July 25, 2000

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects
Region II

FROM: Suzanne C. Black, Deputy Director ***/RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT 99-024, "BRUNSWICK'S REPETITIVE USE OF A BOUNDING HEATUP AND COOLDOWN REACTOR PRESSURE VESSEL TEMPERATURE TRANSIENT ANALYSIS" (TAC NO. MA7247)

INTRODUCTION

On November 23, 1999, you forwarded Task Interface Agreement (TIA) 99-024 to NRR. You requested that NRR respond to questions regarding the licensee's evaluation of a reactor pressure vessel (RPV) pressure/temperature (P/T) event on January 23, 1999, at Carolina Power & Light Company's (CP&L's) Brunswick Steam Electric Plant (BSEP), Unit 1.

On January 23, 1999, CP&L was conducting maintenance on the reactor recirculation pump motor generator sets at BSEP, Unit 1. Upon completion of the maintenance work, during preparations to return the unit to two-loop operation, CP&L determined that it had decreased reactor power and flow to a point where thermal stratification had been induced in the lower head region of the RPV. As a result of this condition, CP&L determined that they had exceeded technical specification (TS) limits for the unit, including: (1) the cooldown rate for the transient had exceeded the maximum allowable cooldown rate for the unit (that is, >100°F/hr); and (2) the difference between the temperature of the reactor coolant at the lower head drain and that for coolant at the top of the reactor core had exceeded the maximum allowable lower head-to-reactor coolant differential temperature limit for the unit (that is, >145°F). In addition, CP&L determined that, after a manual SCRAM of the reactor had been performed, the subsequent heatup of the vessel had also exceeded the maximum allowable heatup rate (that is, >100°F/hr) specified in the TS. P/T limit transients also occurred at BSEP Unit 1 in 1992 and 1995.

To analyze the January 23, 1999 event, CP&L concluded that it was appropriate to use the fracture toughness evaluation of the RPV from the operational transient of 1995 as the basis for justifying continued operation of the vessel after the 1999 transient had occurred. The TIA asked NRR to answer the following questions regarding CP&L's method of evaluating the unit's RPV structural integrity following the operational transient of January 1999:

1. Is CP&L's bounding RPV temperature transient analysis acceptable and correct?

2. Is it acceptable for CP&L to use a previously generated RPV temperature analysis as the basis for complying with the TS requirement to perform an engineering analysis to justify continued vessel operability?

BACKGROUND

Appendix G to Part 50 of Title 10, *Code of Federal Regulations* (10 CFR Part 50, Appendix G) " ... specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences [AOOs] and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime."

Appendix A to Part 50 of Title 10, *Code of Federal Regulations,* "General Design Criteria for Nuclear Power Plants," defines *AOOs* as "... those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit ...."

10 CFR Part 50, Appendix G, requires, in part, that the P/T limits for an RPV must be at least as conservative as those that would be obtained if the methods of analysis and margins of safety specified in Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (henceforth termed Appendix G to the Code) were followed.

The methods of analysis in Appendix G to the Code evaluate the RPV in terms of crack-like flaws that are postulated to initiate from the inside and outside surfaces of the RPV. These flaws are postulated to have a depth equal to one-quarter of the RPV thickness, and a length equal to one-and-a-half times the RPV thickness (i.e., an aspect ratio of 6:1). The methods of Appendix G to the Code then calculate the stress intensities at the crack tips of the flaws. Both primary membrane and bending stresses (i.e., σ_m and σ_b), and secondary bending stresses (i.e., thermal stresses) are used to calculate the stress intensities at the crack tips.

The methodology of Appendix G to the Code requires that the stress intensities arising from primary membrane and bending stresses include the following safety factors (SFs): (1) an SF of 2.0 during normal operating conditions, including heatups and cooldowns of the reactor and during anticipated operational transients; and (2) an SF of 1.5 during leak-rate and hydrostatic testing conditions. The total stress intensity is then compared to a material property for the limiting material in the reactor.

This material property, the reference stress intensity factor (K_{1a}), is a function of the material's temperature and the degree of neutron embrittlement (which is dependent on the amount of copper and nickel in the material and the applied neutron fluence). The membrane stresses and K_{1a} values are then used to determine the actual allowable P/T combinations for the reactor coolant system (RCS). Operation of the RCS is considered to meet the criteria of 10 CFR Part 50, Appendix G, and the safety margins of Appendix G to the Code, if the actual P/T combinations for operation lie below or to the right of the P/T limit curves for the facility.

10 CFR Part 50, Appendix G, allows licensed utilities to operate their units only as long as they can demonstrate that the RCS is in compliance with the requirements of the rule, and with Appendix G to the Code. When NRC-licensed utilities operate their nuclear plants in a manner that exceeds the allowable P/T combinations for the RPV, as established by the P/T limit curves, the RPV must be evaluated to show that it has sufficient remaining safety margin to justify further operation of the plant.

The P/T limit curves for BSEP Unit 1 are contained in TS 3.4.9. This TS, in part, requires CP&L to operate the RCS within the appropriate P/T limit curves for the facility. The TS also, in part, prohibits CP&L from heating up or cooling down the RCS at rates exceeding 100°F/hr for normal operations, or at rates exceeding 30°F/hr for pressure-testing operations. If any of these limits are exceeded, the action statements to TS 3.4.9 require CP&L to: (1) restore the RCS temperature and/or pressure parameters to within the limits within 30 minutes; (2) determine that the RCS remains acceptable for continued operation within 72 hours; or else (3) be in Operating Mode 3 within 12 hours, and Operating Mode 4 within the next 36 hours.

QUESTION 1

The staff was requested to determine if CP&L's bounding RPV temperature transient analysis is acceptable and correct. As discussed with the Region, since certain details of the analysis were not provided, a final determination of the acceptability of the analysis could not be made.

QUESTION 2

The staff was requested to determine if it is acceptable for CP&L to use a previously generated RPV temperature analysis as the basis for complying with the TS requirement to perform an engineering analysis to justify continued vessel operability. This is acceptable, as stated in TS Basis 3.4.9, Actions A.1 and A.2, which reads in part:

> Besides restoring operation within acceptable limits, an engineering evaluation is required to determine if RCS operation can continue. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. The evaluation must verify the RCPB [reactor coolant pressure boundary] integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

OTHER COMMENTS

There was concern that CP&L may not have operated in accordance with Action Statements A.1 and A.2 of TS 3.4.9 following the reactor SCRAMS of 1992, 1995 and 1999. With respect to these requirements, the staff has determined that CP&L met the appropriate conditions of the TS, and therefore was in compliance with the TS requirements.

On August 10, 1993, the staff issued Information Notice 93-62, "Thermal Stratification of Water in BWR Reactor [Boiling Water Reactor] Vessels," to alert BWR licensees that loss of forced circulation through the reactor vessel coupled with isolation from the main condenser may allow cold water to stratify in the bottom of the reactor vessel and cause temperatures to be lower than allowable. BWR licensees also received vendor and industry group notifications regarding similar issues. CP&L has exceeded their 10 CFR Part 50 Appendix G-required P/T

limit curves three times during transient operating conditions that have occurred in the 1990-2000 time frame due to conditions similar to those described in these notifications. Though TS provide a remedy (completing an engineering evaluation) when the P/T limits are exceeded, it is the staff's expectation that when the limits are exceeded (particularly when the licensee has been notified of a specific operating issue) the licensee would submit new P/T curves to avoid multiple events.

The licensee is preparing to submit license amendment requests for Units 1 and 2 for approval of new P/T curves. These curves will be reviewed by the staff to determine if they provide the margin necessary to bound the three events under discussion. In addition, the licensee has indicated that operating procedures and training have been enhanced so that operators can better identify conditions that could lead to thermal stratification.

Docket Nos. 50-324 and 50-325

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