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10 CFR 50.55a(a)(3)(i)
10 CFR 50.55a(g)(6)(ii)(A)(5)

U. S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
VOLUMETRIC EXAMINATION OF CIRCUMFERENTIAL REACTOR PRESSURE VESSEL WELDS

Gentlemen:

On November 10, 1998, the NRC issued 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." This generic letter states that the NRC has completed review of the report entitled "BWR Vessel and Internals Project [BWRVIP], BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." The generic letter further states that boiling water reactor (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 states that licensees still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Carolina Power & Light (CP&L) Company requests approval of an alternative reactor pressure vessel examination for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. Approval of this alternative examination is requested in accordance with 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(g)(6)(ii)(A)(5) for permanently excluding volumetric examination of circumferential reactor pressure vessel welds. The alternative is consistent with the guidance

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contained in Generic Letter 98-05, and CP&L has addressed the two criteria specified by Generic Letter 98-05 in the enclosure of this letter.

CP&L also requests approval of the alternative reactor pressure vessel examination in lieu of the inservice inspection requirements for circumferential welds in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The Code of record for the third ten-year inservice inspection interval is the ASME Code, Section XI, 1989 Edition (no addenda).

BSEP, Unit No. 1 Refueling Outage 13 (i.e., B114R1) is scheduled to begin in March 2002. BSEP, Unit No. 2 Refueling Outage 14 (i.e., B215R1) is scheduled to begin in February 2001. Unless the NRC approves permanent relief from the requirement to perform reactor pressure vessel circumferential weld examinations, these examinations will be required during the refueling outages cited above. In order to support refueling outage activities, approval of this request is needed by February 2001.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing at (910) 457-2073.

Sincerely,

Warren J. Dorman
Manager – Regulatory Affairs
Brunswick Steam Electric Plant

WRM/wrm

Enclosure: Response to Generic Letter 98-05 Criteria

cc (with enclosure):

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ENCLOSURE

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
VOLUMETRIC EXAMINATION OF CIRCUMFERENTIAL
REACTOR PRESSURE VESSEL WELDS

Response to Generic Letter 98-05 Criteria

Background

On August 6, 1992, the NRC published, in the *Federal Register*, an amendment to 10 CFR 50.55a(g) (57 FR 34666). The new regulation revoked previously granted licensee relief requests regarding the extent of volumetric examination of reactor pressure vessel shell welds specified in the American Society of Mechanical Engineers (ASME) Code, Section XI, "Rules For Inservice Inspection of Nuclear Power Plant Components," Table IWB-2500-1, Examination Category B-A, Item B1.10. The new regulation requires volumetric examination of reactor pressure vessel shell welds be performed completely, once, as an augmented examination requirement. These examinations are required to be performed using the 1989 Edition of the ASME Code, Section XI. These examinations are required during the inspection interval when the regulation was approved or the first period of the next inspection interval. For purposes of the augmented examinations, the regulation defined "essentially 100 percent" as more than 90 percent of the examination volume of each weld.

The Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted report BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," dated September 1995. This report evaluated the current inspection criteria for boiling water reactor (BWR) reactor pressure vessel welds, provided recommendations for alternative inspection requirements, and provided a technical basis for these recommended requirements.

On May 12, 1997, the NRC and members of the BWRVIP met to discuss the NRC's review of the BWRVIP-05 report. Based on the guidance provided in NRC Staff Requirements Memorandum M970512B, dated May 30, 1997, the NRC initiated a broader, risk-informed review of the proposal in the BWRVIP-05 report.

Subsequently, in Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05," the NRC indicated that it would consider technically justified alternatives to the augmented examination, in accordance with 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), and 10 CFR 50.55a(g)(6)(ii)(A)(5), from BWR licensees who were scheduled to perform inspections of the reactor pressure vessel shell circumferential welds during the Fall 1997 or Spring 1998 outage periods. The Information Notice indicated that acceptably justified alternatives would be

considered for inspection delays of up to 40 months, or two operating cycles, whichever is longer, for reactor pressure vessel circumferential shell welds only.

On September 18, 1997, the NRC issued an approval of Carolina Power & Light (CP&L) Company's request for Brunswick Steam Electric Plant (BSEP), Unit No. 2 to use an alternative reactor pressure vessel weld examination in accordance with the requirements of 10 CFR 50.55a(g)(6)(ii)(A)(5). On April 21, 1998, the NRC issued an approval of a similar CP&L request for BSEP, Unit No. 1. In these approvals, the NRC authorized a delay, for up to two operating cycles, of the reactor pressure vessel weld examinations.

On November 10, 1998, the NRC issued 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." The generic letter states that the NRC has completed review of the "BWR Vessel and Internals Project [BWRVIP], BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)" report. The generic letter further states 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 states that licensees still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Requested Action

CP&L requests approval of an alternative reactor pressure vessel examination for the BSEP, Unit Nos. 1 and 2. Approval of this alternative examination is requested in accordance with 10 CFR 50.55a(a)(3)(i) and and 10 CFR 50.55a(g)(6)(ii)(A)(5) for permanently excluding volumetric examination of circumferential reactor pressure vessel welds. CP&L also requests approval of the alternative reactor pressure vessel examination in lieu of the inservice inspection requirements for circumferential welds in the ASME Code, Section XI, 1989 Edition (no Addenda), Table IWB-2500-1, Examination Category B-A, Item No. B1.11, volumetric examination of reactor pressure vessel circumferential welds. The code of record for the third ten-year inservice inspection interval is the ASME Code, Section XI, 1989 Edition (no addenda).

Proposed Alternative

CP&L proposes to perform inspections of essentially 100 percent of the longitudinal seam welds in the reactor pressure vessel shell and essentially zero percent of the reactor pressure vessel circumferential seam welds, which will result in partial examination (i.e., approximately two to three percent) of the circumferential welds at their points of intersection with the longitudinal welds. These inspections are being proposed as an alternative to the augmented examinations specified in 10 CFR 50.55a(g)(6)(ii)(A)(2) for circumferential welds, as well as an alternative to the inservice inspection requirements for circumferential welds in the ASME Code, Section XI, 1989 Edition (no Addenda).

Regulatory Requirements

The requirement for inservice inspections, which includes reactor pressure vessel circumferential weld inspection, derives from 10 CFR 50.55a. 10 CFR 50.55a requires inservice inspection and testing of the Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code, and applicable addenda, as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC in accordance with 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) allows alternatives to the requirements of paragraph (g) to be used, when authorized by the NRC, if (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In accordance with 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components must meet the requirements, except the design and access provisions and the preservice examination requirements, 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 the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 10-year interval, subject to the limitations and modifications listed therein. The components, including supports, may meet the requirements in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed, and subject to NRC approval.

The ASME Code, Section XI, 1989 Edition, Examination Category B-A, Item No. B1.11, circumferential reactor pressure vessel shell welds, requires a volumetric examination of essentially 100 percent of the weld length of all circumferential welds during the 10-year inservice inspection interval. Performance of these examinations is required in accordance with the ASME Code, Section XI, Examination Requirements/Figure No. IWB-2500-1 and the non-destructive examination requirements of the ASME Code, Section V, Article 4, paragraph T-441.3.2. The ASME Code requirements are supplemented by Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

Technical Basis For Proposed Alternative

The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds; however, the following information is provided to demonstrate the conservatism of the NRC analysis relative to the reactor pressure vessels for BSEP, Unit Nos. 1 and 2.

The BWRVIP-05 report was transmitted to the NRC in September 1995. The NRC's safety evaluation for the BWRVIP-05 submittal was transmitted to Carl Terry, Chairman of the BWRVIP, in a letter dated July 30, 1998. On March 7, 2000, the NRC issued a supplement to the Final Safety Evaluation of the BWRVIP-05 report. In this supplement, the NRC concludes that the reactor pressure vessel failure frequency due to failure of the limiting axial welds in the boiling water reactor fleet is below 5×10^{-6} per reactor-year, consistent with regulatory Guide 1.154, given the assumptions described in the NRC's supplemental safety evaluation. Therefore, both the NRC and BWRVIP evaluations support a conclusion that the failure probability of BWR reactor pressure vessel circumferential welds is orders of magnitude lower than that of the axial welds.

In Generic Letter 98-05, the NRC stated that the estimated failure frequency of the BWR reactor pressure vessel circumferential welds is well below the acceptable core damage frequency (CDF) and large early release frequency (LERF) criteria discussed in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." Furthermore, the NRC indicated that the estimated frequency of reactor pressure vessel circumferential weld failure bounds the corresponding CDF and LERF that may result from a reactor pressure vessel weld failure. On this basis, the NRC concluded that the proposal in the BWRVIP-05 report, as modified by two criteria, was acceptable and that 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 two criteria discussed below. The generic letter states that licensees still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Generic Letter 98-05, Criterion 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.

CP&L Response

The NRC evaluation of the BWRVIP-05 report uses the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate reactor pressure vessel failure probabilities. Three key assumptions in the PFM analysis are:

1. the neutron fluence was that estimated to be end-of-license (EOL) mean fluence;
2. the chemistry values are mean values based on vessel types; and
3. the potential for beyond design basis events is considered.

The following information is provided to demonstrate the conservatism of the NRC analysis for BSEP, Unit Nos. 1 and 2. Changes in RT_{NDT}, the nil ductility reference temperature, may be used as one of the means for monitoring the amount of irradiation embrittlement. In the case of BSEP, Unit Nos. 1 and 2, the single circumferential weld joint located within the reactor pressure vessel beltline would be the limiting circumferential weld within the vessel (i.e., relative to RT_{NDT}). For plants with reactor pressure vessels fabricated by Chicago Bridge & Iron (CB&I), the mean end-of-license neutron fluence used in the NRC PFM analysis was 0.51E+19 n/cm^2. The highest fluence values anticipated at the end-of-license is 0.1150E+19 n/cm^2 for BSEP, Unit No. 1 and 0.1302E+19 n/cm^2 for BSEP, Unit No. 2. These end-of-license fluence values were conservatively estimated for 32 effective full power years (EFPY) of operation. Thus, the fluence effect on embrittlement is lower for BSEP, Unit Nos. 1 and 2, than described in the NRC analysis and significant conservatism exists in the already low circumferential weld failure probabilities as related to BSEP, Unit Nos. 1 and 2.

As shown in Table 1, the calculated embrittlement shift in RT_{NDT} (i.e., ΔRT_{NDT}) at the end-of-license is 36.5°F for the BSEP, Unit No. 1 vessel and 12.7°F for the BSEP, Unit No. 2 vessel. By comparison, Table 2.6-4 of the NRC's Final Safety Evaluation of the BWRVIP-05 Report indicates an embrittlement shift of 109.5°F for CB&I fabricated vessels. Therefore, the calculated ΔRT_{NDT} values for the BSEP, Unit Nos. 1 and 2 vessels are less than, and thus bounded by, the embrittlement shift assumed in the NRC's safety evaluations for the BWRVIP-05 report.

Furthermore, as seen in the attached Table 1, the calculated Upper Bound RT_{NDT} values for beltline welds at end-of-license are 83.0°F and 35.5°F for BSEP, Unit Nos. 1 and 2, respectively. These results are based on the initial RT_{NDT} values for these weld joints provided in CP&L's letter dated November 16, 1995 (Serial: BSEP 95-0572). Using the 10 CFR 50.61 generic value of -56°F for initial RT_{NDT} yields end-of-license Upper Bound RT_{NDT} values of 30.4°F for BSEP, Unit No. 1 and -7.0°F for BSEP, Unit No. 2.

For comparison, the calculated Upper Bound RT_{NDT} for circumferential welds in CB&I fabricated vessels utilizing the information in Table 2.6-4 of the NRC's Final Safety Evaluation for the BWRVIP-05 report is 100.5°F. Thus, the calculated Upper Bound RT_{NDT} values for the BSEP, Unit Nos. 1 and 2 reactor pressure vessel circumferential welds are clearly bounded by the calculated limiting RT_{NDT} for CB&I vessels, thus providing additional assurance that the BSEP, Unit Nos. 1 and 2 vessel welds are also bounded by the BWRVIP-05 report and NRC plant specific analyses.

Generic Letter 98-05, Criterion 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.

CP&L Response

In Generic Letter 98-05, 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. However, 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.

For a boiling water reactor to experience such an event, the plant would require several operator errors to occur. The NRC's assessment described several types of events that could be precursors to BWR pressure vessel cold over-pressure transients. At the meeting of August 8, 1997, the NRC described several types of events that could be precursors to BWR reactor pressure vessel cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U.S. BWR. Also, at the August 8 meeting, the NRC identified one actual cold over-pressure event that occurred during shutdown at a non-U. S. BWR. This event apparently included several operational errors that resulted in a maximum reactor pressure vessel pressure of 1150 psi with a temperature range of 79°F to 88°F.

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 CP&L'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. Training material pertaining to the RPV has been revised to include a discussion of NRC Generic Letter 98-05 in order to further raise awareness of the potential for cold over-pressurization. In addition, periodic operator training reinforces management's expectations for strict procedural compliance.

CP&L 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, Unit Nos. 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 10 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.

A review of industry operating experience indicates that 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 Work Control 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 and Reactor Core Isolation Cooling systems, as well as the normal feedwater 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. 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 60 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 pump from the Control Room via a keylock switch. Additionally, the injection rate of one SLC pump is approximately 41 gpm; the injection rate of two SLC pumps is approximately 82 gpm. These flow rates give the operator ample time to control reactor pressure in the case of an inadvertent injection.

Review of Low Pressure Injection Sources:

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 No. 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, Unit Nos. 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.

Summary

The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR reactor pressure vessel circumferential shell welds is orders of magnitude lower than that of the axial shell welds. Based on an assessment of the materials in the circumferential weld in the beltline of the BSEP reactor pressure vessels, the conditional probability of reactor pressure vessel failure should be less than or equal to that estimated in the NRC's analysis. Based on operator training and established procedures that have been implemented, the probability of cold over-pressure transients will limit the frequency of cold over-pressure events to the amounts specified in the NRC's June 30, 1998, safety evaluation.

TABLE 1

BSEP RPV Shell Weld Information

	BSEP, Unit No. 1	BSEP, Unit No. 2	USNRC Limiting Plant Specific Analyses Parameters at 32 EFPY, Safety Evaluation Table 2.6-4
Neutron fluence at the end-of-license [1]	0.1150E+19 n/cm^2	0.1302E+19 n/cm^2	0.51 E+19 n/cm^2
Initial (unirradiated) reference temperature • CP&L values estimated by Method 4 of MTEB 5-2 and submitted in CP&L's Response dated November 16, 1995, to Generic Letter 92-01 • NRC generic value	 +10°F (Estimated Value) -56°F (Generic Value)	 +10°F (Estimated Value) -56°F (Generic Value)	-65°F
Weld chemistry factor (CF)	82.0 °F	27.0 °F	134.9°F [3]
Weld copper content	0.06 %	0.02 %	0.10 %
Weld nickel content	0.87 %	0.90 %	0.99 %
Increase in reference temperature due to irradiation (ΔRT_{NDT})	36.5°F	12.7°F	109.5°F
Margin term	36.5°F [2] 49.9°F [3]	12.7°F [2] 36.3°F [3]	56°F [4]
Mean adjusted reference temperature (Mean ART)	46.5°F [2] -19.5°F [3]	22.7°F [2] -43.3°F [3]	44.5°F
Upper bound adjusted reference temperature (ART)	83.0°F [2] 30.4°F [3]	35.5°F [2] -7.0°F [3]	100.5°F [4]

Notes:
1. The end-of-license fluence was conservatively projected to 32 EFPY.
2. Value based on usage of an initial RT_{NDT} of +10°F (estimated using MTEB 5-2, Estimation Method 4).
3. Value based on usage of an initial RT_{NDT} of -56°F (generic value).
4. This value not shown in Table 2.6-4 of NRC Safety Evaluation; value was calculated from other values taken from table.