March 7, 2005

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - SAFETY EVALUATION FOR RELIEF REQUEST RR-34, CONTROL ROD DRIVE SYSTEM HYDRAULIC LINES (TAC NOS. MC5037 AND MC5038)

Dear Mr. Gannon:

By letter dated November 4, 2004, as supplemented by letters dated November 24 and December 16, 2004, Carolina Power & Light Company (CP&L, the licensee) requested relief from ASME Code Section XI, Sub-article IWA-4300, pertaining to the structural integrity of insert, withdrawal, and charging water piping for the Control Rod Drive (CRD) System at Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The relief request was made pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii).

The Nuclear Regulatory Commission (NRC) has reviewed the proposed alternative in the subject relief request. The results are provided in the enclosed Safety Evaluation.

Based on the information provided in the licensee’s submittal, the NRC staff concludes that the licensee has demonstrated that compliance with the ASME Code repair/replacement requirements for the degraded CRD piping would result in hardship without a compensating increase in the level of quality and safety. Therefore, the proposed request for relief pursuant to 10 CFR 50.55a(a)(3)(ii) for BSEP, Units 1 and 2, is approved. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions regarding this approval, please contact Ms. Brenda Mozafari, Senior Project Manager, at 301-415-2020.

Sincerely,

/RA/

Michael L. Marshall, Jr., Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosure: Safety Evaluation

cc w/encl: See next page

If you have any questions regarding this approval, please contact Ms. Brenda Mozafari, Senior Project Manager, at 301-415-2020.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-34

CONTROL ROD DRIVE SYSTEM HYDRAULIC LINES

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 November 4, 2004, as supplemented by letters dated November 24 and December 16, 2004, Carolina Power & Light Company (CP&L, the licensee) requested NRC approval of a relief request (RR-34) for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, pertaining to the structural integrity of insert, withdrawal, and charging water piping for the Control Rod Drive (CRD) System. The licensee's relief request RR-34 was verbally authorized on November 29, 2004, to support its scheduled CRD piping inspection.

1.1 Background

During July of 2004, the licensee discovered through-wall leaks on three CRD insert lines at Unit 2. The subsequent extent-of-condition inspection was performed on the most susceptible piping at both units and additional degraded CRD piping was found. Four insert lines were repaired by installing an American Society of Mechanical Engineers (ASME) Code-compliant mechanical clamp device on each affected piping in accordance with Code Case N–532-2 to control leakage and to maintain structural integrity of the piping. The licensee believes that the root cause of these leaks and degradation is due to chloride-induced transgranular stress corrosion cracking (TGSCC) because the overhead salt water drain lines were found to be leaking.

The licensee has developed a proactive action plan to examine other piping within the CRD system that may be susceptible to this condition at both units. The proposed relief request would allow the licensee to disposition the defects through engineering evaluation in lieu of complying with the Code requirements of removing or reducing the size of any defects that are identified in the affected CRD piping.

Enclosure

2.0 REGULATORY EVALUATION

Inservice Inspection (ISI) of ASME Code Class 1, 2, and 3 components are performed in accordance with Section XI of the ASME Code and the applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated in 10 CFR 50.55a(a)(3), alternatives to the requirements of (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.

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 pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) 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 reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the third 10-year ISI interval at BSEP Units 1 and 2 is the 1989 edition, with no addenda. The third 10-year ISI began on May 11, 1998, and will end on May 10, 2008.

2.1 System/Components for Which Relief is Requested

The licensee requested relief for the following components:

-1-C11-1/2-905 (Charging Water, 137 drives)
-2-C12-1/2-905 (Charging Water, 137 drives)
-1-C11-3/4-905 (CRD Drive Water Withdrawn, 137 drives)
-2-C12-3/4-905 (CRD Drive Water Withdrawn, 137 drives)
-1-C11-1-905 (CRD Drive Water Withdrawn, 137 drives)
-2-C12-1-905 (CRD Drive Water Withdrawn, 137 drives)
-1-C11-1-905 (CRD Drive Water Insert, 137 drives)
-2-C12-1-905 (CRD Drive Water Insert, 137 drives)

The affected CRD piping is classified as an ASME Code, Class 2 component and is comprised of small diameter (i.e., less than or equal to 1-inch nominal pipe size), SA-376 Type 304 stainless steel pipe.

The normal service pressure of the CRD insert and withdraw lines is approximately 1030 psig (i.e., normal reactor operating pressure). The service pressure in these lines increases approximately 260 to 275 psi when a drive signal is given. The normal service temperature of the CRD insert and withdraw lines is approximately 120^0 F.

The normal service pressure of the CRD charging water lines is approximately 1400-1500 psig (i.e., the normal charging water pump pressure). The normal service temperature of the CRD charging water lines is approximately 120^0 F.

2.2 Code Requirements for Which Relief is Requested

ASME Code Section XI, Sub-article IWA-4300, Defect Removal, requires that a defect be removed or reduced to an acceptable size in accordance with Article IWA-4000. The licensee stated that there is substantial hardship to comply with this requirement for defects found in the affected CRD piping. Discussion of the hardship is provided in Section 3.0 below.

2.3 Licensee Proposed Alternative and Its Bases

In lieu of the ASME Code Section XI, Sub-article IWA-4300 requirement to remove the defect or reduce it to an acceptable limit, the licensee proposed, as an acceptable alternative, to disposition any CRD piping defects through an engineering evaluation that demonstrates the affected piping will maintain structural integrity.

The licensee's performance of an engineering evaluation uses the methods of the 1989 Edition of ASME Code, Section XI, Appendix C, "Evaluation of Flaws in Piping," and the non-destructive examination (NDE) data collected for the piping defects. In the flaw evaluation, the defect depth cannot be characterized and is assumed to be equal to the nominal pipe wall thickness. Consideration for potential defect growth during the period from the date of identification to the expected date of repair/replacement will be factored into the evaluation and acceptance criteria. In the crack growth calculations, the licensee assumes a potential crack growth rate of 0.036 in/month (5.0×10^{-5} in/hr) for each crack tip.

The licensee will perform an initial follow-up liquid penetrant examination of the defects within 30 days to ensure no significant crack growth is occurring, such as a crack growth rate in excess of that assumed in the engineering evaluation. Subsequent follow-up liquid penetrant will be performed at least every 90 days. In addition, the degraded piping will be monitored on a daily basis to identify any leakage that may develop.

The licensee stated that in the unlikely event that a degraded CRD pipe should begin leaking, or defect growth should be observed beyond that considered in the engineering evaluation, the control rod associated with the affected CRD line will be declared inoperable. The control rod will be inserted and disarmed. The licensee has also committed to perform an ASME Code-compliant repair/replacement activity for the affected portion of the CRD system piping no later than the next refueling outage where a CRD piping defect is identified, regardless of whether a mechanical clamping device is installed or an engineering evaluation is performed.

3.0 STAFF EVALUATION

In accordance with ASME Code Section XI, IWA-4300, when a defect is found in a component, the defect is required to be removed or reduced to an acceptable size. The licensee has provided acceptable reasoning of the hardship that it will incur in order to comply with this Code requirement when a defect is found in the affected CRD piping. The licensee's discussion of the hardship is summarized below:

(1) Ultrasonic testing (UT) cannot be used to characterize the depth of the defect on these lines because the piping diameters are very small (less than or equal to 1-inch nominal pipe size). Therefore, the affected piping cannot be accepted by evaluation and must be declared inoperable. Consequently, the associated control rod must be declared inoperable and

required to be inserted and disarmed to accomplish its safety function. While plant operation may continue with one or more control rods inserted and disarmed, a significant power reduction may be required. In addition, inserted control rods will experience reduced rod lifetime and may require early replacement and disposal at significant personnel radiological exposure and financial costs.

(2) To perform an appropriate Code repair/replacement, the affected portion of the CRD piping would require cutting and replacement. Since the affected CRD piping cannot be isolated from the CRD mechanism inside the primary containment, the unit needs to be shut down in order to perform the Code repair/replacement.

In lieu of performing Code repair in accordance with Section XI, IWA-4300 to remove the defect or reduce it to an acceptable size, the licensee proposed an alternative to disposition any identified CRD piping defects through an engineering evaluation to ensure that the affected piping will maintain structural integrity. In a response to the staff's request for additional information (RAI), the licensee stated that the engineering evaluation will be submitted to the NRC for review no later than 5 working days following completion of the evaluation. The licensee has also committed in its submittal that Code repair/replacement will be performed on all degraded CRD piping no later than the next refueling outage.

The licensee's engineering evaluation will follow the methodology delineated in Appendix C, "Evaluation of Flaws in Piping," of the 1989 Edition of the ASME Code, Section XI. The Appendix C evaluation will calculate a critical flaw size below which the structural integrity of the affected piping will be maintained during plant operation. In calculating the final flaw size at the end of the operating period, the initial length of the flaw is based on the results of the liquid penetrant examinations. The depth of the flaw is assumed to be through-wall and is equal to the nominal pipe wall thickness. This crack depth assumption is conservative and will provide additional safety margin in the calculation of the critical flaw size.

To calculate the potential crack growth for the remainder of the plant operating period, the licensee assumed a crack growth rate of 0.036 inch /month (5.0×10^{-5} inch/hour) for each crack tip. However, the NRC staff notes that this crack growth rate is the bounding crack growth rate accepted by the NRC for the intergranular stress corrosion cracking (IGSCC) of sensitized stainless steel in a boiling-water reactor (BWR) environment. In the licensee's initial submittal, adequate justification is not provided for using the proposed crack growth rate to calculate the potential crack growth in the CRD piping, which is degraded by chloride-induced TGSCC. The NRC staff requested the licensee to provide additional information (RAI) to support the use of the assumed crack growth rate. In its response to the staff's RAI, the licensee identified some laboratory test data, which were discussed in EPRI Final Report 10002792, "Materials Handbook for Nuclear Plant Pressure Boundary Applications, December 2002." The test data were reported by A. J. Sedriks to demonstrate the effects of chloride concentration and exposure time on the susceptibility of 304 stainless steel to chloride-induced stress corrosion cracking (SCC). Based on the test data presented in Figure 3-23 of the subject EPRI report, the crack growth rate at 120^0 F is estimated to be about 1.8×10^{-5} inch/hour, which is about a factor of 2.8 less than the assumed crack growth rate. The testing was performed on sensitized type 304, annealed type 304 and annealed type 304L stainless steel materials. The test temperature of 120^0 F is chosen because it is equal to the normal service temperature of the CRD insert and withdraw lines. Double cantilever beam specimens were used in the testing at a mode of constant load. The testing was performed in an environment of 22-percent NaCl solution with a range of stress intensity factors (K) from 36.4 to 45.5 ksi-in$^{1/2}$. The licensee stated that the concentration of 22-percent NaCl is near the solubility limit for NaCl in water and is applicable to the actual dryout condition of the affected CRD piping lines.

The staff reviewed the effect of stress intensity on the crack growth rate of Type 304L stainless steel, which was presented in Figure 3-22 of the subject EPRI report. The NRC staff notes that when the stress intensity factor (K) exceeded 30 ksi-in$^{1/2}$, a plateau behavior for the crack growth rate is shown in the Figure. Therefore, the test data presented in Figure 3-23 are considered very conservative when applied to the affected CRD piping because the testing was performed at bounding conditions with respect to the NaCl concentration and the level of K.

Based on a review of the test data provided by the licensee, the NRC staff has determined that the use of the assumed crack growth rate of 5.0×10^{-5} inch/hour to evaluate the potential crack growth in the affected CRD piping is conservative. This will be further supported by the licensee's monitoring program to ensure there is no excessive crack growth occurring. The licensee will monitor the crack growth in the affected piping periodically with an initial re-examination by liquid penetrant within 30 days after detection of the defect and subsequent re-examination every 90 days. In addition, daily monitoring for leakage will also be performed. The licensee's monitoring program is considered adequate because its inspection frequency is shorter than the time period (remaining operating period) considered in the crack growth evaluation.

Based on the above, the NRC staff has determined that the licensee's proposed alternative in relief request RR-34 as discussed in its submittal and summarized in this safety evaluation will provide reasonable assurance that the structural integrity of the affected CRD piping will be maintained during the plant operation. The NRC staff has also determined that to implement Code repair of the degraded CRD piping during plant operation will result in hardship without a compensating increase in the level of quality and safety. Therefore, the NRC staff finds that the licensee's relief request RR-34 is acceptable.

4.0 CONCLUSION

Based on the information provided in the licensee's submittal, the NRC staff concludes that the licensee has demonstrated that compliance with the ASME Code repair/replacement requirements for the degraded CRD piping would result in hardship without a compensating increase in the level of quality and safety. Therefore, the proposed request for relief pursuant to 10 CFR 50.55a(a)(3)(ii) for BSEP, Units 1 and 2 is approved. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: W. Koo, NRR

Dated: March 7, 2005