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Ms. Mary Drouin
Office of Nuclear Regulatory Research
Mail Stop T-10-E-50
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Comments on Draft NUREG-1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance (61 FR 65248)

Dear Ms. Drouin:

This letter is in response to Mr. Miraglia's letter, which was dated December 11, 1996. Mr. Miraglia's letter encouraged Carolina Power & Light Company (CP&L) to provide comments on the draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report" and identified three particular areas in which specific comments would be welcomed. Attached to this letter are CP&L's specific comments in response to Mr. Miraglia's letter as well as CP&L's general comments on NUREG-1560.

NUREG-1560 was comprehensive and thorough. It cited over 500 safety enhancements that were identified and evaluated as a result of the Individual Plant Examinations (IPEs), almost half of which were implemented before the IPEs were submitted. This fact indicates that the industry and the NRC focus first on safety.

CP&L looks forward to working with the NRC to revitalize progress on risk-based regulation.

Sincerely,

WSOrser

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Attachment

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cc: Mr. J.B. Brady, USNRC Resident Inspector - HNP, Unit 1
Mr. B.B. Desai, USNRC Resident Inspector - HBRSEP, Unit 2
Mr. N.B. Le, USNRC Project Manager - HNP, Unit 1
Ms. B.L. Mozafari, USNRC Project Manager - HBRSEP, Unit 2
Mr. C.A. Patterson, USNRC Resident Inspector - BSEP, Units 1 and 2
Mr. D.C. Trimble, Jr., USNRC Project Manager - BSEP, Units 1 and 2

CP&L Comments on Draft NUREG-1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance (61 FR 65248)

Specific Comment for Mr. Miraglia's letter - Updated IPE Information

CP&L concurs with NRC's assessment that "... most licensees have updated their IPEs and have made hardware and procedural/operational modifications that have not been reflected in the docketed information." For CP&L's plants, the currently estimated Core Damage Frequencies (CDFs) are lower than the estimates submitted in 1992 (Brunswick and Robinson) and 1993 (Harris). These reduced values have resulted from plant safety improvements as well as Probabilistic Safety Assessment (PSA) model enhancements to remove known excessive conservatism. For the NRC's information, the following are the current CDF values for the CP&L plants:

	IPE CDF Value	Current CDF Value
BNP	2.7E-5 per year	9.2E-6 per year
HNP	7.0E-5 per year	3.6E-5 per year
RNP	3.2E-4 per year	8.0E-5 per year

These values are subject to change, either higher or lower, as the models are updated to reflect plants modifications that were installed in past outages and to incorporate peer review comments.

Specific Comment for Mr. Miraglia's letter - Need for Followup Regulatory Activities

With respect to "... the potential need for followup regulatory activities by the NRC based on IPE insights," CP&L does not consider additional NRC review of the IPEs to be warranted, because the information is now dated and the IPEs have served the intended purpose of identifying potential vulnerabilities to severe accidents. Further progress on risk-based regulation would better encourage the investment of additional resources into understanding current PSA models.

Specific Comment for Mr. Miraglia's letter - Insights In Light of Global Perspectives

CP&L recommends caution in using global perspectives to draw meaningful conclusions about any particular nuclear plant. Each IPE reflects a unique set of plant design, operating philosophies, and PSA modeling practices; this was particularly true for the Level 2 part of the analyses. For example, the treatment of induced steam generator tube ruptures was inconsistent across the industry. Also, as an example of uniqueness at a CP&L plant, CP&L reiterates the Brunswick drywell situation as described in Section 12.2.1.2 of the report: Brunswick has a, unique for Mark I, concrete enclosed containment design which precludes shell melt-through as an important contributor to early containment failure. The IPEs were necessitated by the individuality of the plants; therefore, their greatest strength is within the context of the plant for which they were prepared. Therefore, CP&L recommends that NUREG-1560 be revised to specify an emphasis on the context when CDFs are used as figures of merit.

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General Comment on NUREG-1560 - Information Evaluated

1. The IPEs were snapshots in time and are now 4 to 5 years old. Improvements in plant operation, modifications and maintenance practices have been made since the original submittals. If the CDFs and CCFPs listed in NUREG-1560 cannot be revised to quantify these improvements, CP&L recommends that a qualifying statement be added to the conclusion of NUREG-1560 to represent that the values presented are a snapshot in time.

2. For completeness, CP&L recommends that the conclusions of NUREG-1560 be revised to included the additional information which the utilities provided at the request of the IPE reviewers.

3. The report noted, on page 1-3 of Volume 1, that "...the staff used information from each IPE, even if a licensee's IPE/PRA [Probabilistic Risk Assessment] was unacceptable (in part or overall), and no adjustment or modification was made." CP&L recommends that NUREG-1560 be revised to contain a statement as to the evaluated potential effect of this limitation on the conclusions.

General Comment on NUREG-1560 - Quality of PSAs

1. Consistent with NRC's policy as reflected in "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statements" (60 FR 158), the quality of the PSA should be commensurate with the proposed application. However, NUREG-1560 implied or could be interpreted to require a complex, highly detailed "state-of-the-art" PSA for all applications. The attributes of a "quality" PSA go well beyond the point of diminishing return for most potential applications. These attributes might include: general elements such as extended mission time for core melt progression beyond 24 hours where Large Early Release Frequency is the risk measure of interest, initiating events such as common cause failure of two DC buses, accident sequence quantification requirements such as rigid truncation limits rather than application specific limits and uncertainty analysis elements such as propagation of uncertainties through the Level 2 model. In many instances these requirements are not necessary for practical, risk-management decisions, especially when PSA results are used as supplemental information. CP&L's policy is that PSA results will not be the sole basis for making operational or maintenance decisions. These decisions require a blended approach which factors in defense-in-depth requirements, probabilistic and deterministic information (when available), and sound professional judgment. CP&L recommends that NUREG-1560 be revised to reflect NRC policy as previously stated in Volume 60 of the Federal Register Number 158.

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General Comment on NUREG-1560 - Human Reliability Analysis (HRA)

1. NUREG-1560 was inconsistent in its statements about HRA. Specifically, there was a statement, on page xviii of Volume 1, that in most cases there was little evidence that the HRA quantification method, per se, had a major impact on results. Later, there was considerable text devoted to the inconsistency and variability of HRA methods (Section 5 3.1, pages 5-13 through 5-15) and its identification as the most important shortcoming for some of the IPEs (Volume 1, page xix). CP&I recommends some reconciliation or revision of these statements.

2. Although HRA and common cause failure analysis of PSAs will always be imperfect, CP&L considers, based on their overall impact on the results, that PSAs with imperfect HRAs can still be useful tools. In fact, CP&L considers operator error estimates to be reasonably accurate, especially within the context of other PSA uncertainties, if dependencies have been considered and if a thorough evaluation of the available time, training, procedures and environment has been performed. CP&L recommends that NUREG-1560 be revised to describe appropriate limitations for the use of HRA and common cause failure analysis.

General Comment on NUREG-1560 - Peer Review

1. NUREG-1560 stated, on pages 6-12 and 14-66, that other utility PSA practitioners are unsuitable. Because much of the PSA expertise currently resides in utility staff, CP&L disagrees with the NUREG's position on peer review. CP&L recommends that this view and the NUREG-1560 text be altered to a more realistic and practical position.

General Comment on NUREG-1560 - Other

1. CP&L disagrees with the use of Conditional Containment Failure Probability (CCFP) as a safety indicator. As documented in the PSA Applications Guide and presentations to the Advisory Committee on Reactor Safety PRA Subcommittee, it is not an appropriate figure of merit. CP&L recommends that NUREG-1560 be revised to reflect this.

2. CP&L disagrees with NUREG-1560's assessment that the use of the MAAP code and the use of industry position papers produced results of similar quality. Based upon significant experience in the application of the MAAP code and containment PSA modeling, CP&L found that proper use of the MAAP code resulted in much better plant-specific insights than those obtained with position papers. For this reason, CP&L considers the code and its results to be sufficiently robust to identify vulnerabilities. The NUREG itself noted that much new information had come to light on post-core melt phenomena. The state of knowledge about these types of events was not definitive. Therefore, CP&L recommends that NUREG-1560 be revised to eliminate its characterization of a severe accident analysis code as inadequate.