CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461

MAY 18 1995

SERIAL: BSEP 95-0210

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO VERBAL REQUEST FOR ADDITIONAL INFORMATION
REVIEW OF INDIVIDUAL PLANT EXAMINATION SUBMITTAL
(TAC NOS. M74387 AND M74388)

Gentlemen:

The purpose of this letter is to document information provided by Carolina Power & Light Company (CP&L) during an April 20, 1995 telephone conference conducted with members of the Nuclear Regulatory Commission staff and their contractor, Science and Engineering Associates (SEA) concerning CP&L's August 31, 1992 response to Generic Letter 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities — 10 CFR 50.54(f)." Carolina Power & Light Company's response to the NRC questions are provided in Enclosure 1.

Please refer any questions regarding this submittal to Mr. George Honma at (910) 457-2741.

Sincerely,

R. P. Lopriore
Manager — Regulatory Affairs
Brunswick Nuclear Plant

GLM/WRM/wrm

Enclosures

pc (with enclosures):
Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, NRC Senior Resident Inspector - Brunswick Units 1 and 2
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
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BACKGROUND:

By letter dated August 31, 1992, Carolina Power & Light Company (CP&L) submitted a response to NRC Generic Letter 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities — 10 CFR 50.54(f)." Subsequently, by letter dated August 9, 1994, the Nuclear Regulatory Commission (NRC) staff requested additional information concerning the CP&L response. By letter dated February 27, 1995, CP&L provided responses to the NRC staff questions.

SUMMARY:

On April 20, 1995, a telephone conference was conducted with CP&L, NRC, and the NRC contractor, Science and Engineering Associates (SEA). During the discussions of CP&L's response to Staff Question F.E.10(b), CP&L was requested to characterize the relative effect on core damage frequency of a loss of net positive suction head (NPSH) due to containment venting.

Carolina Power & Light Company stated that this scenario presumes that containment pressure cannot be controlled above NPSH limits whenever repeated containment venting cycles occur. Although CP&L believes this is a conservative assumption, specific thermal-hydraulic timing studies were not available to demonstrate the validity of the assumption. Therefore, a sensitivity study was performed to respond to the NRC question.

The Individual Plant Examination (IPE) fault tree gates that involved loss of venting were examined. For venting to be successful, the venting path and operator actions must succeed and a low pressure injection source must be available. The fault tree for loss of venting contains a logic gate that combines the venting faults with the low pressure injection faults. The loss of low pressure injection gate of the fault tree assumes that the suppression pool is the suction source for low pressure injection and does not consider other sources, such as the Condensate Storage Tank. The gate for loss of low pressure injection represents the probability of a loss of NPSH due to the operator failing to terminate venting, which is the appropriate event to evaluate in the sensitivity study. From the IPE results, the Fussell-Vesely importance for the operator action, CAC-XHE-FC-VNT65 (1.49E-2) and the failure probability for the operator action (1.07E-1) were used to estimate the Risk Achievement Worth (RAW). The RAW represents the relative increase in core damage frequency if all ventings resulted in loss of NPSH and loss of low pressure injection, which therefore provides the insights for the sensitivity study. The following relationships were used to estimate the RAW:

RAW = CDF (Probability of Loss of NPSH = 1) / CDF (Base Case)

RAW = 1 - (Fussell-Vesely Importance) * (1 - 1/Basic Event Probability)

RAW = 1 - (1.49E-2) * (1 - 1/1.07E-1)

RAW = 1.12

Therefore, the worst-case increase in core damage frequency would be 12 percent, which is relatively not significant. This sensitivity study does not take credit for recovery actions, for which there would be a significant amount of time available. Also, if the suppression pool were unavailable due to the loss of NPSH, other water sources could be available for injection. Finally, this study does not consider that injection requirements at this point in time of the sequence would be on the order of 100 gallons per minute or less. These mitigating factors would reduce the overall increase in core damage frequency. In conclusion, the postulated assumption has an insignificant effect on core damage frequency.

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
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LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage*
1. None	N/A