Docket No. 50-333                                                    May 20, 1992

Mr. Ralph E. Beedle
Executive Vice President - Nuclear
Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION - JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M74411)

Based on our ongoing review of the Individual Plant Examination (IPE) for the James A. FitzPatrick Nuclear Power Plant, we have identified the need for additional information in order to complete the review. We, therefore, request that you provide a response to the questions enclosed. In order to facilitate our current review schedule, we request that you provide written responses to the list of questions within 90 days of the date of this letter.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P. L. 96-511.

Sincerely,
Original Signed By:
Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
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Project Directorate I-1
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Enclosure:
As stated

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See next page

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ENCLOSURE

JAMES A. FITZPATRICK
INDIVIDUAL PLANT EXAMINATION (IPE)
REQUEST FOR ADDITIONAL INFORMATION

1. With regard to the peer-review process please provide:

   (A) A summary of the in-house peer-review group findings, including recommended changes, and the disposition of recommendations. (NUREG-1335 notes the benefit of having the IPE reviewed in-house.)

   (B) A listing of technical findings and recommendations of the three outside consultants that reviewed the IPE and a discussion of the disposition of any recommendations.

2. Discuss the treatment of plant-specific design and operational provisions that assure the long term makeup capability to the condensate storage tank (CST) in order to achieve the successful long term operation of the High Pressure Coolant Injection (HPCI) system or the Reactor Core Isolation Cooling (RCIC) system (after its suction switched back to the CST from the suppression pool) and the long term Control Rod Drive (CRD) injection to the reactor vessel during the containment venting scenario.

3. With regard to the treatment of internal flood, discuss the IPE's assessment of failure of the check valves located inside the drain system between two independent rooms having independent safety components.

4. Provide a concise discussion of the IPE's treatment of Power Conversion System (PCS) recovery (if it would have been lost during the initial 30 minute period of the transient). Include in this discussion the dependency information between the condenser and the reopening of the MSIVs and bypass valves.

5. Provide a concise discussion of recovery of failed Residual Heat Removal (RHR) pumps, Residual Heat Removal Service Water (RHRSW) pumps, and Core Spray (CS) pumps due to common cause failures as documented in Table 3.3.4.1 of the IPE. Include in this discussion the mission time versus recovery time involved for injection and long term decay heat removal, and the availability of overriding equipment involved, if any.

6. Discuss the treatment of DC load shedding, if needed, following a station blackout scenario, or loss of AC buses 10500 and 10600 scenarios. Does FitzPatrick take credit for additional batteries for long term HPCI and RCIC initiation and controls to avoid a core damage event? If so, please describe treatment and justification for credit.

7. Provide a summary discussion of the process used to address pressurization of the wetwell air space following a postulated pipe break event (subsequent to a successful scram or fail-to-scram event) in the Safety Relief Valve (SRV) discharge piping.

8. Describe the process used to estimate train level unavailability due to test and maintenance and human errors. Discuss the estimation of these components of train level unavailability for the Electrical System (transformers and inverters) and RHR System (injection mode, spray mode, pool cooling mode and shutdown cooling mode) as examples of the application of the above process.

9. Provide a concise discussion of the treatment of mechanical failure and the overall electrical failure of the Reactor Protection System (RPS) and basis for the probabilistic estimates including derivations used and applicability.

10. Discuss the process used to treat unavailability of the coolant injection function through the Control Rod Drive system to the reactor and the basis for the probabilistic estimates.

11. Discuss the process used to examine the nitrogen ventilation and purge valves as part of sequence development in addition to any individual systems analysis.

12. Section 3.3.2.2 of the IPE acknowledges that the exposure time for various operating and standby components, and demand spectra (assigned cumulative number of demands for components) for standby components have been estimated for FitzPatrick. Briefly describe the calculations made in estimating these two parameters in the Service Water system and the HPCI system.

13. Describe the process used to treat the following: (A) Common cause failure (fail-to-start mode) of two pumps, (B) Common cause failure (fail-to-continue-to-run mode) of two pumps, (C) Common cause failure (fail-to-open on demand) of two MOVs, (D) Common cause failure of two LPCI batteries to supply power to their loads. Also, describe the treatment of plant-specific common cause factor estimates for two and three stuck-open failure mode of the SRVs.

14. Provide a discussion of the treatment of pressure locking of motor operated double disc gate valves and flexible wedge gate valves (experienced at FitzPatrick in 1988 and 1991, respectively), and impact of corrective actions taken upon the IPE results.

15. Generic letter 88-20 requires licensees to certify that their IPE reflects <u>current</u> plant design and operation. It is our understanding that the operational data provided in Appendix D has been utilized to determine plant specific hardware failure rates only and for the limited period of 1980 to 1986 Since 1986, many changes have occurred, such as design changes, parts supplier changes, manufacturing specification changes, equipment aging, etc., as well as changes in plant personnel training and the plant maintenance programs. This generates a question of whether the FitzPatrick IPE addresses the current plant status. Please provide a discussion of the impact of plant changes that have occurred since 1986 and the effect of failure rate estimates for the more current period. (Use references as appropriate)

16. FitzPatrick has a wealth of operating experience from which to update and improve generic human reliability estimates (which would otherwise need

to be utilized in the IPE). Please discuss the process used to capitalize on this experience, specifically with regard to the generation of human error probabilities (HEPs) and perception of human error in the overall results.

17. Please identify those instances in which performance shaping factors (PSFs) are used to modify HEPs according to the difficulty of the tasks under analysis, and discuss the rationale for the PSF selection. It appears that the operator response to extremely difficult situations has been evaluated optimistically. For example, for the Anticipated Transient Without Scram (ATWS) initiating event, where the operator has 1 to 3 minutes to recognize that it is an ATWS, the operator must enter EOP-2, follow EOP - 2 to the point where he is directed to enter EOP - 3, enter EOP - 3 and verify that he must initiate Alternate Rod Injection (ARI) and Recirculation Pump Trip (RPT) and override ADS. The IPE, on the basis of FitzPatrick's good operator training, assumes an HEP less than 1E-5. Describe the PSFs used to account for the stressful situation and the limited time for operator response.

18. The human reliability analysis (HRA) is based on generic basic human error probabilities (BHEPs) modified by recovery factors (RFs) "which limit undesirable consequences of human error by allowing for human redundancy ....." (pg. 3-379). Thus the HRA reduces the generic BHEP value of 0.03 through the use of RF(s). In the example given on page 3-379 for the calibration of a pressure transmitter, the generic value of 0.03 was used as the HEP for this task for the typical or nominal plant. The generic BHEP is then reduced by a factor of 0.01 to account for post-calibration testing and independent verification. We call your attention to page 5-6 of NUREG/CR-4772 which provides guidance for the use of the methodology you have adopted. Please note that Step 2 on page 5-6 states that "No downward adjustment (of the BHEP) should be made without a more thorough HRA of the kind specified in NUREG/CR-1278". It is our understanding that the BHEP value is assumed to already account for normal or typical "checks & balances" for operator actions. Therefore, the application of RFs to further reduce the BHEP value should be based upon procedures, QA techniques, independent verifications, maintenance practices, etc. which are significantly superior to those typically found in the average or nominal plant. Please take a sample of 5 or 6 nominal human error probabilities (NHEP) values from table 3.3.1 and discuss the RF values used to adjust the BHEP value and discuss how they are supported by factors for FitzPatrick which clearly demonstrate that the FitzPatrick "checks and balances" are significantly better than those normally utilized in the typical or nominal plant.

19. Please describe and discuss your analysis of operational experience (i.e. LERS, training material and procedures updates, maintenance and surveillance test records, etc.) used to identify human error initiated events and common cause failures.

20. Please specify the BHEP and any RFs used to estimate the probability (NHEP) of failure to vent the wetwell (local operation) upon demand (i.e. Containment Pressure $\geq$ 44 psig), and discuss the basis for selection of the BHEP and RF values. Relevant factors to be discussed include the EOP covering containment venting, location and operator access to vent valves and/or their controls, training of the operators required to perform the venting function as well as the effect of such factors as stress, time, and environmental conditions such as temperature and radiation levels expected to exist in the locality of the vent valve controls.

21. Table 4.5.1.1 indicates an internal containment failure pressure for Peach Bottom (PB) of 150 psig. NUREG-1150 identifies an estimated mean failure pressure of 148 psig for PB. In Section 4.5.1 <u>Static Overpressure Containment Failure</u>, you use a containment failure pressure of 159 psig for Peach Bottom and reduce it by 12-13% (to account for thinner vent line bellows at FitzPatrick) to obtain a failure pressure of 140 psig for the FitzPatrick IPE. Please provide your basis for using the 159 psig value as a basis for determining the estimated failure pressure rather than the 150 psig value from your comparison of FitzPatrick vs. Peach Bottom Major Plant Features (Table 4.5.1.1) or the 148 psig value from NUREG 1150 (Vol.1, page 4-12). Use of the 148 or 150 psig values would result in an estimate of failure pressure for FitzPatrick of about 130 psig. Please discuss the effects of this lower value on the timing and probability of overpressure containment failures. In addition, Section 4.6.1-<u>Selection of the CET</u> seems to indicate that, in spite of the above comparison between PB and FitzPatrick, the PB containment probabilities and failure modes were used in the FitzPatrick CET. Please clarify this statement and discuss the comparison of the two plants and how it has been used to assign values to the FitzPatrick CET.

lease clarify the apparent discrepancy concerning the amount of ircalloy available. Tables 4.2.2.1 and 4.5.1.1 indicate a Zircalloy core inventory of 111,216 lb. However, Table 4.3.2.2 indicates a total core inventory of 131,051 lb. Which value is correct? Which value was used in the IPE? In the event that the smaller value is incorrect and was used in the IPE, discuss the impact of the larger value.

23. With regard to Section 4.5.4-<u>Containment Isolation System (CIS) Failures</u> please identify the CIS failure probability(s) used in the IPE, and contributors to CIS failure. Please identify the necessary failures for the three SBO bypass leak paths identified in Section 4.5.4 and provide the basis for your conclusion regarding their improbability.

24. With regard to Section 4.5.5-<u>Containment Electrical Penetration Failures</u>, please provide plots of containment atmosphere temperature vs. time from the MAAP-3.0B analysis for accidents with Direct Containment Heating (DCH). Compare the electrical penetration environmental qualification temperatures to the temperature profiles predicted for DCH events from

the MAAP runs, and provide your basis for concluding that the probability of electrical penetration failures is so small that they need not be considered as a possible containment failure mode. Please identify and discuss the process used to treat any active or passive equipment located in the drywell which is assumed or required to function during DCH events.

25. With regard to Section 4.5.6.2-Containment Drywell Melt-through, please discuss the consistency of your IPE insights with those described in draft NUREG/CR-5423, "The Probability of Liner Failure in a Mark-1 Containment", dated January 1990, (or the more recent final report dated August, 1991.) Discuss the effects of the insights from this most recent work upon the liner failure probabilities shown in Table 4.5.6.1.

26. On page 4-55 in the third line from the top of the page, please identify the starting event for the 24 hr. termination of the analysis of Core-Concrete Interactions (CCI), i.e. is the 24 hrs. measured from the start of initiating event, core damage, vessel failure or CCI?

27. Examination of Figures 4.7.4.3 and 4.7.4.5 seem to indicate that for PDS-1 there is a probability of early containment failure of 0.038 from some mechanism other than drywell melt through, drywell overpressure rupture, or wetwell venting. Is this representative of containment bypass leaks (i.e. event V and/or containment isolation failure)? This unidentified mechanism seems to have a frequency of 3.9 x $10^{-8}$/yr and accounts for 2.1% of all core melt events. Please clarify this and discuss its significance.

28. Generic Letter 88-20 Supplement 1, dated August 29, 1989, requests that BWR licensees with a Mark I Containment design address the specific Mark I Containment Performance Improvements (CPIs) identified in the supplement to GL 88-20 and references 1 and 2 below. Please examine the suggested CPIs and provide your evaluation of the value/impact associated with the suggested improvements and any sensitivity with regard to estimated core damage frequency. (Use references as appropriate.)

29. Please discuss the containment walkdowns performed to confirm that the IPE represents the as-built, as currently operated plant. Please identify the operations staff and level-2 experts who participated in containment walkdowns.