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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION OF THE SECOND TEN YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF REGARDING THE REACTOR PRESSURE VESSEL SUPPORT SKIRT FOR CAROLINA POWER AND LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 DOCKET NUMBERS: 50-325 AND 50-324

1.0 INTRODUCTION

The Technical Specifications for Brunswick Steam Electric Plant, Units 1 and 2, state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 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 difficulties 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 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 ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(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 Brunswick Steam Electric Plant, Units 1 and 2 second 10-year inservice inspection (ISI) interval is the 1980 Edition through Winter 1981 Addenda. The components (including supports) may meet the requirements set forth 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 therein and subject to Commission approval.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed. In a letter dated May 18, 1995, Carolina Power and Light Company submitted to the NRC its second ten-year interval inservice inspection program plan request for relief regarding the reactor pressure vessel support skirt for Brunswick Steam Electric Plant, Units 1 and 2.

2.0 EVALUATION AND CONCLUSIONS

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its second ten-year interval inservice inspection program plan request for relief regarding the reactor pressure vessel support skirt for Brunswick Steam Electric Plant, Units 1 and 2.

Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Letter Report attached. The Staff concludes the Code requirement to perform VT-3 visual examination on the interior surface of the reactor pressure vessel support skirt is impractical. The licensee did not propose an alternative examination; however the licensee's performance of the Code-required VT-3 visual examination on the exterior surface of the reactor pressure vessel support skirt and ultrasonic examination of the reactor pressure vessel-to-reactor pressure vessel support skirt welds provide reasonable assurance of continued structural integrity. Therefore, relief is granted as requested, pursuant to 10 CFR 50.55a(g)(6)(i).

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TECHNICAL LETTER REPORT ON THE SECOND TEN-YEAR INTERVAL INSERVICE INSPECTION REQUEST FOR RELIEF REGARDING THE REACTOR PRESSURE VESSEL SUPPORT SKIRT FOR CAROLINA POWER AND LIGHT COMPANY'S BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 DOCKET NUMBERS: 50-325 AND 50-324

1.0 INTRODUCTION

In a letter dated May 18, 1995, the licensee, Carolina Power and Light Company, submitted a request for relief regarding the reactor pressure vessel support skirt examination. This request for relief is for the second 10-year inservice inspection (ISI) interval, which began in July 1986, at Brunswick Steam Electric Plant, Units 1 and 2. The Idaho National Engineering Laboratory (INEL) staff has evaluated the subject request for relief in the following section.

2.0 EVALUATION

The Code of record for the Brunswick Steam Electric Plant, Units 1 and 2, second 10-year ISI interval is the 1980 Edition through Winter 1981 Addenda. The information provided by the licensee in support of the request for relief has been evaluated and the basis for disposition is documented below.

Request for Relief: Examination Category F-A, Reactor Pressure Vessel Support Skirt

Code Requirement: Table IWF-2500-1, Examination Category F-A, requires 100% visual examination (VT-3) of plate and shell type supports each inspection interval as defined by IWF-1300-1.

Licensee's Code Relief Request: Relief was requested from performing 100% of the Code-required VT-3 visual examination on the interior surface of the reactor pressure vessel support skirt.

Licensee's Basis for Requesting Relief (as stated):

"The reactor pressure vessel [RPV] support skirt is accessible from openings in the sacrificial shield wall and affords examination of the external surface of the support skirt. The interior surface of the support skirt is insulated. In order to examine the internal surface of the support skirt, removal of the insulation is required.

The skirt insulation is constructed of inner and outer casings of 304 stainless steel and inner radiation shields of aluminum alloy (e.g. mirror-type insulation). Removal of the insulation will require disassembly by either unscrewing each piece of the insulation or by cutting the insulation for removal. The pieces of insulation are layered around the internal circumference in a stacking manner.

The size of the insulation sections is a minimum of 20 inches by 30-5/8 inches. The access to the interior of the reactor pressure vessel skirt is 18 inches in diameter. Removal of the insulation and leaving the insulation in the reactor pressure vessel skirt area during examination is not possible because of the limited space between the control rod drives and the skirt area.

Removal of the insulation from the reactor pressure vessel skirt would result in permanent damage to the insulation. Re-installation of the existing insulation would not be possible due to this damage.

If the existing insulation were required to be removed and therefore permanently damaged, new insulation would be required to be designed and installed. This would result in hardship to Carolina Power & Light Company for the following reasons:

1. Removal and re-installation of the skirt insulation is estimated to result in approximately 1.4 person-REM of personnel radiation exposure for each unit.

2. The existing insulation, if removed, would have to be disposed of as low-level radioactive waste. The cost for disposal of this material is estimated to be in excess of $26,000 per unit.

3. Preparation of design documentation for new insulation would result in additional cost to Carolina Power & Light Company.

4. Purchase of material for new skirt insulation is estimated to be approximately $30,000 per unit.

5. The reactor vessel skirt material has ample ductility and is expected to exhibit significant plastic deformation prior to fracture. Any service induced damage would be associated with buckling failure and would be evident during visual examination of the reactor pressure vessel support skirt exterior surface. In addition, the RPV-to-RPV support skirt welds receive an ultrasonic (UT) examination in accordance with ASME Code Section XI."

Licensee's Proposed Alternative (as stated):

"Perform no VT-3 examination of the interior surface. The VT-3 examination of the exterior surfaces will determine the general mechanical and structural condition of the component."

Evaluation: The Code requires a VT-3 visual examination of the reactor pressure vessel support skirt. Due to difficulties in insulation removal and replacement, the licensee proposes to perform the VT-3 visual examination on only the exterior surface of the support skirt. Performing a VT-3 visual examination on the interior surface of the support skirt would require insulation removal. However, the insulation cannot be stored in the reactor pressure vessel skirt area during examination because of limited space, and it cannot be removed from the area without damage because the access opening is insufficient (18 inch diameter opening vs 20 x 30-5/8 inch insulation sections). Removal of the insulation from the reactor pressure vessel skirt would, therefore, result in permanent damage to the insulation and require new insulation to be designed and installed, which is an undue hardship. Consequently, performing 100% of the Code-required VT-3 visual examination is impractical.

The licensee proposed no alternative examination. However, examination of the exterior surface, in conjunction with the volumetric examination of the RPV-to-RPV support skirt weld, will provide reasonable assurance of continued structural integrity. Therefore, it is recommended that relief be granted as requested, pursuant to 10 CFR 50.55a(g)(6)(i).

3.0 CONCLUSION

Based on the above analysis and the information submitted, the INEL staff concludes that 1) performing the Code-required VT-3 visual examination on the interior surface of the reactor pressure vessel support skirt is impractical, and 2) performing the Code-required VT-3 visual examination on the exterior surface of the reactor pressure vessel support skirt and ultrasonic examination of the reactor pressure vessel-to-reactor pressure vessel support skirt welds will provide an acceptable level of quality and safety. Therefore, it is recommended that relief be granted as requested, pursuant to 10 CFR 50.55a(g)(6)(i).