 64 EFPY Mean ΔRT_{NDT} provided by the NRC.

Although a conditional failure probability has not been calculated, the fact that the BSEP values for Mean ΔRT_{NDT} at the end of the 60-year license are less than both the 32 EFPY and 64 EFPY values provided by the NRC leads to the conclusion that the BSEP RPV conditional failure probability is bounded by the NRC analysis. Therefore, the TLAA has been projected through the end of the period of extended operation.

The NRC found Brunswick's UFSAR supplement summary description consistent with the NRC analysis for the TLAA of reactor vessel circumferential weld examination relief in license renewal SER Section 4.2.5.4. Therefore, based on this assessment, the NRC found the UFSAR supplement summary description for the TLAA of the reactor vessel circumferential weld examination relief acceptable.

54 EFPY fluence values for BSEP Units 1 and 2 were updated in November 2012 for the period of extended operation (i.e., Reference 11), and were found to be lower than those given in SER Table 4.2.5-1. Therefore, the neutron fluence values used in the NRC SER are still bounding.

Based on the information presented in this request and the Brunswick license renewal application (i.e., Reference 12) with the NRC SER (i.e., Reference 7), the RPV circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds as stated in the NRC's Safety Evaluation dated July 28, 1998 (i.e., Reference 2).

<u>Condition 2 – Operator Training and Established Procedures that Limit the Frequency of Cold Over-Pressure Events</u>:

Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC's July 30, 1998, safety evaluation.

<u>Condition 2 Basis for Use</u>:

The procedures and training to limit cold over-pressure events are similar as those approved by the NRC in their safety evaluation for the June 21, 2000, relief request (i.e., Reference 4). The NRC's Safety Evaluation (i.e., Reference 5), Section 3.1.2, *Cold Overpressure Transient Probability*, concluded:

The staff concludes that a non-design basis cold overpressure transient is unlikely to occur at Brunswick 1 and 2, and that the information the licensee provided about Brunswick high pressure injection systems, operator training, and plant-specific procedures provides sufficient basis to support approval of the alternative examination request.

The information submitted previously in Reference 4 is provided below and remains applicable to BSEP, Units 1 and 2.

In Generic Letter 98-05 (i.e., Reference 3), the NRC stated that beyond design-basis events occurring during plant shutdown could lead to cold over-pressure events that could challenge reactor pressure vessel integrity. The NRC noted that the industry's response concluded that condensate and control rod drive pumps could cause conditions that could lead to cold over-pressure events that could challenge reactor pressure vessel integrity.

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. The following discussion provides further information to support Duke Energy's conclusion.

Operator Training to Prevent Reactor Pressure Vessel Cold Over-Pressure Events

Periodic operator training reduces the possibility of a low-temperature over-pressure event occurring. Training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves is provided. In addition, periodic operator training reinforces management's expectations for strict procedural compliance.

Duke Energy continuously reviews industry operating experience to ensure BSEP procedures consider the impact of actual events, including low-temperature over-pressure events. Appropriate changes to procedures and training are then implemented to preclude similar situations from occurring at BSEP, Units 1 and 2.

Procedural Controls to Prevent Reactor Pressure Vessel Cold Over-Pressure Events

BSEP has procedures in place which monitor and control reactor water level, pressure, and temperature during cold shutdown and refueling operations. These procedures minimize the likelihood of a low temperature over-pressure event from occurring. These procedures are reinforced through normal, periodic operator training.

During normal cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with operating procedures. The plant procedure for unit shutdown limits reactor pressure to less than or equal to 40 psig while flooding up to cold shutdown water level and requires frequent monitoring of reactor pressure to ensure that this limit is not exceeded.

The Operations procedure governing Control Room activities requires that operators frequently monitor for indications and alarms, to detect abnormalities as early as possible, and immediately notify the Senior Reactor Operator (SRO) of any changes or abnormalities in indications.

Furthermore, this procedure requires that changes which could affect reactor water level, pressure, or temperature, be performed only under the knowledge and direction of the SRO. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. Finally, the status of plant conditions, any on-going activities which could affect critical plant parameters, and contingency planning are discussed by operators at each shift turnover. This ensures that on-coming operators are cognizant of any activities which could adversely affect reactor water level, pressure, or temperature.

Inadequate work management is a potential contributor to a cold over-pressurization event. At BSEP, work performed during outages is scheduled by the Outage Management group. Dedicated SROs provide oversight of outage schedule development to avoid conditions which could adversely impact reactor water level, pressure, or temperature. From the outage schedule, a plan-of-the-day (POD) is developed listing the work activities to be performed. These PODs are reviewed and approved by management, and a copy is maintained in the Control Room. Changes to the PODs require management review and approval. Additionally, the detailed outage schedule receives a thorough shutdown risk assessment review to ensure defense-in-depth is maintained.

During outages, work is coordinated through the Outage Command Center, which provides an additional level of Operations oversight. In the Control Room, the SRO is required to maintain cognizance of any activity which could potentially affect reactor level or decay heat removal during refueling outages. The Control Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Pre-job briefings are conducted for work activities that have the potential of affecting critical reactor parameters. These briefings are attended by the cognizant individuals involved in the work activity. Expected plant responses and contingency actions to address unexpected conditions or responses that may be encountered are included in the briefing discussion.

Procedural controls for reactor temperature, level, and pressure are an integral part of operator training. Specifically, operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits.

Review of High Pressure Injection Sources

With regard to inadvertent system injection in a low-temperature condition, the high pressure make-up systems, the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, as well as the normal feedwater (FW) supply by the reactor feedwater pumps, are all steam driven. During reactor cold shutdown conditions, no reactor steam is available for operation of these systems. Therefore, it is not possible for these systems to contribute to an over-pressure event while a BSEP unit is in cold shutdown.

During cold shutdown conditions, reactor pressure vessel level and pressure are controlled with the Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The reactor is not taken solid during these times except for performance of hydrostatic testing. If either of these systems were to fail, operators would adjust the other system to control level. Under these conditions, the CRD system typically injects water into the reactor at a rate of less than 100 gpm. This slow injection rate 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 pressure-temperature limits.

The Standby Liquid Control (SLC) system is another high pressure water source to the reactor pressure vessel. However, there are no automatic starts associated with this system. SLC injection requires an operator to manually start the SLC pumps from the Control Room via a keylock switch. Additionally, the injection rate of one SLC pump is approximately 43 gpm; the injection rate of two SLC pumps is approximately 86 gpm. These flow rates give the operator ample time to control reactor pressure in the case of an inadvertent injection.

<u>Review of Low Pressure Injection Sources</u>

The condensate booster pumps are capable of injecting water at up to approximately 400 psig. Following 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. 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.

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. Therefore, the potential for an over-pressure event which would exceed the pressure/temperature limits, due to the inadvertent actuation of these systems, is very low.

Two precursor events are identified for BSEP, Unit 1 in Table C-1 of the NRC's Safety Evaluation for the BWRVIP-05 report. Both of these events involved inadvertent injection of low pressure makeup systems. The first event resulted in an injection of the Low Pressure Coolant Injection and Core Spray systems, and the second event resulted in an injection of the Core Spray system. Neither of these events resulted in a violation of the pressure-temperature limits.

Based on the above, the probability of a cold over-pressure event occurring at BSEP, Units 1 and 2, is considered to be less than or equal to the probability used in the analysis described in the NRC independent evaluation performed in the assessment of the BWRVIP-05 report.

<u>Inspection of Axial Welds</u>

The axial weld seams (i.e., Examination Category B-A, Item No. B1.12) and their intersection with the associated circumferential weld seams continue to be examined in accordance with ASME Code 2001 Edition, 2003 Addenda Section XI Table IWB-2500-1 requirements.

Unit 1 axial weld seams were last examined in the March 2008 refueling outage, with no unacceptable indications. The fourth 10-year interval examinations are scheduled for the March 2018 refueling outage. Unit 2 axial weld seams were last examined in the March 2017 refueling outage, with no unacceptable indications.

7. Duration of the Proposed Alternative

The duration of this proposed alternative is for the period of extended operation ending September 8, 2036, for BSEP, Unit 1, and December 27, 2034, for BSEP, Unit 2.

8. Precedents

The similar proposed alternative was previously approved for BSEP as a permanent relief (that is, for the remaining term of operation under the original license) by the NRC during the second 10-year inservice inspection interval (i.e., Reference 5).

Based on issuance of renewed facility operating licenses, similar requests have been approved for the James A. Fitzpatrick Nuclear Power Plant (i.e., Reference 8); Peach Bottom Atomic Power Station, Units 1 and 2 (i.e., Reference 9); and Susquehanna Steam Electric Station, Units 1 and 2 (i.e., Reference 10).

9. References

1. EPRI Report TR-105697, *BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),* dated September 29, 1995, ADAMS Accession Number ML032200246 **(EPRI Proprietary)**

2. Letter from Mr. G. C. Lainas (NRC) to Mr. C. Terry (BWRVIP Chairman), *Evaluation by the Office of Nuclear Reactor Regulation Related to the Review of the Topical Report by the Boiling Water Reactor Vessel and Internals Project: BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation (BWRVIP-05)*, dated July 28, 1998.

3. 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,* dated November 10, 1998.

4. Letter from Warren J. Dorman (Carolina Power & Light Company) to U.S. Nuclear Regulatory Commission Document Control Desk, *Volumetric Examination of Circumferential Reactor Pressure Vessel Welds*, dated June 21, 2000.

5. Letter from Richard P. Correia (USNRC) to J. S. Keenan (Carolina Power & Light Company), *Safety Evaluation for Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i) for Reactor Pressure Vessel Circumferential Shell Weld Examinations (TAC Nos. MA9299 and MA9300)*, dated September 14, 2000.

6. Letter from Maurice Heath (USNRC) to James Scarola (Carolina Power & Light Company), *Issuance of Renewed Facility Operating License Nos. DPR-71 and DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. MC4639 and 4640)*, dated June 26, 2006, ADAMS Accession Number ML061660358.

7. NUREG-1856, *Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2*, June 2006.

8. Letter from Stephen S. Koenick (NRC) to Brian Sullivan (Entergy Nuclear Operations, Inc.), *James A. Fitzpatrick Nuclear Power Plant – Relief Request for Proposed Alternative for the Implementation of BWRVIP-05 (CAC No. MF8361)*, dated January 6, 2017, ADAMS Accession Number ML16355A429.

9. Letter from Meena Khanna (NRC) to Michael J. Pacilio (Exelon Generation Company, LLC), *Peach Bottom Atomic Power Station, Units 2 and 3 – Requests for Relief I4R-51 and I4R-52 (TAC Nos. ME5392, ME5393, ME5394 and ME5395)*, dated January 24, 2012, ADAMS Accession Number ML112770217.

10. Letter from Meena Khanna (NRC) to Timothy S. Rausch (PPL Susquehanna, LLC), *Susquehanna Steam Electric Station, Units 1 and 2 – Relief Requests for the Fourth 10-Year Inservice Inspection Interval (TAC Nos. MF2705 and MF2714)*, dated June 9, 2014, ADAMS Accession Number ML14141A073.

11. Westinghouse Report, WCAP-17660-NP, Revision 0, *Neutron Exposure Evaluations for Core Shroud and Pressure Vessel Brunswick Units 1 and 2*, November 2012.

12. Letter from Cornelius J. Gannon to U.S. Nuclear Regulatory Commission Document Control Desk, *Application for Renewal of Operating Licenses*, dated October 18, 2004.