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ILLINOIS POWER

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March 7, 1997

Ms. Mary Drouin
Office of Nuclear Regulatory Research
Mail Stop T-10-E50
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: Illinois Power Comments on Draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report"

Dear Ms. Drouin:

This letter provides Illinois Power (IP) comments on draft NUREG-1560. This NUREG was issued for public comment, as identified in the November 14, 1996, Federal Register. Clinton Power Station (CPS) personnel responsible for the CPS Individual Plant Examination (IPE) Program have reviewed the draft NUREG and have the following comments.

- Pages 6-1, 6-2, and particularly Chapter 14 infer that a Probabilistic Risk Assessment (PRA) must meet all the "Characteristics of a Quality PRA" before being applied. Many of these characteristics are not necessary for all applications such as the examples listed below:

  - Use of PRA to prioritize/schedule the more important motor operated valves (GL 89-10) before those less important ones so valves contributing more to core damage frequency are done in earlier outages.

  - Use of PRA in determining the training periodicity of certain simulator scenarios for the Licensed Operator Requalification Program.

  - Use of PRA to provide outage planning and maintenance schedulers the risk impact when taking equipment and combinations of equipment out of service.

  The use of this inference may make application of PRA impractical since the cost of meeting all these characteristics is costly. As such, clarification should be made, to wit, the characteristics required depend on the application involved.

- On the last paragraph of Section 11.2.3.4, Page 11-46 describes that "At Clinton, a loss of a non-safety DC bus initiating event results in a loss of feedwater and the condenser • • •." This description is not accurate. The loss of feedwater does not necessarily result in loss of the condenser. Loss of the condenser, as well as loss of the other injection systems, is based on random and dependent failures modeled into the linked fault trees. The event tree is shown on Figure 3.1-3 of the CPS IPE submittal dated September 23, 1992. The fault tree linking and quantification is described in Sections 3.2.2 and 3.3.7 of this submittal.

- On the second complete paragraph of Section 11.2.3.3, Page 11-43 (et. al.) indicates that other plants do not indicate that feedwater runback results in MSIV closure as does Perry. The reason that the other plants do not indicate this is that Perry is unique among at least the Boiling Water Reactor (BWR) 6s in having an Anticipated Transient Without Scram (ATWS) feedwater runback. The other BWR-6s have a recirculation pump runback instead.

- On the last line of Table 5.2 on page 5-6, River Bend is identified as a BWR-6 with a Mk I containment when it actually has a Mk III containment. This erratum is also made on Table 13.2 of page 13-15.

We at Illinois Power appreciate this opportunity to comment on the NUREG prior to its issuance. We sincerely hope that the above comments are reviewed and as appropriate, incorporated into NUREG-1560 prior to its issuance.

Sincerely yours,

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DUKE POWER

USNRC

March 3, 1997

Chief, Rules Review and Directive Branch
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Washington, D. C. 20555-0001

Subject: Duke Power Company Comments on Draft NUREG-1560

This is in response to the December 12, 1996 NRC letter, inviting comments on the draft NUREG-1560, "INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE". Duke Power Company has had the opportunity to perform a cursory review of Vol. 1 Part 1 and Vol. 2 Part 2-5 of the draft report series, and our comments from this limited review are provided here.

Comment 1: The Individual Plant Examination (IPE) process (together with certain pre-IPE studies by some licensees) has resulted in the identification and recognition of the important severe accident sequences and improvements in the plant capability to deal with severe accidents across the board in US nuclear power plants. As such, the draft NUREG correctly recognizes that the IPE program resulted in improving the overall safety of nuclear power plants.

Comment 2: The report presents useful information on the estimated core damage frequencies of various types of accidents and the range of core damage frequencies for the various types of plants. It also presents important conclusions on the containment capabilities for severe accidents. Our overall impression of the information presented in the report is that the staff has put in a very good effort in assimilating and synthesizing the many pieces of information from the IPEs and presenting the key results and conclusions with respect to the severe accident risk for US plants in an informative manner.

Comment 3: As described in our IPE submittals, the Duke plants performed pre-IPE Probabilistic Risk Assessment (PRA) studies which led us to implement several plant enhancements. Subsequently and as part of the IPE program,



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additional accident sequences were considered for cost-effective improvements. Those deemed worthwhile and cost-effective have now been implemented. We have reviewed the list of important PWR plant improvements listed in the draft NUREG. We conclude that the list is similar in scope and intent to the ones Duke previously considered - namely, improve electric AC reliability, ensure reliability of Emergency Core Cooling System (ECCS) sump re-circulation, achieve high reliability of feed-and-bleed operation, achieve high reliability of the Decay Heat Removal (DHR) system, ensure reliability of service water system, and prevent Reactor Coolant Pump seal loss-of-coolant accidents (LOCAs).

Since the IPE studies, the vital DC system batteries at Catawba have been replaced to improve reliability. Efforts are underway now to replace the vital DC system batteries at McGuire. Improvements have been made at Catawba in the Nuclear Service Water system and the Component Cooling system reliability. In addition, a modification to provide a backup cooling capability for the high head safety injection pumps at Catawba is scheduled for implementation in 1997. This modification would further reduce the core damage sequence frequencies for accidents initiated by either loss of the Nuclear Service Water system or the loss of the Component Cooling system. Also, the Emergency Operating Procedure changes to reduce the likelihood of post-core-damage, thermally induced steam generator tube failures have been made.

With respect to Oconee, the IPE/PRA models and data were updated in 1995, following the Keowee PRA study, to take into account the plant changes and current operational data. A summary report presenting pertinent information on the current estimated severe accident risk for Oconee is being sent to the NRC under separate cover.

At this time, the McGuire and Catawba PRAs are being updated. Copies of the summary report will be sent to the NRC when the reports are completed.

Comment 4: Concerning any follow-up regulatory activities, we suggest that the investigation and regulatory considerations not be limited just to the high core damage frequency (CDF) or conditional containment failure probability (CCFP) issues. Areas where the risk impact is small and the safety benefit is not appreciable should also be investigated for reduced regulatory burden, similar to the improvement in the Appendix J rules recently

promulgated. In this regard, it is pointed out that at this time there are a number of resource intensive regulatory issues the NRC and the licensees are investigating: GL-96-01, GL- 96-06, accuracy and availability of design basis information, process for performing 10 CFR 50.59 evaluations, etc. It is suggested that the NRC utilize the results, conclusions, and insights from the IPE program and the risk-based decision making to ensure that the response to these issues is properly balanced with respect to nuclear safety.

Through the IPE and related processes, our understanding of the qualitative and quantitative probabilities of accidents and failure modes of equipment important to safety has substantially improved from that of the 1960s, when 10 CFR 50.59 was implemented. We suggest that this improved understanding be utilized in defining a threshold of "increase in probability" for which NRC review is needed, as contemplated in 10 CFR 50.59. This will enable focusing licensee and NRC efforts on the more risk and safety significant issues and areas of plant operation.

Comment 5: In grouping plants on the basis of "high" CDF or CCFP, the criteria as to what and why constitute "high" should be formulated carefully and objectively, considering the precision in the estimates of these probabilistic events. For example, is a plant with an estimated CDF of 1.04 E-4 less safe than a plant with an estimated CDF of 0.96 E-4?

We appreciate the opportunity to provide these comments.

Very truly yours,

M. S. Tuckman

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