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

NRC PDR

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NOV 7 1978

Docket No: 50-466

Mr. G. W. Oprea, Jr.
Executive Vice President
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas .77001

Dear Mr. Oprea:

SUBJECT: OUTSTANDING SAFETY REVIEW ISSUES - ALLENS CREEK NUCLEAR GENERATING STATION, UNIT 1

We forwarded requests for supplemental information on December 29, 1977; February 17, 1978; March 8, 1978; March 24, 1978; and April 14, 1978. On the basis of our review of your responses provided in the Preliminary Safety Analysis Report as amended by amendments through Amendment 45, we identified a number of outstanding safety review issues in the enclosure to our letter of July 21, 1978. On the basis of our review of your responses provided in the Preliminary Safety Analysis Report as amended through Amendment 47, we have concluded that the items described in the enclosure continue to be outstanding review issues.

Within 10 days after receipt of this letter, please advise us of the date by when you can provide information that we need for resolution of the issues in the enclosure.

Sincerely,

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosure:
Request for Additional
Information

cc: See next page

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ccs:
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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

010 <u>Auxiliary Systems</u>

010.5 In Section 3.1.2.4.17.1 of the PSAR, "Evaluation Against Criterion 46," you state "The essential Services Cooling Water System will be designed to permit testing of system operability with simulation of emergency reactor shutdown or LOCA conditions and transfer between normal and emergency power sources." Provide clarification that this commitment includes consideration of the coincident loss of the cooling lake as included in the design basis events (Section 9.2.5.3.1.2 of the PSAR) in the evaluation against the requirements of Criterion 46 of the General Design Criteria.

110.0 Mechanical Engineering

110.2 For the PSAR to be in agreement with the staff position as discussed in Standard Review Plan 3.6.1 and 3.6.2 and Appendix F to the Safety Evaluation Report, the first sentence in Section 3.6.2.2.4(g)(3) must be removed. That sentence now states, "Piping welds subject to 100 percent volumetric inspection will be those short sections of process pipe themselves which serve as a part of containment, i.e., no guard pipe exists to contain the rupture in this section."

110.3 and 110.4 The staff position remains as stated in the enclosure to our letter of July 21, 1978.

110.6(3) 110.17 130.20 In your response to Item 130.20, provided by Amendment No. 47 to your PSAR, you stated, "The applicant commits to apply the generic resolution of this issue to the design of Allens Creek. However, for cases where the generic resolution cannot be practically implemented such as steel plate structures within the containment boundary the applicant will justify the acceptability of the design to the satisfaction of the NRC staff." Provide clarification that for each such case construction or installation will not be completed until NRC staff approval of the justification has been obtained.

110.7 You have not provided sufficient information in the PSAR to enable us to complete our review of the design criteria to be used for supports for ASME Class 2 and 3 components. Specifically, design criteria and loading combinations have not been provided for standard and plate and shell type supports in the balance-of-plant scope of supply and for all ASME Class 2 and 3 supports in the nuclear steam supply system scope of supply.

For ASME Class 2 and 3 linear supports in the balance-of-plant scope of supply the stress limits and the methods used to combine responses as described in the PSAR are not completely acceptable. The information is not sufficient to enable us to complete our review of the design criteria proposed for these supports. In particular and for the information in Table 3.9.8 of the PSAR (1) the factor of 1.2 under an upset condition does not exist in NF, (2) no faulted limits are given and (3) verification that the faulted buckling limit complies with F.1370(c) should be provided.

110.18 The staff in its generic review has not completed its review of recommendations 5,6 and 7 of the report ORNL/SUB/2913.8. Therefore, for Allens Creek, reliance or those recommendations is not acceptable at this time. Revise your commitment to omit a dependency on those recommendations. A commitment to the generic resolution which results from the ongoing discussions between the NRC and the BWR Mark II Owner's Group would also be accept .le.

130.0 <u>Structural Engineering</u>

130.6 On the basis of discussions with your representatives in a meeting in Bethesda, Maryland on October 30, 1978, we agreed to further review your proposed analysis methodology with modifications that you proposed to delineate in an amendment to your Preliminary Safety Analysis Report. Generally, we understand that the methodology would be modified for frequencies less than 8 Hertz to utilize accelerations calculated using your methodology modified to accommodate a free-field Regulatory Guide 1.60 response spectra control motion at an elevation corresponding to the bottom of the reactor building mat. In addition to a description of the modifications to the methodology and the bases as described in the meeting, provide the following additional information for our review:

(1) For systems and components in the reactor building, auxiliary Building, and fuel handling building with natural frequencies less than 8 hz, we understand that you will increase the amplitude of the floor response spectra for design by a multiplier determined by the ratio of the Flush b/Flush a response spectra within designated frequency ranges. Provide the explicit criteria to be used to establish these multipliers. The Flush b response spectra should be based on the use of the envelope of G_{AVE}, $G_{AVE}*1.5$, and $G_{AVE/1.5}$, or the response spectra based on G_{AVE} should be broadened by plus or minus 15%.

(2) In order to justify the use of the Flush a analysis, provide a comparison of the design shears and moments as computed by the Flush a analysis and the Spring an analysis for the (reactor building, including the shield building, steel containment, drywell, and the RPV pedestal).

211.0 Reactor Systems Branch

211.2 With regard to the postulated loss of a CRD pump (Item 211.2-07/27/78), our position remains:

(1) You should provide the bases that will be used to determine that unacceptable impairment of control rod scram capability has occurred,

(2) You should provide automatic protection or demonstrate that 20 minutes is available for operator action, and

(3) You should describe initial and periodic test programs that will be used to demonstrate that this capability is maintained for plant lifetime.

You should also describe how the leak performance of the check valves will be continuously monitored via monitoring of accumulator pressure, as stated in the response.

211.3 Your response to Item 211.3 requires supplemental discussion. In particular, we note that this break size (~.02 ft^2) produces a peak cladding temperature in excess of the temperature produced by a large break DBA previously analyzed. The following additional information should be provided.

(1) Justify that the system provided for diversion of LPCI flow meets single failure criteria so that diversion before 10 minutes need not be considered.

(2) Provide further justification that a diesel failure causing loss of the LPCS is more limiting than a loss of the LPCI for core cooling. It is not apparent in your discussion that CCFL effects on

reflood times were included. Discuss the relative effects of these low pressure systems upon the parameters in the LOCA calculations, e.g., reflood, core heat transfer.

(3) Provide a sensitivity study showing peak clad temperature as a function of break size for small break LOCA's assuming diversion will be initiated at 10 minutes. Perform this study for HPCS and recirculation line breaks. For the most limiting break, provide the following figures:

(a) Water level inside the shroud as a function of time during the LOCA

(b) Reactor vessel pressure vs. time

(c) Convective heat transfer coefficient vs. time

(d) Peak clad temperature vs. time

(e) ECCS flow rate vs. time.

(4) Justify that diversion at times greater than 10 minutes will have less severe consequences than diversion at 10 minutes (considering appropriate break sizes for later diversion).

(5) Provide a discusssion which balances the need for LPCI diversion for this break size ($\sim$.02 ft^2) with the need for abundant core cooling (GDC 35). For example, this discussion could relate to Figure 6.2-33 with regard to the likelihood of LPCI diversion for this size break.

211.22 In your additional response in Amendment 47 to the PSAR you stated, "The applicant will adopt the generic resolution of this issue for the ACNGS. The applicant reserves the right to provide an acceptable alternative at a later date." Provide a commitment that alternatives will be provided for staff review and will not be constructed or installed until staff approval in obtained.

211.26 Additional information is needed relative to detection of leakage into the HPCS and the RCIC systems to either show conformance with the position of Regulatory Guide 1.45 or to demonstrate that such leakage does not need to be considered.

361.0 Geosciences Branch

361.5 In Section 9.2.5.3.2 of the PSAR you state, "In the event that the rate of sediment accumulation is such that it appears that the allowable level of accumulation will be exceeded during the life if the plant, the sediment will be removed before that allowable limit is reached." In addition to level of sediment accumulation, limits on slope of the surface of the accumulated sediments should be considered to assure that unacceptable consequences will not result from sediment flow into pumps intakes during design basis events. State the allowable configurations for accumulated sediments within the cooling lake and provide a preliminary description of the technical specifications that will be used to assure maintenance of acceptable sediment configurations. Include criteria, procedures, and technical specifications for maintaining sediment configurations.