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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO SALEM ATWS EVENT, ITEMS 3.1.3 AND 3.2.3

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated November 11, 1983, the Carolina Power & Light Company (CP&L, the licensee) submitted a response to our Generic Letter 83-28 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. This review covered Items 3.1.3 and 3.2.3.

2.0 BACKGROUND

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open upon an automatic reactor trip signal from the Reactor Protection System. This incident occurred during the plant startup and the reactor was tripped manually by the operator about 30 seconds after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the under voltage trip attachment. Prior to this incident, on February 22, 1983, at Unit 1 of the Salem Nuclear Power Plant, an automatic trip signal was generated based on steam generator low-low level during plant startup. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip. Following these incidents, on February 28, 1983, the NRC Executive Director for Operations (EDO), directed the staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the Salem unit incidents are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission (NRC) requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System Reliability Improvements.

Item 3.1.3 (Post-Maintenance Testing of Reactor Trip System (RTS) Components) requires licensees and applicants to identify, if applicable, any post-maintenance test requirements for the RTS in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Item 3.2.3 extends this same requirement to include all other

safety-related components. Any proposed Technical Specification changes resulting from this action shall receive a pre-implementation review by NRC.

3.0 EVALUATION

Our review of the licensee's submittals was performed with the assistance of EG&G, Idaho, Inc. The submittal from CP&L was reviewed to determine compliance with items 3.1.3 and 3.2.3 of the generic letter. First, the submittal was reviewed to determine if these two items were specifically addressed. Second, the submittal was checked to determine if there were any post-maintenance test requirements specified by the Technical Specifications that were suspected to degrade rather than enhance safety. Last, the submittal was reviewed for evidence of special conditions or other significant information relating to the two items of concern.

The review of Generic Letter 83-28, Item 4.5.3 may result in proposed changes to the Technical Specifications requirements for surveillance testing frequency and out-of-service intervals for testing. The primary concern of Item 4.5.3 is the surveillance testing intervals. Items 3.1.3 and 3.2.3 are specifically directed at post-maintenance test requirements. These concerns are essentially independent. However, the evaluation of these concerns are coordinated so that any correlation between these concerns will be adequately considered. Since no specific proposal to change the Technical Specifications has been submitted, there is no identifiable need at this time for correlating the reviews of item 4.5.3 with this review.

We have reviewed the November 7, 1983 CP&L response to Items 3.1.3 and 3.2.3 of Generic Letter 83-28. Within the response, the licensee's evaluation for Items 3.1.3 and 3.2.3 is that, following a review of the Brunswick Standard Technical Specifications, no testing requirements which tended to degrade the safety of either the Reactor Trip System components or that for the other safety-related components were identified. The scope of the review included only those requirements which were clearly defined in Technical Specifications as post-maintenance tests. The licensee committed to continue the review for Items 3.1.3 and 3.2.3.

4.0 CONCLUSION

The licensee stated that it has reviewed its Technical Specification requirements to identify any post-maintenance testing which could be demonstrated to degrade rather than enhance safety and found none that degraded safety. Based on our review, assisted by our contractor, EG&G, Idaho, Inc., we find that the licensee's submittal, with respect to the Brunswick facilities, is acceptable.

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Dated: July 16, 1985