March 3, 2000

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear, Inc.
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) - INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS OYSTER CREEK NUCLEAR GENERATING STATION (TAC NO. M83652)

Dear Mr. Roche:

Based on our review of the Oyster Creek Individual Plant Examination of External Events (IPEEE) submittal dated December 29, 1995, and the response to our previous request for additional information, we are unable to conclude at this time that the licensee has met the intent of Supplement 4 to Generic Letter 88-20. The licensee's previous response to our RAI was incomplete. The licensee stated, in their response dated May 21, 1998, that additional information would be supplied for question 1 on the fire analysis and question 3 on the hurricanes, floods, high winds, transportation, and other external events (HFO) analysis. In addition, based on our review of your previous response to the seismic questions, we are unable to conclude that the intent of the Generic Letter has been met, with respect to the seismic analysis. The enclosed request for additional information on the IPEEE seismic analysis was developed by our contractor, Brookhaven National Laboratory. Your response to these questions, as well as the fire question and HFO question mentioned above, are necessary in order for us to complete our review.

The enclosed RAI was discussed with George Busch of your staff on December 23, 1999, and it was agreed that your response would be submitted within 60 days of the date of this letter. If you have any questions, please call me at the earliest opportunity. I may be reached at (301) 415-1261.

Sincerely,

/RA/

Helen N. Pastis, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosure: RAI

cc w/encl: See next page

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Enclosure: RAI
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SUPPLEMENTAL REQUEST FOR ADDITIONAL INFORMATION

OYSTER CREEK NUCLEAR GENERATING STATION

SEISMIC PORTION OF THE OYSTER CREEK

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

1) The discussion in the submittal and the RAI responses regarding the use of certain non-safety systems in the probabilistic risk assessment (PRA) model lacks sufficient detail. The concern is that these systems were not designed for the seismic motion of interest, or they may be housed in non-safety structures. In addition, there is a question whether all the failure modes and all the components of such systems were adequately considered. While these systems already fail with high probability in seismic ground acceleration ranges SEIS3 and SEIS4, centered at .54g and .72g, respectively (See Table 3-1 of the submittal), significant failure probabilities of such systems in the seismic ground acceleration range SEIS2 (centered at 0.36g) might lead to substantial changes in the results of the seismic analysis. Therefore, please provide the following information regarding the firewater makeup to the isolation condenser, and the use of the combustion turbines and turbine building equipment for power recovery after a seismic event:

(a) What types of firewater pumps are used (e.g., electric, diesel driven, etc.)? Are any parts of the firewater makeup system, from the pumps to the isolation condenser, seismically qualified? If so, which ones?

(b) Were structural, support, and other failure modes of all the components of the firewater makeup system evaluated? Please list which components were evaluated, what the failure modes and the calculated capacities were, and which failure modes and components were omitted, and why.

(c) Are there any special considerations regarding operator actions related to the use of this system after a seismic event, which do not exist for such actions in the internal PRA (e.g., impeded access for actions outside the control room, or timing of such actions)? If so, what were these considerations and how were they taken into account?

(d) For power recovery using the non-seismic combustion turbines, which failure modes and components were considered? Was failure of the combustion turbines' batteries considered (e.g., due to failure of the battery spacers)? If so, what were the results of the evaluation? If not, please provide a justification for not doing so.

(e) There are potential failure modes of the emergency switchgear located in the turbine building, as well as the underground power cables to the combustion turbines, related to block wall failures, soil liquefaction or turbine building structural failure. How were these failure modes incorporated into the analysis of power recovery following a seismic event?

2) It is not clear from the RAI response to the original seismic question 3, sent previously, whether the recirculation pump support failure was considered in developing the Nuclear Steam Supply System (NSSS) seismic capacity numbers. Failure of such supports has been shown to be an important contributor to seismically induced loss-of-coolant accidents (LOCAs) and to the seismic core damage frequency (CDF) in some boiling-water reactors (BWRs).

   (a) Please provide the seismic capacity estimated for the recirculation pump supports and discuss how this capacity was obtained.

   (b) Was the failure of the recirculation pump supports considered as one of the failure modes in developing the NSSS piping seismic capacity? If it was not, what is the justification for omitting the support failure?

3) The RAI response to the original seismic question No. 4, regarding the basis for the screening threshold fragility, states that the threshold fragility, when combined with the mean seismic hazard curve, results in an annual probability of failure of 8.8E-6 per year. The threshold fragility has a median ground acceleration capacity of 1.0g and a high-confidence-of-low-probability-of-failure (HCLPF) capacity of 0.3g. Our interpretation of the application of this threshold for screening fragilities is that if a component had an estimated median ground acceleration capacity greater than 1.0g and a HCLPF capacity of at least 0.3g, then the component was omitted from the seismic logic model, unless all components in a system were screened out, in which case the threshold fragility curve was used for a surrogate component to represent the system. Moreover, from Table 3-8 of the original submittal, it seems that the only case where such a surrogate component was used was for the fire protection system.

   As stated in the GPU IPEEE submittal, the total seismic CDF is 3.6E-6 per year. Since components were not included in the seismic logic model if their median ground acceleration capacity and HCLPF capacity exceeded the corresponding capacities of the threshold fragility, and the threshold fragility leads to an annual seismic failure frequency of 8.8E-6 per year, it appears that an appreciable underestimate of the seismic core damage frequency can occur. If the combination of a seismic event, and failure of a screened-out component (or two completely correlated similar components) leads to core damage, then such a sequence would contribute about 9E-6 per year to the core damage frequency. This exceeds the stated estimate of 3.6E-6 per year for the seismic CDF. With the low screening threshold fragility curve used, the seismic PRA analysis may fail to identify important seismic sequences, and important insights into seismic system response may be overlooked. In light of the above, please:

   (a) Increase the threshold for component screening, such that the annual failure probability of an individual component screened out with this threshold would represent only a small fraction of the total seismic CDF. (The screening threshold should be established such that the convolution of the threshold mean fragility curve with the mean site hazard curve leads to an annual component failure probability much less than the currently calculated mean seismic core damage frequency of 4E-6 per year. Then the insights as to dominant seismic sequences would not be affected by the screened out components.)

(b) Evaluate the effect of this threshold increase on the existing seismic sequences.

(c) Determine whether new seismic sequences could be established as a result of the threshold increase.

(d) Update the seismic CDF value.

(e) Report the results of your re-analysis.