June 9, 1994

Docket Nos. 50-254
and 50-265

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) ON THE QUAD CITIES STATION, UNITS 1 AND 2, INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NOS. M74456 AND M74457)

The Nuclear Regulatory Commission staff is reviewing the IPE submittal dated December 13, 1993, for Quad Cities Station, Units 1 and 2. Additional information is required from Commonwealth Edison Company in order for the staff to complete its review. We request that you provide a response to the enclosed RAI within sixty days from the date of this letter to meet our review schedule.

If you have any questions, please contact me.

Sincerely,

Original Signed By:

Chandu P. Patel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosure: RAI

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Mr. D. L. Farrar
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

cc:

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION FOR QUAD CITIES IPE REVIEW

1. Two aspects of the IPEP methodology are not clearly explained in the Submittal.

   (a) The manner in which the methodology considers dependencies among events in the event trees is unclear. It appears that the event trees do not consistently use split fractions, since more than two branches were not developed for all events on the event trees. It also appears that fault tree linking was not used either. The Submittal states: "Some nodes were determined to be dependent on other nodes which preceded them on the plant response trees. In order to account for these dependencies, and ensure correct quantification of the accident sequence, conditional failure probabilities were calculated and used in place of the fault tree quantification results as appropriate." [IPE, Section 4.5] It is not clear how this approach rigorously accounts for shared components among systems. Please explain by way of examples.

   (b) It is not clear what recovery means in the IPE. Is it taking credit for extra equipment or operator actions restoring for equipment? Regardless, it appears that recovery is included in the PRT models before initial calculations were performed instead of, as typically done, being applied to the dominant sequences. Please provide (1) a clear definition of "recovery," (2) a description of the treatment of recovery and (3) the data used for the recovery of offsite power.

2. All transient initiating events except loss of offsite power were modeled as a general transient. Please explain (a) what plant-specific initiating events comprise the general initiating event category, (b) how the event-specific effects of these initiating events on the availability of mitigating systems were considered in the models, (c) discuee why loss of HVAC and loss of instrument air were screened out as initiating events.

3. The submittal does not provide, unlike a typical PRA/IPE, a description of any event sequence for any PRT; this makes the review of the event trees very difficult. Please provide descriptions of the event sequences for the general transient event tree.

4. The success criteria for a large LOCA indicates that 1 LPCI pump can be used for mitigation and that 1 RHRSW pump is adequate for containment cooling. Please explain whether (a) the success criteria for LPCI accounted for leakage at the jet pumps' slip and bolted joints, and (b) the analysis supporting 1 RHRSW pump considered the potential fouling conditions in the RHR heat exchanger(s).

5. The submittal provides no detailed information on the quantification of interfacing system LOCAs as initiating events. Please discuss (a) your treatment of the systems modeled for interfacing systems LOCAs and (b)

the source of data used to quantify the probability of failure for components exposed to beyond design basis pressure.

6. The submittal does not address dual unit core damage. Quad Cities has three diesel generators, one of which is shared between the two units as a swing diesel, and either of the diesels can mitigate loss of offsite power at a unit. Station blackout at one unit involves increases the likelihood of station blackout at the other unit, due to the unavailability of the shared diesel generator. Please provide an estimate of the frequency of dual unit core damage.

7. The success criteria for core damage considers fuel temperature but not peak cladding temperature (PCT). PCT is the limiting parameter for maintaining coolable geometry except for rapid overpower transients. Please explain (a) why was PCT not considered in the criteria for core damage; and (b) how can a coolable geometry be assured based on consideration of fuel temperature alone.

8. Does the small LOCA initiating event include LOCAs due to failures in recirculation pump seals? If not, what is the justification for not considering pump seal LOCAs? If so, discuss the basis for the small frequency assigned to a small LOCA.

9. The systems/sequences success criteria for preventing core damage have numerous differences from the success criteria used in other PRA/IPE studies of BWRs. Please respond to the following issues:

   (a) The submittal states that reactor trip is not required following a large LOCA. The ECCS water is not borated. The following studies of BWRs required reactor trip following a large LOCA: WASH 1400, NUREG 1150 for Peach Bottom, NUREG 1150 for Grand Gulf, IPE for Browns Ferry, IPE for Fermi, and IPE for Perry. What is the basis for not requiring reactor trip?

   (b) The submittal states that only one relief valve is required to depressurize so that LPCI or CS can be used. Other PRA/IPEs assumed that more than one valve is required; for example, the NUREG 1150 study for Peach Bottom assumed that 3 are required. Please discuss any calculations that support this assumption.

   (c) The submittal models containment venting for the back-end analysis, but the front-end success criteria do not address containment venting. If the containment is vented with a hot suppression pool, adequate NPSH can be lost for ECCS pumps pulling from the suppression pool; this was the assumption in the NUREG 1150 analysis of Peach Bottom. Please explain how this aspect was addressed in the Quad Cities IPE.

   (d) Credit is taken for supplying containment with water from the Standby Coolant System (SBCS) following a large LOCA if suppression pool cooling is lost. This system involves using feedwater to supply water to the containment from the condenser hotwell with the hotwell makeup supplied from service water. This preserves adequate NPSH for the LPCI and CS pumps. Please (1) discuss all support systems and operator actions required to implement this option, (2) provide sample calculations regarding time required and time available for the operator actions and (3) explain how overfill of containment is controlled.

(e) Credit is taken for switching long term cooling with LPCI or CS from the suppression pool to the CCST if containment cooling is lost following a transient, to prevent loss of adequate NPSH for the pumps. Please (1) describe all support systems and operator actions required to implement this option, (2) provide sample calculations regarding time required and time available for the operator actions and (3) explain how overfill of containment is avoided.

(f) Explain why the success criteria for a large LOCA do not address the need to close the recirc discharge isolation valve in the intact recirculation loop to prevent loss of LPCI-injected water out the break.

(g) Explain why the success criteria for LOCAs do not address LOCAs outside containment, for example in steam and feedwater lines, and the need to isolate the breaks to prevent loss of suppression pool inventory from ECCS out the break.

(h) Explain why the success criteria for ATWS do not discuss operator action to inhibit depressurization to prevent reactivity increase due to the injection of large quantities of cold water.

(i) Explain why the success criteria indicate that if the core is cooled using feedwater, SSMP or CRD, containment failure has no impact on the ability to maintain core cooling.

(j) Explain why the success criteria for a small LOCA do not consider the use of RCIC.

(k) Explain how did the submittal model the requirements for cooling of electrical switchgear, battery rooms, and the control room.

(l) The success criteria for medium LOCA indicate that use of HPCI followed by use of either LPCI or CS can cool the core for 24 hours without containment heat removal. What is the justification for not requiring containment heat removal for a medium LOCA?

10. The Submittal uses the following modeling for an ATWS: OSL1 is operator action to use 1 of 2 SLC pumps to borate, and the PRT indicates that if OSL1 is not successful, OSL2 can successfully provide boration, and OSL2 is operator action to use 2 of 2 SLC pumps. Please explain (a) why are

there two actions; (b) whether these actions are independent? and (c) how can 2 pumps be used if one pump is not used.

11. The common cause failure values used in the IPE are lower than those typically used in other IPE/PRAs. For example, the beta factor for failure of 2 MOVs is a factor of 5 to 9 lower than the beta factor typically used and the beta factor for DGs is a factor of 10 lower than the beta factor typically used. Screening common cause data for plant-specific applicability based on expert opinion is a rather questionable approach because common cause events address classes of events as opposed to specific events as typically are screened for applicability. (a) Please provide the justification for the low common cause factors used in the IPE. (b) Explain why the common cause factors used for MOVs are applicable in general and also to the RHR system MOVs. (c) Justify the low values used for common cause failure of the relief valves. (d) Why was common cause failure of the RCIC and HPCI turbine driven pumps not addressed?

12. The submittal screened out internal flooding as a contributor to core damage frequency. Please provide (a) a description of the flood sources considered, (b) the locations of these sources, (c) the equipment failed as a direct result of the floods, and (d) the criteria used for screening out the floods. Also, the DET noted that the electrical switchgear in the turbine building is subject to flooding from overhead water bearing lines, and in fact one of the MCCs had a tarpaulin installed to prevent water from dripping into the switchgear. How did the IPE address flooding-induced failure of electrical switchgear in the turbine building?

13. Please identify the core damage frequency from the SAM accident sequences, and summarize the major actions that can be taken to prevent core damage in these sequences.

14. Please explain the involvement of Quad Cities plant operations and maintenance personnel in the development and review of the PRTs, and fault trees.

15. Please address the following data related issues: (a) does plant-specific data support failure rates used for MOVs in general, and for MOVs in the RHR system, (b) is the use of generic data for failure of relief valves supported by plant-specific data, (c) why is the frequency for an inadvertent open relief valve so low, 0.01 per Table 4.1.1-1 of the Submittal, compared to values in other BWR IPE/PRAs, and (d) why is the value for inadvertent opening of a relief valve in Table 4.1.1-1 given as 0.01 when footnote 1 of this table implies the value should 0.1?

16. Please explain the following aspects of the systems modeling used in the IPE. (a) What is the impact of loss of recirc pump seal cooling with RBCCW during the mitigation phase of a transient and how was this considered in the IPE? (b) How did the low turbine backpressure trip

setpoint of RCIC, 25 psig, affect the availability of RCIC in the IPE model? (c) What battery lifetime during station blackout was used and what is the basis for this value?

17. The results of the submittal indicate that station blackout is a dominant contributor to core damage. For loss of all AC power at one unit, credit is taken for supply of power by crosstie from the other unit via the 14-1 and 24-1 buses. (a) Please describe how the crosstie is accomplished. (b) What other actions must be taken to provide power to both units using one diesel generator dedicated to one unit if the swing diesel generator has failed and the dedicated diesel at one unit has failed?

18. Human errors that occur during routine operations, for example, during calibration or restoration of equipment after test or maintenance, are called *pre-initiator* human events. These types of errors may leave a system in undetected disabled state and therefore unavailable at demand. In many PRAs such errors were found to be significant. The submittal does not provide a discussion for these types of events. Table 4.4.2-2, however, lists at least one pre-initiator, "Failure to restore 1/2-4101A(b) diesel fire pump following test or maintenance" which indicates that some pre-initiator human error analysis was performed. Please:

    (a) Provide a brief and concise discussion of how pre-initiator events important to system and component unavailability were identified.

        (1) Include a description of the reviews of the test, maintenance and calibration procedures performed for the systems and components modeled.

        (2) Include a description of discussions held with appropriate plant personnel from the maintenance, training, and operations departments on the interpretation and implementation of the plant's test, maintenance and calibration procedures.

        (3) Include descriptions of actual test, maintenance, or calibration activities observations performed in order to better evaluate how existing error control procedures may impact the availability of the system(s) (or component(s)) on which these activities are performed.

    (b) Provide a brief discussion of the quantitative or qualitative screening process that might have been used in order to identify the most significant pre-initiator human errors.

    (c) Provide the list of pre-initiator-type errors that were finally modeled (usually on the fault tress) and quantified *and provide examples of their quantification process.*

19. Provide examples demonstrating how dependencies associated with pre-initiator human errors were addressed and treated in the IPE to assure that important accident sequences were not eliminated. These dependencies could, for example, affect the availability of many safety systems simultaneously, or could affect the availability of only a certain class of systems (e.g., complete dependence may be assumed for miscalibration of all reactor water level sensors). Dependencies are identified through the examination of factors such as:

- Plant conditions (e.g., poor lighting)
- Human engineering (e.g., labels, accessibility etc.)
- Performance by same crew, same time
- Adequacy of training
- Adequacy of procedures
- Interviews with training, operations and various crews

20. Human actions that are needed during an abnormal event for mitigation are called *post-initiator* human events. These events involve failure to properly respond to an abnormal event by either not performing the required activities as directed by the plant's procedures (e.g,. EOPs), or not recognizing the critical faults and taking proper action. *Post-initiator* human events can be further distinguished as:

- *Response type actions*, those human actions performed in response to the first level directive of the EOPs. For example, suppose the EOP directive instructs the operator to determine reactor water level status, and another directive instructs the operator to maintain reactor water level with system x. These actions — reading instrumentation to determine level and actuating system x to maintain level — are response type actions.

- *Recovery type actions*, those performed to recover a specific failure or fault. For example, suppose system x failed to function and the operator attempts to recover it. This action-diagnosing the failure and then deciding on a course of action to "recover" the failed system-is a recovery type action.

Section 4.4.2.1.2 of the submittal states that the Quad Cities General Abnormal Procedures (QGAs) do not explicitly provide direction to the operator, and therefore, the operator is acting from memory. It is assumed that the operator will respond to the requirements of the QGAs based upon how they have practiced the evolutions during training. Therefore, for those actions of concern to the IPE, Job Performance Measures (JPMs) were used rather than operating procedures.

(a) Of the Actions in Tables 4.4.2-1 and 4.4.2-2 please indicate:

(1) Which Actions can be characterized according to the above terminology as response-type and which can be characterized as recovery-type.

(2) Which Actions have JPMs and which do not have JPMs.

(3) For those that JPMs exist, please provide an example demonstrating how the JPM was used for modeling the errors and discuss how Quad Cities' use of JPMs in lieu of plant emergency and operating procedures assured that potentially significant human actions or steps within an action were not overlooked.

(4) For those actions for which JPMs do not exist, please discuss how operator response was modeled and provide an example illustrating the process.

(5) Provide a brief and concise description of discussions that were held with appropriate plant personnel (e.g., operators, shift supervisors, and training) during Phase 1 of the HRA regarding the *interpretation and implementation of plant procedures* (or JPMs) in order to *identify important actions to be modeled* as well as critical steps within an action, and to *understand exactly how specific components are manipulated when responding to an accident sequence.* Such discussions would assure an accurate representation of operator response into the plant model. They would also serve as a vehicle to improve the knowledge of operations staff on important aspects of their performance during an abnormal event.

21. Section 4.4.2.1.2 of the submittal indicates that there are two parts of post-initiator operator actions: (a) detection, diagnosis, and decision, and (b) action execution. The submittal states that the HEPs for two types of actions were taken from the "appropriate tables" in Chapter 20 of THERP (NUREG/CR-1278). Similarly, probabilities for recovery actions were taken from "appropriate tables" in Chapter 20 of THERP. Please:

(a) Identify the exact Tables used and provide examples of the assigned HEP values for each of the two parts of post-initiator operator actions and for both types, response-type and recovery-type actions.

(b) Provide examples of HRA trees used to analyze human actions involving the two parts of operator actions.

(c) For each of the examples, discuss the underlying assumptions and plant-specific assessments used for assigning these values.

(d) Provide examples to demonstrate how the probabilities for the two parts (1) detection, diagnosis, and decision, and (2) execution were combined to provide the final estimate for the HEP.

22. Section 4.4.2.1.1 seems to distinguish between two types of post-initiator operator "action executions": time-critical (non-use of procedures) and non-time-critical (use of procedures). The IPE states that time available for operator response was determined from MAAP results and that the Quad Cities General Abnormal Procedures (QGAs) are in a flow-chart format and provide very general guidance for the operators. Therefore, the line up of systems directed by the QGAs is accomplished from memory by the operators, without initial reliance on procedures. For time-critical actions, then, the initial operator response is expected to be from memory. However, the operators are expected to consult with the procedures as time permits. This represents a recovery opportunity which is dependent upon enough time being available and is included in the "slack time" recovery. In addition, Section 4.4.2.1.2 notes that a "slack time" recovery factor is applied to actions that are to take place greater than an hour after the initiating event. "Slack time" refers to the amount of time available to the operator over and above that necessary to diagnose and perform the action. However, no information is provided in the submittal regarding the evaluation of the expected time needed for operators to complete actions. Please address the following:

(a) For each of the 24 operator actions listed in Table 4.4.2-1 and actions in Table 4.4.2-2, identify which actions were considered as non-time critical and which as time-critical, under different accident conditions.

(b) For those identified as *non-time critical*, please discuss:

(1) How the time needed for preceding activities was taken into account for determining these actions as non-time critical.

(2) The underlying hypotheses for MAAP time calculations.

(3) Explain whether MAAP estimations were further confirmed by the input of plant operations personnel and actual time measurements (through simulator or walk throughs).

(c) For those identified as *time-critical*, explain:

(1) How "slack time," i.e., time available minus time needed, was determined for a specific action performed under different accident conditions.

(2) How the time needed for preceding activities was taken into account.

(3) How the time for detection, diagnosis, decision, was differentiated from time for execution.

(4) The underlying hypotheses for MAAP time calculations and explain whether MAAP estimations were further confirmed by

the input of plant operations personnel and actual time measurements (through simulator or walk throughs).

(d) Provide both time available and time needed for each time-critical action in Tables 4.4.2-1 and 4.4.2-2.

(e) *Provide examples* of the quantification process of time-critical tasks clearly indicating how the operator performing from memory can recover due to "slack time" and how the final mean value was assigned.

(f) Explain *by way of example(s)* how the quantification of actions that are to take place greater than an hour after the initiating event differs from the quantification of non-critical time actions.

23. Section 4.4.2.1 of the submittal states that in order to take credit for operator recovery from an error by an independent cue (a procedure check, an alarm, or other persons checking), the nominal HEPs were modified by factors taken from THERP "Table 20-22 (3) Appendix A "checking that involves special short term one-of-a kind checking with alerting factors". The staff was not able to identify Table 20-22 (3) Appendix A in THERP (NUREG/CR-1278). Item (3) of Table 20-22, "Estimated Probabilities that a checker will fail to detect errors made by others," refers to recovery due to "special short term one-of-a kind checking with alerting factors." Also, this Table is appropriate for errors associated with pre-initiator error quantification. Further, it appears that the equations from Table 20-17 were used for error detection rather than values from Table 20-22. Please explain.

24. In order to account for the effects of stress, the nominal HEPs were modified by stress factors (of 1, 2, or 5) taken from THERP Table 20-16 (NUREG/CR-1278). As indicated in THERP (NUREG/CR-1278), however, the values extracted from this Table are the factors suggested for the quantification of errors associated with "routine," (pre-initiator) activities rather than values suggested for "dynamic" tasks such as the tasks during a post-initiator event. Thus, the IPE used modifiers for stress by at most a factor of 5 although THERP suggests the use of an HEP of 0.25 for tasks performed under "extreme high stress." Please explain.

25. Section 4.4.2.1.2 states for the first hour into the accident, only "identifiable recovery opportunities" are credited. The probabilities associated with these recovery opportunities are taken from the appropriate tables in THERP. The submittal also notes that a non-recovery probability of .54 was considered applicable to operator actions taking place after the first 15 minutes of the scenario. This takes credit for the availability of the shift control room engineer. In addition, the submittal states that an additional credit (multiplying by .21) was applied for actions occurring one hour (slack time) into the accident. The submittal also states that the applicable recovery

factor for operator actions that are required later than one hour into the sequence is the product of the two recoveries (.54 X .21 = .11). Subsequently, however, (page 4-158 of the submittal) it is stated that the .11 factor was applied for all cases of "slack time" recovery. Please:

(a) Identify the THERP table used for the "identifiable recovery opportunities."

(b) Provide examples and discuss how recovery factors were combined; was recovery applied on each individual subtask or was it applied after an initial quantification of the action?

(c) It appears that if an action is to be executed "one hour" into the accident two recovery factors were applied (.54 and .21). Usually in THERP only one recovery factor is applied (either .54 or .21). Further, it is not clear how these recoveries were applied with regard to time available and time required considerations. The simple fact that an action will be needed "one hour into the accident," does not justify recovery credit, unless time and other considerations were explicitly examined. Please explain by way of examples.

(d) For the actions of Table 4.4.2-1 identify which recovery factors were applied.

(e) Briefly discuss whether the two types of recovery opportunities (identifiable recovery opportunities occurring within the first hour vs "slack time" recovery opportunities occurring after the first hour), are considered to be independent of each other.

26. Section 4.4.2.1 states that a "decision tree" was used to assign dependency levels between plant response tree (PRT) nodes. The submittal also states that the formulae for conditional probability of a task "n", given failure of previous task "n-1" for each level of dependence, as presented in Table 20-17 of NUREG/CR-1278, were utilized to appropriately modify the HEP for any given operator action or subtask. Please:

(a) Provide a copy of the decision tree and provide examples explaining its use in the Quad Cities HRA.

(b) Provide examples of the application of Table 20-17 of the THERP handbook to evaluate dependencies among subtasks within a single operator action.

(c) Clearly indicate how levels of dependency were assessed.

27. Table 4.4.2-1 provides mean HEPs for numerous cases of operator actions. However, the submittal does not contain descriptions of these cases in order to understand how the cases differ. For example, what is the

difference between the two cases of operator action OAT "operator action to initiate ARI system," that would result in the mean HEP that differs by two orders of magnitude? For the top ten operator actions listed in Table 4.4.2-1,

(a) Provide the event descriptions for each case, clearly indicating how stress, dependency and recovery factors were addressed.

(b) Provide examples of the quantification process for several of the class I cases within each of the top ten operator actions. If class I cases are not appropriate for a particular operator action, use class II or class III examples.

28. Table 4.4.2-2 indicates that the mean HEP for "operator fails to initiate core spray following failure of automatic initiation" is 7.5E-03. However, in the Dresden IPE submittal, this operator action is given a mean HEP of 1.0. This seems to be a large difference in HEP values for two very similar plants. Please provide the quantification and event description for this operator action.

29. Table 4.4.2-1 lists a mean HEP of 5.1E-02 for ORP "Operator manually initiates recirculation pump trip." Operator action OSPC "Operator action to initiate suppression pool cooling," however lists several HEPs, for example, 5.5E-03 for case 4 and 5.2E-05 for case 2. Further, mean HEPs of 1.7E-04 and 3.8E-04 were listed for operator action OIADS, "Operator action to inhibit ADS." Please discuss why lower HEPs are found for a relatively complicated operator actions such as OSPC and OIADS as compared to the much less complicated action, ORP.

30. The IPE submittal states that, "There were no accident sequences that dropped below the core damage frequency criteria because the frequency had been reduced by more than an order of magnitude by credit taken for human recovery actions not defined in the Quad Cities emergency procedures." NUREG-1335, Section 2.1.6, item 5, states "In addition to sequences reported under the screening criteria, any sequence that drops below the core damage frequency criteria because the frequency has been reduced by more than an order of magnitude by credit taken for human recovery action should be discussed." NUREG-1335, then, applies to all actions, not just those that are non-proceduralized. Please (a) discuss whether credit for any proceduralized or non-proceduralized human recovery action resulted in a sequence being reduced by more than an order of magnitude to a value below the screening criteria, and (b) identify and briefly discuss any sequence that was reduced to below the screening criteria because of this credit.

31. The submittal notes that 81 insights were identified in five different categories (plant-specific procedure enhancements (51%), hardware enhancements (26%), training (6%), information (15%), and test and maintenance (2%)). However, no specific insights were provided in the IPE. Further, the submittal suggests that human error is a relatively minor contribution to plant risk, yet only 26% of the 81 IPE insights

were purely hardware-related. This would seem to imply that the remaining 74% of the insights (four categories) were related to human reliability in some way. Please (a) provide specific examples of the insights from each of the five categories, (b) briefly discuss the contribution of human reliability to risk that is apparently a factor in the four of the five categories of the IPE insights.

32. Page 1—12 states that "the wetwell or drywell may be vented through either the Standby Gas Treatment ... system or *directly to the station chimney through the 8—inch "hardened" vent.*" (Emphasis added.) Pages 4—100 and 4—102 state venting with the Augmented Primary Containment Vent (APCV) system "is performed through the *18—inch vent dampers*" and that "the *exhaust duct* ultimately vents directly to the main chimney." (Emphasis added.) Further, pages 4—100 and 4—102 are replete with references with the words "damper" and "duct." (The drawings provided (Figures 4.2.1.16—1 through —3) do not clarify the issues.)

    (a) Is the APCV line(s) hardened (i.e. a pipe) from the drywell and wetwell to the chimney, i.e. no ductwork anywhere between the primary containment and the chimney? (Note: Figure 4.3—6b (page 4—126) shows a line entering the "Radwaste Ventilation (sic) Duct" upstream of the main chimney.) If ductwork is used, discuss how the ductwork was considered (assumptions) in the context of venting.

    (b) What is the size of the hardened vent line, 8— or 18—inches? There are no line sizes or duct/pipe identifications on the figures.

33. Page 4—123 states the SGTS is to be used "to control or maintain torus gas space pressure below 20 psig." What is the design pressure of the ductwork associated with the containment vent through the SGTS? Has this ductwork been tested at the design pressure to verify its leak tightness?

34. There is no discussion of any consideration of enhancing the reactor pressure vessel (RPV) depressurization system reliability, as per the CPI recommendations. Provide a discussion of the potential benefits of enhanced RPV depressurization capability and whether such enhancements will be made at Quad Cities.

35. The submittal states that the BWROG EPGs have been implemented into the Quad Cities EOPs and included in the IPE. Which revision of the BWROG EPGs?

36. Provide the conditional containment failure probability for "early," "late," "bypass," and containment intact and the definitions for "early" and "late."

37. Of the conditional drywell containment failure probability (77.4%), what is the percentage that has a flooded drywell?

38. There is an interconnection between the fire protection system and the RHR sytem (Figure 4.2.1.8-1). Is the procedure in place for using this interconnection? If not, when will the procedure be in place?

39. Provide a discussion as to why the reduced effectiveness of the suppression pool for containment flooded scenarios has a more negative impact on risk, than the potentional benefit of preventing reactor vessel failure.

40. Why were post-core damage recovery actions not evaluated as part of the IPE? Recovery of AC power, drywell sprays, and fan coolers have been considered as a part of several other IPE submittals.

41. Table 4.5.6-1 on Page 4-240 indicates that "DW liner is unlikely to melt through if there is water on the floor, but more likely if the DW floor is dry."

    (a) What use was made of the results from the MAAP sensitivity runs simulating drywell melt through (MAAP calculations Q930701A through D)?

    (b) What fraction of the core damage frequency was composed of sequences that involve a dry containment floor?

42. Section 4.7 indicates that 81 insights were obtained from the IPE.

    (a) Which, if any, of the insights are relevant to the containment systems and post-core damage phase of severe accidents?

    (b) Have any of the identified insights been used as a part of the EOPs?

    (c) Provide a summary of the insights (and their use in the EOPs, if any) for drywell flooding, Interfacing Systems LOCA (ISLOCA), NRC strategies, and containment performance (as grouped on Page 4-300).

43. CECo has decided not to pursue the external vessel cooling strategy (Section 5.3.2 on Page 5-11). What risk benefit model was used to justify the exclusion of this strategy? Address the observation that, given that more than 78% of the core damage sequences result in vessel breach and containment failure, there could be sufficient justification in pursuing accident management strategies that would prevent vessel breach.

44. Table 4.5.5-2 on Page 4-214 lists the sequences analyzed deterministically to characterize source terms for other, similar sequences. In that table, the source terms for the "MEFG" PDS group is used to characterize source terms for almost all other sequences such as LOCAs and transients. However, this PDS grouping has no containment failure and very low releases. PDSs that are represented by this

sequence such as TEFE and SEFB are not similar to this sequences and entail containment failure and radiological releases. Why was this PDS (MEFG, Sequence 2) chosen to represent releases for the sequences such as LEAB, LLAB, SEFB, SLFG, TEFB, and TEFG?

45. Section 4.3.3.2 lists and discusses a number of "unlikely" containment failure modes.

(a) A number of parameters are listed that impact containment pressurization during a DCH event on Page 4-137.

(1) What fraction of the core damage frequency involve vessel breach at high pressure?

(2) Have the uncertainty associated with these parameters been considered with respect to containment pressurization? If not, please explain.

(3) What plant-specific analyses were performed to determine that DCH was not a potential early containment failure mode?

(4) The statement is made on Page 4-137 that "the most significant means of preventing DCH" is the use of the ADS. However, for different sequences, the ADS may not be available. What is the conditional probability of ADS failure, given core damage?

(b) Page 4-138 indicates that ex-vessel steam explosions are not a threat to containment integrity.

(1) What is the basis for this conclusion?

(2) Were any plant-specific analyses performed to arrive at this position?

(3) What is the impulse capacity of the pedestal wall and the drywell structures?

(c) Page 4-138 indicates that core-concrete interaction (CCI) can be ruled out as a threat to containment failure.

(1) Describe the analyses performed to arrive at this conclusion.

(2) Was the uncertainty in the debris coolability by overlying pool of water taken into account in these models?

(3) Were the effects of non-condensible gas generation taken into account in these models?

(d) Why were breaks outside of containment sequences not considered as possible bypass sequences?

46. In the evaluation of source terms for the various accident sequences, what were the decontamination factors used for the suppression pool under saturated and subcooled conditions? Was credit taken for retention of fission products within the reactor building? If so, please discuss the decontaimination factor and its basis.

47. Was the probability of safety relief valve tail pipe vacuum breaker valve failures considered? How was this issue treated in the IPE?

48. Interfacing systems LOCA in RHR and core spray piping was treated in the IPE submittal, and a core damage frequency of 6.3E-10 per reactor year was calculated for containment bypass. How were other sequences involving breaks outside the containment considered, such as, HPCI, RCIC, and SLC line breaks?

49. Table 4.4.5-3 (page 4-182 of the submittal) provides a list of equipment, for which the analysis of equipment survival under severe accident conditions has not been performed. However, this list includes equipment important to back-end analyses, such as, RHR pumps, ADS valves, and drywell coolers. Was this equipment considered as a part of the equipment survival analyses?