September 14, 2000

Mr. J. S. Keenan, Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2 - SAFETY EVALUATION FOR PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(i) FOR REACTOR VESSEL CIRCUMFERENTIAL SHELL WELD EXAMINATIONS (TAC NOS. MA9299 and MA9300)

Dear Mr. Keenan:

By letter dated June 21, 2000, you requested that the NRC approve an alternative to performing examinations of the reactor pressure vessel (RPV) circumferential shell welds at the Brunswick Steam Electric Plant, Units 1 and 2. These examinations are required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and by the augmented examination requirements of Section 50.55a(g)(6)(ii)(A)(2) to Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a(g)(6)(ii)(A)(2)). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and is consistent with the guidance provided in Generic Letter (GL) 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, and the staff evaluation of the BWRVIP-05 report issued July 28, 1998.

The staff has determined that you have provided an acceptable demonstration that the appropriate criteria in GL 98-05 and the staff's evaluation of the BWRVIP-05 report have been satisfied regarding permanent relief (that is, for the remaining term of operation under the existing license) from inservice inspection requirements for the volumetric examination of RPV circumferential welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11).

The NRC staff finds that authorization of your alternative examination would provide assurance of structural integrity and, therefore, an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), your proposed alternative examination is authorized. The staff's Safety Evaluation is enclosed.

Please contact Allen Hansen at (301) 415-1390 if you have any questions regarding this matter.

Sincerely,

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management

Docket Nos. 50-324 and 50-325

Enclosure: As stated

cc w/encl: See next page

Please contact Allen Hansen at (301) 415-1390 if you have any questions regarding this matter.

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Richard P. Correia, Chief, Section 2
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Docket Nos. 50-324 and 50-325

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cc w/encl: See next page
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVES FOR EXAMINATION OF REACTOR PRESSURE VESSEL SHELL WELDS

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

CAROLINA POWER & LIGHT COMPANY

DOCKET NUMBERS 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated June 21, 2000, Carolina Power and Light Company (CP&L), the licensee for the Brunswick Steam Electric Plant, Units 1 and 2, requested that NRC approve an alternative to performing examinations of the reactor pressure vessel (RPV) circumferential shell welds (Reference 1). These examinations are required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and by the augmented examination requirements of Section 50.55a(g)(6)(ii)(A)(2) to Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a(g)(6)(ii)(A)(2)). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(g)(6)(ii)(A)(5) and is consistent with the guidance provided in Generic Letter (GL) 98-05, "Boiling Water Reactor [BWR] 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 (Reference 2), and the staff evaluation of the BWRVIP-05 report issued July 28, 1998 (Reference 3).

1.1 Regulatory Requirements

Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class I, 2, and 3 components are to meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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 120-month interval, subject to the limitations and modifications listed therein.

10 CFR 50.55a(g)(6)(ii)(A) requires that licensees perform an augmented RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code. The final Rule was published in the *Federal Register* on August 6, 1992 (57 FR 34666). By incorporating into

ENCLOSURE

the regulations the 1989 Edition of the ASME Code, the NRC staff required that licensees perform volumetric examinations of "essentially 100 percent" of the RPV pressure-retaining shell weld, during all inspection intervals. 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

1.2 BWRVIP-05 Report

By letter dated September 28, 1995, as modified and supplemented by letters dated June 24, and October 29, 1996, and May 16, June 4, June 13, and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the proprietary report BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations" (Reference 4). As modified, the BWRVIP report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of essentially 100 percent of the axial (that is, longitudinal) welds and essentially none of the circumferential RPV shell welds, except at the intersection of the axial and circumferential welds, thereby including approximately 2-3 percent of the circumferential welds. In addition, the report provided proposals to revise ASME Code requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420(b) of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued a Safety Evaluation (SE) of the BWRVIP-05 report (Reference 3). This evaluation concluded that the failure frequency of RPV circumferential welds in BWRs was sufficiently low to justify elimination of inservice inspection (ISI) of these welds. In addition, the evaluation concluded that BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation indicated that examination of the circumferential welds shall be performed if axial weld examinations reveal an active, mechanistic mode of degradation.

In the BWRVIP-05 report (Reference 4), BWRVIP concluded that the conditional probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. As a part of its review of the report, NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The staff assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential welds during the current 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant, as indicated in Tables 2.6-4 and 2.6-5, respectively (Reference 3). The failure frequency for an RPV is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The staff determined the conditional probability of failure for longitudinal and circumferential welds in BWR vessels fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified a cold overpressure event in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The staff estimated that the probability for the occurrence of the limiting overpressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the staff's evaluation (Reference 3)

identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

1.3 Generic Letter 98-05

On November 10, 1998, the NRC issued GL 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on RPV Shell Welds." GL 98-05 stated that BWR licensees may request permanent (that is, for the remaining term of operation under the existing license) relief from ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV welds (ASME Code Section XI, Table IWB-2500-I, Examination Category B-A, Item 1.11, "Circumferential Shell Welds"), upon demonstrating that:

(1) the limiting conditional failure probability for circumferential welds satisfies the values specified in the NRC staff's July 28, 1998, SE; and

(2) licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998, SE.

Licensees would still need to perform the required inspections of "essentially 100 percent" of all axial welds.

2.0 INFORMATION PROVIDED BY LICENSEE

2.1 Code Requirements for Which Relief is Sought

The licensee identified the following Code requirements from which relief is sought:

ASME Section XI, 1989 Edition (no addenda), Table IWB-2500-1, Examination Category B-A, Item No. B1.11, volumetric examination of RPV circumferential welds. Permanent relief (that is, for the remaining term of operation under the existing license) is requested.

2.1.1 Components for Which Relief is Sought

The requested permanent relief from the Table IWB-2500-1 requirements applies to ISI Class 1, Code Category B-A, "Pressure Retaining Welds in Reactor Vessel," Item B1.11, "Circumferential Shell Welds."

2.2 Licensee's Basis for Relief

The licensee's request is based upon provisions in the NRC SE of the BWRVIP-05 report (Reference 3) and the guidance outlined in GL 98-05 (Reference 2). These documents provide the basis for the elimination of ISIs of BWR RPV circumferential shell welds. The NRC SE of the BWRVIP-05 report evaluated the conditional failure probability of circumferential welds for the limiting plant-specific case of BWR RPVs manufactured by different vendors, including CB&I, using the highest mean irradiated RT_{NDT} to determine the limiting case.

Since the Brunswick RPVs were fabricated by CB&I, the licensee compared the mean irradiated RT_{NDT} to that for the limiting CB&I case described in Table 2.6-4 of Reference 3. For Brunswick, the single circumferential weld joint in the RPV beltline is the limiting circumferential weld within the vessel (that is, relative to RT_{NDT}). For RPVs fabricated by CB&I, the mean end-of-license neutron fluence used in the NRC analysis was 0.51E+19 n/cm^2. The highest neutron fluence values anticipated at the end-of-license are 0.115E+19 n/cm^2 for Brunswick, Unit No. 1, and 0.130E+19 n/cm^2 for Brunswick, Unit No. 2. The licensee estimated the end-of-license neutron fluence values for 32 effective full-power years of operation.

The licensee calculated the embrittlement shift in RT_{NDT} (that is, ΔRT_{NDT}) at the end-of-license to be 36.5°F for the Unit No. 1 vessel and 12.7°F for the 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 109.5ºF for the limiting case for CB&I fabricated vessels. The calculated ΔRT_{NDT} values for the Brunswick vessels are lower than for the limiting case of the NRC analysis, and thus bounded by the embrittlement shift assumed in the NRC's SE of the BWRVIP-05 report.

The licensee calculated the mean adjusted reference temperatures (Mean ARTs) using both an initial unirradiated reference temperature of 10°F, which it estimated using Method 4 of MTEB 5-2, and a temperature of -56°F, which is the generic value specified in 10 CFR 50.61 for Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes. Using an unirradiated reference temperature of 10°F, the licensee calculated the mean ARTs at the end-of-license to be 46.5°F and 22.7°F for Units 1 and 2 respectively. Using the 10 CFR 50.61 generic value of -56°F for the initial RT_{NDT}, the licensee calculated end-of-license values of -19.5°F and -43.3°F for Units 1 and 2, respectively. The mean ARTs are the values the staff used in determining the conditional failure probabilities in Section 3.1 of this SE.

To satisfy the second condition of GL 98-05 for relief consideration, the licensee described operator training, procedural controls, and high- and low-pressure injection sources. The licensee provides periodic operator training to reduce the possibility of a low-temperature overpressure (LTOP) event. It provides training on brittle fracture limits and compliance with the technical specification pressure-temperature limit curves. It revised training material to include a discussion of GL 98-05 to further raise awareness of the potential for cold overpressurization. CP&L continuously reviews industry operating experience to ensure that Brunswick procedures consider the impact of LTOP events. It then makes appropriate changes to procedures and implements training to preclude similar situations at Brunswick. CP&L has procedures in place to monitor and control reactor water level, pressure and temperature during cold shutdown and refueling operations. These procedures minimize the likelihood of an LTOP 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 licensee reviewed high pressure injection sources. It stated that, for inadvertent system injection in a low-temperature condition, the High Pressure Make-up, 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. Since no reactor steam is available to operate these systems, they could not cause an overpressure event while a Brunswick unit is in cold shutdown.

During cold shutdown, RPV level and pressure are controlled with the Control Rod Drive (CRD) and reactor water cleanup 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. Additionally, the injection rate of one SLC pump is approximately 41 gpm and 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.

The licensee reviewed low pressure injection sources and found the potential for an overpressure event that would exceed the pressure/temperature limits very low due to the inadvertent actuation of the relevant systems. The condensate booster pumps can inject water at up to about 400 psig. 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. 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 system pumps have shutoff heads of approximately 313 psig and 250 psig, respectively. The Brunswick pressure-temperature limit curves for hydrostatic testing allow pressures up to 313 psig at a temperature of 70°F.

Based on the above, the licensee considers the probability of a cold overpressure event occurring at Brunswick, Unit Nos. 1 and 2, to be less than or equal to the probability used in the analysis described in the NRC evaluation performed in the assessment of the BWRVIP-05 report.

2.3 Licensee's Proposed Alternative Examination

10 CFR 50.55a(a)(3) allows licensees to propose alternatives to the requirements of 10 CFR 50.55a(g). The licensee proposed, as an alterative, to perform 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 licensee would permanently defer examination of the circumferential welds until expiration of the plant’s current operating license.

3.0 STAFF EVALUATION

The staff's review focused on confirming that the licensee has adequately documented that the conditions for relief outlined in the SE to the BWRVIP-05 report and GL 98-05 are satisfied.

3.1 Relaxation From Circumferential Weld Examination Requirements

3.1.1 Circumferential Weld Conditional Failure Probability

The staff SE (Reference 3) provides a limiting conditional failure probability of 2×10^{-7} per reactor year for a limiting plant-specific mean RT_{NDT} of 44.5°F for CB&I-fabricated RPVs. However, it turns out that, inadvertently, the staff did not identify the Brunswick Unit 1 RPV as the limiting plant for CB&I-fabricated RPVs in its SE. To validate the acceptability of the failure frequency for the Brunswick Unit 1 RPV, the staff performed calculations using the Brunswick Unit 1 value of 46.5°F, the highest mean ART for both units at end-of-license, according to the method described in its SE. The calculations showed that the failure frequency for the Brunswick Unit 1 RPV is 6×10^{-10} per reactor year. This frequency is more than a factor of 100 less than those for the limiting CE and B&W RPVs, 2.8×10^{-8} and 8.2×10^{-8} per reactor year, respectively (based on an event frequency of 10^{-3} per reactor-year). Since the Brunswick Unit 2 RPV has a mean RT_{NDT} at end-of-license of 22.7°F, the failure frequency for Unit 2 will likewise be bounded by the results of the SE on the BWRVIP-05 report. Therefore, this finding demonstrates that the circumferential welds of the RPVs of the Brunswick units will continue to satisfy the limiting conditional failure probability in the staff's SE on the BWRVIP-05 report, consistent with the requirements of GL 98-05.

3.1.2 Cold Overpressure Transient Probability

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 a sufficient basis to support approval of the alternative examination request.

4.0 CONCLUSIONS

The staff has reviewed the licensee's submittal and finds that the licensee has provided an acceptable demonstration that the appropriate criteria in GL 98-05 and the staff's evaluation of the BWRVIP-05 report have been satisfied regarding permanent relief (that is, for the remaining term of operation under the existing license) from ISI requirements for the volumetric examination of RPV circumferential welds, ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

The NRC staff concludes that authorization of the licensee's alternative examinations would provide assurance of structural integrity and, therefore, an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination is authorized.

5.0 REFERENCES

1. Letter to U.S. NRC Document Control Desk from W. Dorman (CP&L), “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," dated June 21, 2000.

2. GL 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.

3. “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 Recommendations (BWRVIP-05),” dated July 28, 1998.

4. EPRI Technical Report TR-105697, “Boiling Water Reactor Vessel and Internals Project: BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),” dated September 28, 1995.

Principal Contributor: M. Banic

Date: September 14, 2000

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Brunswick Steam Electric Plant
Units 1 and 2

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