I.

September 19, 1995

Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations (NA2I)
Florida Power Corporation
ATTN: Manager, Nuclear
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15760 W Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - INDIVIDUAL PLANT EXAMINATION (IPE) - CRYSTAL RIVER NUCLEAR GENERATING PLANT, UNIT 3 (TAC NO. M74401)

Dear Mr. Beard:

Based on our ongoing review of your IPE submittal, we have identified the need for additional information relating to the internal event analysis of your IPE, including the core damage frequency analysis and the containment performance improvement program. The enclosure summarizes our request for additional information.

We request your response within 60 days of receipt of this letter. This requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

If you have any questions regarding this matter, please contact me at (301) 415-1471.

Sincerely,

L. Raghavan, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure: As stated

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Florida Power Corporation

Crystal River Unit No. 3
Generating Plant

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ENCLOSURE

Request for Additional Information

Crystal River

1. Feed and bleed is an important procedure and is a backup method of decay heat removal when other methods (e.g. main and emergency feedwater) have failed. This procedure is mentioned in the IPE as contributing to the relatively low CDF. Please provide a discussion of the feed and bleed procedure used at Crystal River Unit 3 (CR-3) including identification of the valves primarily relied on for feed and bleed. What fraction of time is the block valve to the PORV closed and how is this accounted for in the model? Is the block valve credited if the PORV fails to reclose?

2. In NUREG-1335 the NRC staff requested that success criteria be provided by initiating event.

   a) Please provide all the success criteria listed by initiating event, the basis for these criteria (e.g. expert judgement, realistic calculation, FSAR, etc.) and whether they reflect the plant as-built, as-operated, at the time the IPE analysis was performed.

   b) Please provide the definition of core damage assumed in the analysis (e.g., peak cladding temperature).

   c) Please provide the bases for using the Oconee 3 success criteria for LOCA initiating events and for primary and secondary heat removal.

3. The short term decay heat removal function (i.e. heat removal via the steam generators (SG)) is not included in the small LOCA event tree. While the SGs will help in decay heat removal, they may also reduce the time to HPI initiation by half (for very small LOCAs), due to RCS shrinkage and inventory loss. Please explain why this function is not addressed for small LOCAs.

4. This question is related to the success criteria used in the LOCA event trees.

   a) Please provide the bases as to why containment heat removal systems are not needed for prevention of core damage after events such as large LOCAs. Past PWR PRAs have found that failure of containment heat removal in accidents can cause subsequent failure of core injection systems.

   b) Please provide the bases as to why core flood tanks are not needed in large LOCAs, even though the submittal states that the flood tanks are needed to prevent excessive peak cladding temperatures (page 23).

5. Please explain why the plant-specific initiating event data are taken only from the period 1987 to 1991.

6. The following questions concern common cause data:

    a) In the IPE submittal, data is provided for common cause failure of EFW pumps, even though one pump is turbine driven and the other is motor driven. Please explain why these pumps are susceptible to common cause failures.

    b) Data are missing for the common cause failure of main feedwater pumps, batteries, and breakers. Please clarify how such data are used in the model, or, if not modeled, the basis for exclusion.

    c) Please explain how the use of brackish water in the service water components will affect their common cause failure rate and how this is accounted for in the common cause data, the failure data, and in the model.

    d) It is not clear how the common cause data are estimated. The submittal refers to an analysis in an early version of NUREG-1150 (in NUREG/CR-4550, Vol. 1) which adjusts the beta factors in such a way that reported frequencies are assigned to the 95th percentile level instead of to the mean. The rationale is that the beta factor is the ratio of common cause to all reported failures on a particular component, and as independent failures tend to be under reported in the LERs (while common cause failures are not), the calculated beta factors are assigned to the 95th percentile in order to avoid overly conservative values. We found that the CR-3 IPE submittal table of common cause failures is consistent with a similar table which appears in this earlier version of the NUREG/CR-4550 report. This seems an overly non-conservative approach, and a discussion is needed as to why the 95th percentile is appropriate. Also it should be noted that a later version of the same NUREG/CR report shows higher values of beta factors while no mention is made of the adjustment method in this later version. Please explain why the beta factors used in the IPE are appropriate.

    e) In the common cause data, no distinction is made between failure to start and failure to run (e.g., of diesel generators, pumps). Is the same beta factor used for both failure modes? If so, please justify. Is common cause failure of the makeup pumps to start included in the model? If not, please justify.

7. In many PRAs reactor coolant pump (RCP) seal LOCAs are significant contributors to the CDF either as an initiating event or as a system failure consequential to another initiator.

    a) Please provide a discussion of the RCP seal LOCA model used.

b) Provide the probability vs. leakage rate vs. time data and a discussion of important test results.

c) Provide a description of operator actions and their timing following the loss of one or the other (or both) methods of seal cooling. Please indicate if any of these actions are not proceduralized and provide the basis for crediting any such non-proceduralized actions.

d) Is seal cooling isolated in accidents such as steam line break inside containment? If so, what are the operator procedures, and how is this treated in the model?

8. NUREG-1335, Section 2.1.6 part 4, requests "a thorough discussion of the evaluation of the decay heat removal function." Section 3.4.6 of the IPE, Decay Heat Removal Evaluation, does not provide specifics and insights on vulnerabilities of DHR systems. Please provide a discussion of insights derived for DHR and its constituent systems, and provide the contribution of DHR and its constituent systems (including feed and bleed) to core damage frequency and the relative impact of loss of support systems on the frontline systems that perform the DHR function.

9. Section 3.4.5, Vulnerability Screening, indicates no vulnerabilities at the CR-3 plant. Please provide the definition of vulnerability used in the screening process and a discussion as to why no vulnerabilities exist at CR-3 according to that definition.

10. It is not always clear from the IPE submittal whether the plant improvements described are being proposed for further consideration or were actually implemented. Please provide the following:

a) The specific improvements that have been implemented, are being planned or are under evaluation.

b) The status of each improvement, i.e. whether the improvement has actually been implemented, is planned (with a scheduled implementation date), or is under evaluation.

c) The improvements that were credited (if any) in the reported CDF.

d) If available, the reduction to the CDF or the conditional containment failure probability that would be realized from each plant improvement if the improvement were to be credited in the reported CDF (or containment failure probability). Alternatively, if available please provide the increase in the CDF or the conditional containment failure probability if the credited improvement were to be removed from the reported CDF (or containment failure probability).

e) The basis for each improvement, i.e., whether it addressed a vulnerability, was otherwise identified from the IPE review, was developed as part of other NRC rulemaking (such as the Station Blackout Rule), etc.

11. In flood scenario No. 1, mention is made of main feed water (MFW) and service water (SW) leakage frequencies, but it appears that only the SW frequency is used.

    a) Is there any MFW piping in this scenario? If so, why isn't the MFW leak frequency included (this frequency is much higher than SW leak frequency)?

    b) Why is the resulting accident scenario a spray and not a flood?

    Was consideration given to the spray effect of the fire suppression equipment actuation in the flood scenarios? If not, please justify.

12. No dependency matrix is included in the submittal. For instance, it is not clear if the PORV needs instrument air and if this is modeled. Please provide the dependency tables as requested in NUREG-1335.

13. Please provide a discussion and data about the recovery probabilities used for various types of loss of offsite power events.

14. NUREG-1335 requests that support system failures be incorporated into the model. Loss of HVAC has been shown at some plants to be an important contributor to the CDF. Please provide a discussion of the impact of loss of HVAC in rooms containing safety related equipment, including rooms with pumps, electrical equipment and the control room. The discussion should include the following: systems in the areas considered; basis for elimination, describing the method of assessment, including calculations and tests; credited operator actions, alarms, procedures and staged equipment. Please indicate whether consideration was given to equipment isolation at temperatures lower than those necessary to cause damage to the equipment. If this was not considered, please provide the basis for exclusion.

15. The following questions concern data:

    a) In Table 3.3-2, component failure data, there are generic and plant- specific failure data for component "motor driven pump" or "turbine driven pump." In other words, no distinction is made between various types of pumps (e.g., HPI vs. LPI) and thus they are all considered to have the same failure data. As there are differences among pumps in each class (different design, operating characteristics, environment, etc), one would expect them to be treated differently. Please provide the basis for using common data for different pumps. It should be noted that the common cause data (beta factors) used do distinguish among various types of pumps, as expected.

    b) Is the failure data for "relief valve fail to open" applicable to pressurizer PORV and SRVs? If not, which data are used for these valves. If yes, please provide the basis for using the same failure data for both the PORV and the SRVs. Please explain why there are no data for PORV failure to reseat after passing liquid.

c) In Table 3.3-2, some entries are apparently out of place, e.g., RCS cold leg LOCA (value 9.42E-6/ry) and common cause failure of motor driven pump (value 7.9E-7/ry). Please explain what these data mean and how these values were derived and verify that these are the correct values. For instance, the CCF value in this table could not be reproduced by applying the beta factors from Table 3.3-4 to the pump failure to run data; furthermore the CCF data is specialized by pump type, whereas the CCF entry in Table 3.3-2 is not. Please explain.

d) In Table 3.3-2, the plant-specific turbine driven pump failure rate to run is much lower (almost two orders of magnitude) than the generic failure rate specified in the CR-3 IPE. Furthermore, there is no specialization by pump type. It would be expected that the turbine-driven MFW pumps would have a much longer experience, and thus better plant-specific statistics than the turbine-driven EFW pump which is infrequently used. Moreover, these two types of pumps have different design, construction, capacities, control systems, etc. Please verify that:

   (i) the plant specific experience (from the stated data collection period) bears out the lower failure rate;

   (ii) all important types of turbine-driven pumps (e.g. main feedwater and emergency feedwater) have experienced this marked improvement in performance, and that calculating the plant-specific data from generic data for these different types of pumps results in essentially the same failure rate for both the MFW and the EFW turbine driven pumps.

e) Please describe the process used to derive plant-specific data and to insure that it formed the basis for the plant-specific failure rates used (e.g., was it some kind of a Bayesian updating calculation?).

16. The following is a question on LOCAs:

a) The LOCA frequencies are lower than those used in NUREG-1150. For instance, the medium LOCA frequency in the CR-3 IPE is lower by a factor of 2, while the large LOCA frequency is lower by a factor of 10. The very small LOCA is apparently not included in the CR-3 IPE submittal as an initiating event. In view of the fact that small and medium LOCA contribute 64% to the CDF, these initiating event frequencies are important. Please provide the bases for estimation of the LOCA frequencies used in the submittal.

b) Medium and small LOCAs contribute 64% to the total CDF. This is a much higher fraction than that reported for several other PWRs. Are there plant features which account for this? Please discuss the underlying reasons and provide any insights.

c) Please explain if loss of component cooling water due to a pipe break would be an initiating event and why it wasn't included in the initiating event analysis.

17. NUREG-1335 requests that the licensee consider ISLOCA and containment bypass sequences in their IPE. Please discuss any consideration given to ISLOCA and its contribution to the CDF. Show the high-low pressure boundaries, their locations, their failure modes and failure rates and the resulting ISLOCA frequencies. Show which scenarios would lead to bypassing of the containment and their frequencies.

18. It is not clear in the submittal if plant changes due to the station blackout rule were credited in the analysis. Please submit the following:

a) Report whether plant changes (e.g. procedures for load shedding, AC power) made in response to the station blackout rule were credited in the IPE and which specific plant changes were credited.

b) If available, give the impact of all of these plant changes on the total plant CDF and on the station blackout CDF (i.e. reduction in total plant CDF and station blackout CDF).

c) If available, give the impact of each individual plant change to the total plant CDF and the station blackout CDF (i.e. reduction in total plant CDF and station blackout CDF).

d) Report any other changes to the plant, separate from those strictly in response to the station blackout rule, that nonetheless may reduce the station blackout CDF. In addition:

- Report whether these changes are implemented or planned.
- Report whether credit was taken for these changes in the IPE.
- If available, discuss the impact of these changes to the station blackout CDF.

19. The submittal does not state the freeze date of the analysis or indicate whether any exceptions to the freeze date configuration of the plant were assumed in the analysis. The freeze date is defined as the date to

which the plant design, configuration, operations and procedures are represented by the analysis.

a) Please identify the freeze date of the analysis.

b) Describe any exceptions to the freeze date configuration.

c) If available, identify the impact of any freeze date exceptions on the CDF, both individually and collectively.

20. Please explain how the independent review was conducted, which areas of the PRA were reviewed, what were the results of the review, and how were the review questions resolved.

21. The IPE PRA was based on an earlier PRA from the late 1980's, with any plant changes accounted for in the new model. The old PRA was reviewed by Argonne National Laboratory in 1988, under contract to the NRC, in report NUREG/CR-5245 (published in 1989). This review found the original PRA to be of good quality on the whole, albeit with some areas for improvements noted, such that the original PRA CDF of 5.6E-5/ry was updated to 1.1E-4/ry in the ANL study. What are the specific reasons why the IPE CDF of 1.4E-5/ry is smaller than the original CR-3 PRA CDF of 5.6E-5/ry, or the Argonne review estimate of 1.1E-4/ry? Please state the changed assumptions, plant configuration and/or operating and emergency procedures, modeling differences, comparison and derivation of specific data and the bases for such.

22. Please provide the bases for using 1.0E-8/ry as the cutoff in sequence quantification (except for SGTR sequences). Most PRAs use 1.E-9/ry or 1.E-10/ry. The original CR-3 PRA used 1.E-9/ry. Please provide an estimate of the CDF value of the residual, i.e. the error from uncounted sequences and state how the estimate was computed.

23. The CR-3 IPE submittal documentation does not appear to address the significance of the human to cause, contribute to, and mitigate the consequences of an accident. In fact, Section 3.3.5 of the submittal consists in total of an eight (8) page section which includes two tables listing 86 different human actions (i.e., 31 latent and 55 dynamic human actions).

    Please provide a detailed description of the process that was used to identify and select the human actions evaluated using several examples from Tables 3.3-5 and 3.3-6 to illustrate this process.

    For pre-initiator (latent) human actions, please respond to the following:

    - How were miscalibration errors selected and treated?

    - How were failure to restore errors selected and treated?

- How were performance shaping factors (PSFs) determined and used? For example, Subsection 3.3.5.1 states "The value used for "psfs" was 0.1." However, in Table 3.3.5 it appears that PSFs of 1.0 and 0.0 were also used. Please also clarify these apparent discrepancies.

For post-initiator (dynamic) human actions, please also respond to the following:

- How were response and recovery type actions selected?

- How were human errors of omission considered and quantified?

- How were human dependencies selected and quantified?

- How were "available" & "required" times determined and used in Table 3.3-6?

- How were plant specific PSFs used? Include the process used to determine how these plant-specific factors were used to estimate post-initiator human reliability.

- Pertaining to the 16 important operator actions listed in Subsection 3.4.2.4, please provide a description of the process used to identify the important operator actions, including examples where appropriate. How was it verified that plant emergency procedures, design, operations, and maintenance and surveillance procedures were examined and understood prior to development of the modeled operator actions?

24. It is not clear from the submittal how the screening process was utilized to help differentiate the more important pre-initiator ("latent") human actions. Table 3.3-5 lists the thirty-one latent human errors used in this IPE. To simplify the data management, all of the specific latent events were related to one of the generic events, as described in the submittal.

    Since a screening process was used, please provide:

    a) The rationale for the screening process used,

    b) A discussion of how the analysts assured that important human events were not erroneously eliminated, and

    c) How the analysts assured that the potential contribution of the human events eliminated was negligible.

25. In Section 3.5.3, for Flood No. 10 the IPE submittal states that "Using a TRC [Time-Reliability Correlation] estimate for the probability that a flood is not isolated within 115 minutes yields a non-recovery factor of 6.8E-06." If it is assumed that the CDF is really the HEP in

Figure 3.3-1, then the above number could be interpolated, but only for the situation awareness, situation assessment, and response planning. However, Table 3.3-6, which lists the dynamic human error events used in the PRA, indicates that all of the 55 post-initiator (dynamic) basic events listed have human error probabilities between 1.0E-04 and 3.7E-01.

Please explain the use of the 6.8E-06 for a non-recovery factor and why it is not listed in Table 3.3-6. Noting that this HEP is smaller than all 55 reported in Table 3-3.6, please discuss the impact on CDF of this relatively low value.

26. RCP seal LOCAs can be significant to CDF either as initiating events or as system failures consequential to another initiator. Table 3.3-6 of the submittal shows the RCP seal LOCA related human "basic event," i.e. "crew fails to trip RCPs," with an HEP of 1E-02 based on TRC, Rule-Based (No Conflict) within 10 minutes. If Figure 3.3-1 was used (and assuming "HEP" is the correct ordinate label) interpolation of the figure's curve gives an HEP in the range of 3E-02 to 5E-02, which is 3 to 5 times higher.

Please explain the apparent differences between Table 3.3-6 and Figure 3.3-1. Explain why other RCP seal reliability related dynamic human actions (e.g., potential need to isolate seal return from the control room) are not included in Table 3.3-6. Please identify and discuss the significant proceduralized dynamic operator actions and clearly document their human reliability quantification.

27. The plant description in the CR-3 IPE submittal (Section 4.1) is very brief and does not include any figures or tables. This is not consistent with the guidance of NUREG-1335 for plant data and plant description (Section 2.2.2.1), which requests that, in addition to the appropriate narrative explanations and sketches, the plant data should be summarized in tabular form. More detailed guidance is contained in Appendix A of NUREG-1335 (Step 1).

The plant data and plant description provided in an IPE submittal should identify and highlight component, system, and structure data that may be of significance in assessing severe accident progressions, and additional consideration should be given to equipment whose operability is desired during exposure to harsh environments. Examples provided in NUREG-1335 include the cavity design and water availability in the cavity, characterization of the reactor vessel's lower head, flow paths within and out of the reactor cavity, containment geometry and compartmentalization, and the effects of debris aerosols and particulates on the operation of the sprays and fan coolers. Please provide the requested plant data and plant description information as described in NUREG-1335. The information provided should cover the example items listed above.

28. One important plant feature that affects severe accident progression is the configuration of the reactor cavity. It affects the phenomena associated with high pressure melt ejection and ex-vessel debris coolability. It is stated in the IPE submittal that "The reactor cavity configurations for Surry and Zion are quite different from the CR-3 configuration, and the SANDIA/BNL results for Surry and Zion require careful interpretation." (Section 4.5.2, pg. 266). However, this information is not provided in the IPE submittal. Please provide drawings and a more detailed discussion of the reactor cavity, and discuss how the SANDIA/BNL results were interpreted for the CR-3 analysis.

29. The evaluation of isolation failure is discussed very briefly in Section 4.1.2 of the IPE submittal (less than one page). Isolation failure is considered possible only for station blackout sequences. Containment isolation failure is one of the parameters used to define a PDS. The containment isolation failures considered in the IPE include both large and small isolation failures and the isolation failure of the reactor building sump. The probability of isolation failure for SBO sequences (Sequence 7 in Table 4.3.4 of the submittal) is about 2.7%. However, how this value is obtained is not discussed in the IPE submittal. Please provide a more detailed discussion on containment isolation failure (refer to Section 2.2.2.5 of NUREG-1335) and discuss any findings on containment isolation failure related to the five areas identified in Section 2.2.2.5 of NUREG-1335.

30. The relative position of the reactor vessel and the once-through steam generators (OTSG) put CR-3 into the "lowered-loop" type of B&W PWR designs. According to the IPE submittal, natural convection flow paths through the heat exchanger tubes cannot develop in a straight tube heat exchanger, such as the B&W OTSG, as long as the loop seals remain in place. Because of the loop seal, induced SGTR is not considered in the IPE as a potential failure mode. Only natural convection is considered; the probability of induced SGTR due to forced circulation caused by the restart of the RCPs is not addressed in the IPE. However, for some B&W plants, the Inadequate Core Cooling (ICC) guidelines call for the RCPs to be restarted. Continuous operation of the RCPs would clear the seal and cause the high temperature gases to be transported to the SG potentially inducing a SGTR. Please discuss whether there is any mechanism, such as the restart of the RCPs, that may cause the clearing of the loop seal and a creep rupture of the SG tubes for CR-3.

31. Selected sequences for Key Plant Damage States are analyzed using the MARCH3 and CONTAIN 1.1 computer codes. The analysis results provide the times of major events occurring during accident progression and the data for fission product releases. Except for the SBO sequences, secondary cooling is assumed to be available for all other sequences in the analyses. The availability of secondary cooling will affect the RCS conditions and timing of accident progression. Since secondary cooling (the availability of steam generator cooling) is not a PDS parameter, its status is unknown for a PDS sequence. The assumption on secondary

cooling needs justification. Please provide the justification for the availability of secondary cooling for non-SBO sequences and discuss the impact of this assumption on containment event tree evaluation and source term definition.

32. Section 4.6 of the IPE submittal deals with source term characterization. The source terms are obtained by computer code calculations for the five sequences selected to represent the five Key PDSs. Except for the Key PDSs associated with containment bypass and isolation failure, the containment failure modes (i.e., timing and size) that are used in the calculations for the other Key PDSs are not provided in the IPE submittal. The containment failure mode is an important parameter in the determination of environmental releases. For the same Key PDS, a different containment failure mode may result in significantly different environmental releases. Since different containment failure modes may occur for each Key PDS, the use of a single source term, which is calculated based on an assumed containment failure mode, may not be sufficient to characterize the possible source terms associated with the Key PDS. For example, Key PDS (KPDS) K6BA, a small LOCA event, may have either early or late containment failure. Although the source terms for the different failure modes may be significantly different, only one source term is calculated in the submittal for Key PDS K6BA. The release fractions for the volatile fission products (I and Cs) are predicted to be 1E-7 for this Key PDS. This is significantly smaller than the release fractions predicted in other PRAs for sequences with early or late containment failure. Please discuss and justify the containment failure modes used in the calculation of the release fractions presented in Section 4.6. Please also discuss how the assumed failure mode(s) provide a sufficient and adequate characterization of the source terms for the Key PDSs.

33. Tables 4.6.6-1 to 4.6.6-7 of the submittal present the source terms for the Key PDSs. The release fractions presented in these tables include those for your best estimates (50th percentile) as well as those for the 5th and 95th percentiles. It is not clear from the IPE submittal how these uncertainty ranges are obtained. The discussion presented in Section 4.6.6 for the phenomena that cause uncertainties in source term estimate is more qualitative than quantitative. Please provide a more specific discussion on the prediction of the uncertainty ranges and the basis used for the prediction.

34. It is stated in Section 4.7.3 (Containment Phenomenological Event Tree Quantification) that "CR-3 specific analyses of the accident progression were performed for the dominant sequence in each Key PDS, using the MARCH3 code and the CONTAIN code. In addition, certain sensitivity analyses were also performed. These analyses and results are discussed in Section 4.6." It is not clear from Section 4.6 which sensitivity analyses were performed because calculation results are presented in Section 4.6 for only five MARCH3 calculations, one for each Key PDS. Please provide a table listing all sensitivity analyses that were

performed for CR-3 using the MARCH3 and CONTAIN codes and discuss the use of these results in the quantification of the CR-3 IPE.

35. Hydrogen burns prior to vessel breach are not likely to challenge the containment integrity but may consume a significant fraction of hydrogen before vessel breach and thus reduce the probability of a large hydrogen burn later. The probability of an early hydrogen burn is determined in the IPE based on the hydrogen generation predicted by code calculations for selected sequences. It is assumed in the IPE that even though the predicted containment atmospheric conditions cannot support a global hydrogen burn, a standing flame is possible if ignition occurs at the location of the discharge (e.g., the break location of a small LOCA). It is noted that the requirement of an ignition source (i.e., a spark) is not considered in the IPE in the determination of a hydrogen burn. Since the ignition source is required near the location where the hydrogen burn is expected, an ignition source may not be readily available. Please discuss the requirements and the availability of ignition sources for early hydrogen burns. Include in the discussion the impact an increased probability of large hydrogen burns after vessel breach would have on containment performance.

36. The probabilities of global burns (top event HD) and the probabilities of consequential failures are provided in Table 4.7-5. It shows the probability values assigned to the various combustible gas concentration ranges, and, for each concentration range, the probability of the containment not being inerted, the probability of ignition, the probability of burn, the probability of containment failure if a burn occurs, and the probability of containment failure. However, the values presented in this table are not discussed in the submittal, i.e. what they represent and how they are derived. For example, in the derivation of HDA, the probability of a burn for a combined hydrogen and carbon monoxide concentration in the range of 4% to 6% is assigned a value of 0.02895, although the probability of a containment atmosphere in this concentration range is assigned a value of zero. Also, for each concentration range, the total probability of ignition is not equal to the summation of the probabilities of ignition occurring at different concentration levels. Please define the values presented in Table 4.7-5 and discuss the derivation of these values.

37. Although the Containment Performance Event Tree (CPET) quantification involves the use of assumptions and data that have significant uncertainties, the IPE does not provide a sensitivity study, as is requested by NUREG-1335. Please provide a sensitivity study addressing the parameters that are likely to have the largest effect on the likelihood on the time of containment failure and the magnitude of the source term. Use Table A.5 of NUREG-1335 and the results from other PRAs as guidance for selecting sensitivity parameters.

38. In the CR-3 IPE, containment pressure capabilities are evaluated for containment temperature conditions ranging from 300°F to 800°F. The failure modes considered in the analysis do not seem to include the

failure of penetration seals. Please describe what consideration was given to the effect of prolonged high temperature on penetration elastomer seal materials. Particular attention should be paid to seals in areas where standing flames are possible.

39. The CPI recommendation for PWR dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures). This issue is not specifically addressed in the CR-3 IPE submittal. Please identify potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided. Please specifically address how this information is used in the assessment of hydrogen pocketing and detonation. The discussion should cover the likelihood of local detonation and potentials for missile generation as a result of local detonation.

40. Some of the potential containment failure modes identified in NUREG-1335 are not discussed in the CR-3 IPE submittal. These include missiles or pressure loads due to steam explosion, and vessel thrust force due to blowdown at high pressure. Please discuss the effects of these containment failure modes on the CR-3 IPE results.

41. One parameter used for PDS definition in the CR-3 submittal is "The pressure inside the pressure vessel, at the time that the debris melt-through the vessel" (pg. 215 and Table 4.3-3). Since RCS depressurization prior to vessel breach due to hot leg creep rupture is considered in the CPET, the pressure fixed at the PDS stage can only be the RCS pressure at the onset of core melt. To avoid confusion, should the parameter used for PDS definition be the pressure <u>at the beginning of core melt</u> instead of <u>at the time of vessel melt-through</u>? Please clarify this and explain how depressurization prior to vessel failure was accounted for.

42. Figures 4.6.5-1 and 4.6.5-2 seem to be duplicate figures. Both of them show the primary system pressure. The discussion in Section 4.6.5 implies that Figure 4.6.5-2 should show the water level in the reactor vessel, but this is presented in the submittal as Figure 4.6.5-3. From the discussion presented in the IPE, it seems that the figure for containment pressure is missing. Please provide the missing figure and the correct number of the figure.

The discussion in the first paragraph of Section 4.6.5 regarding these figures is also confusing. It is stated in this section that "Figure 4.6.5-1 shows an increase in the system containment pressure due to the loss of BS (defined in the submittal as reactor building spray) followed by a reduction as the steaming rate slows in the vessel." It is not clear what is "system containment pressure" and why the loss of BS affects the primary system. The statement "Figures 4.6.5-1 and 4.6.5-2 show the immediate reflooding of the reactor vessel, and reduction in containment pressure, due to safety systems injection of water" also needs clarification because containment pressure is not shown in either figure.

43. Tables 4.6.6-1 to 4.6.6-5 present the source terms for the five sequences selected to represent the five Key PDSs. These tables are discussed in Section 4.6.1 through 4.6.5. However, the sequences that are used to generate data for Tables 4.6.6-6 to 4.6.6-7 are not discussed in the submittal. Please provide a brief description of these two sequences and discuss their significance in source term definition.

44. It is stated in the IPE submittal that "More detailed documentation of these accident progression models, data, analyses, and results are contained in Section 4.1, 4.2, 4.6, and 4.8." Section 4.8 cannot be found in the IPE submittal. Please clarify. It is mentioned in many places in the IPE submittal that information can be obtained in the Level 2 appendices. However, they are not included as part of the submittal. Please provide the relevant appendices.