October 12, 1994

Mr. W. R. Robinson, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165 - Mail Code: Zone 1
New Hill, North Carolina 27562-0165

Dear Mr. Robinson:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REVIEW OF INDIVIDUAL PLANT EXAMINATION SUBMITTAL - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M74418)

By letter dated August 20, 1993, you responded to Generic Letter (GL) 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities - 10 CFR 50.54(f)." The NRC staff has reviewed the submittal and finds that the additional information identified in the enclosure is needed to enable us to complete our review. The list of questions is related to the internal event analysis of the Harris individual plant examination, including front-end, back-end, and human reliability analysis, and the containment performance improvement (CPI) program.

We request the response be provided within sixty (60) days of receipt of this letter.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you have any questions, please contact me at (301) 504-1458.

Sincerely,
Original signed by:

Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure: Request for Additional Information

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Mr. W. R. Robinson
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REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION FOR THE SHEARON HARRIS NUCLEAR POWER PLANT

GENERAL 1 Please provide a discussion of core damage prevention strategies for interfacing LOCA and excessive LOCA.

GENERAL 2 The IPE submittal discussed how the licensee devoted considerable effort in peer review of the IPE. However, no details of the peer review were included in the submittal. Please provide a discussion of any significant results of that peer review, such as, areas where the final version of the IPE was modified from the draft as a result of peer review comments, as suggested by Section 2.4 of NUREG-1335.

GENERAL 3 Please provide a discussion of the plant changes made after the freeze date that are credited in the IPE models, other than the change for non-vital dc power, and discuss the importance of these changes on the overall CDF. Also discuss the status of any improvements credited in the IPE or planned for the future.

F.E.1 Please provide the success criteria (i.e., core temperature, water level, etc.), and a discussion of the codes used in the IPE to determine if the reactor core went to a core damage state.

F.E.2 The submittal indicates that HVAC for safety related rooms was eliminated from inclusion in the model, as an initiating event or as a support, by comparison to HVAC requirements for the most limiting areas of the plant, or by taking credit for compensatory actions to achieve some means of temporary cooling for the CSIP rooms and the EDG rooms. It was also stated that these last two systems had independent HVAC systems and that, therefore, failure cannot occur due to a single event. The existence of two independent trains of HVAC and the ability to take compensatory action for loss of HVAC may not be sufficient to eliminate areas from being modeled in the search for vulnerabilities. Consideration must be given to (a) frequency of common cause failure of fans, dampers, chillers, and chilled water pumps, (especially for continuously running equipment), as is done in other independent systems with pumps and valves. In addition, consideration must be given to (b) the existence (or lack) of alarms and indication to alert the operators to a failure, (c) the timeliness of the alarms, (d) the rate of temperature rise, (e) the sensitivity of equipment in the room (especially instrumentation), (f) analysis to verify that opening doors is sufficient, and (g) the existence of staged equipment, procedures and operator training. NUREG-1335 indicates that written procedures should exist for recovery actions. Actions to recover failed HVAC systems are such. The above items were not addressed in the submittal. Provide an example of the process by which HVAC, for safety related equipment including the control room and electrical equipment rooms, was eliminated by the comparison approach. Discuss all the other above items (a through g) for those systems eliminated because of consideration of compensatory operator action.

F.E.3 The success criteria identified in the submittal for containment cooling during a large LOCA is cooling of 1 RHR heat exchanger; this is substantially less than the FSAR minimum containment cooling requirements which call for: 1 spray train, 1 RHR heat exchanger, and 2 fan coolers. Discuss the impact on successful operation of the pumps for the following: (a) adequacy of net positive suction head available (NPSHA) for RHR and containment spray pump(s)(CS) taking suction from the containment sump with no containment cooling, (b) the temperatures reached for RHR, CS, HPSI and CCW pumps, and the impact on pump operation, with RHR taking suction from the sump with only one RHR pump and one RHR heat exchanger in service, (c) effect of loss of containment sump water due to evaporation, on adequacy of NPSHA from the sump to the RHR and CS pumps without FC, and (d) has plant specific data for the efficiency of the CCW heat exchangers been factored into the analysis. If not what impact would it have on pump temperature, NPSHA and pump operation.

F.E.4 Please discuss your consideration of the impact of high containment pressure on the ability of the PORVs to remain open during accident conditions, due to lack of differential pressure across the actuator, during manual feed and bleed. Also discuss the EQ related effects on the PORV actuator at sustained, elevated temperature over the long term with minimal containment cooling provided by 1 RHR heat exchanger. Other IPEs have indicated that the failure of the PORVs increase significantly during long term feed and bleed.

F.E.5 With regard to RCP seal LOCA:

(a) Please identify and provide a discussion of the RCP seal LOCA model used in your analysis. Include in your discussion, the timing of RCP seal degradation, the probability of failure and seal flow rates.

(b) The submittal states that 2 minutes are available to trip RCPs given loss of all seal cooling before an RCP seal LOCA develops. Is it expected that a seal LOCA will also develop if the RCPs are not tripped in a short time on loss of cooling (CCW) to the RCP motors and is this included in the model?

F.E.6 With regard to flooding:

(a) The discussion of flooding indicates that all buildings except the reactor auxiliary building (RAB) and the diesel generator building (DGB) were initially eliminated from further consideration, and that upon further application of the screening criteria only 15 RAB areas were retained. 1) What was the criteria used to screen the areas in the RAB and the DGB from further consideration? 2) Of the 15 considered further, what was the criteria used in the second screening to eliminate those not considered from that point on (if based on frequency, identify)?

(b) Other plants have found that rupture of expansion joints has a

higher frequency than the double ended pipe rupture that has been considered in the flooding analysis. 1) Since expansion joints in the ESW are open to the NSW (due to system crossties), how were these failures considered in your analysis? 2) The description for the flood for the SW tunnel indicates that the operators will isolate the NSW and start the ESW pump, which are "reasonably expected responses to the event." On what basis will the operators decide that this is the appropriate action to take to start a pump with a known pipe break? 3) It is not clear that the maximum pipe break is the worst case, if lesser size breaks, even with less flow, would not cause indications to exist that would precipitate operator action in a timely manner. How was this addressed in your analysis?

(c) Other IPE submittal for PWRs have concluded that spray related failures can dominate the CDF from internal flooding initiating events. Identify the criteria (e.g., location of the equipment in relation to the break, the distance beyond which spray/impingement is of no concern, etc.) used to eliminate failure of components due to spray or impingement.

F.E.7 Only one event tree is used to model all transient initiating events. It is not clear how this event tree differentiates between, and models, overcooling and undercooling transients using the same top events, or provides insights for detecting vulnerabilities, since there are no results presented for various events such as feedwater pipebreak, steam line break (upstream and downstream of the isolation valves). There is no description of these events in the submittal. Please provide a description of these important events, including how the transient event tree addresses the following: (a) isolation of the bad SG from the good SGs, (b) control of feedwater to the bad SG, and (c) boration following a main steam line break inside containment to prevent excessive power.

F.E.8 The submittal states that feed and bleed without operator action is successful given HPSI actuation on high containment pressure, since the pumps are of sufficient discharge pressure to open the safety valves, but this was not credited in the IPE. A dominant sequence described in the submittal involves failure of operator action to initiate feed and bleed cooling. Please discuss the reasons, in your search for vulnerabilities, for not taking credit in the IPE for the ability to feed and bleed without operator action when it appears that the frequency of a dominant sequence could be reduced by doing so.

F.E.9 The submittal indicates that only pipe breaks are considered credible events which would cause a total loss of SW. It is not clear that this is an appropriate consideration of other failures in the system. Since the ESW and the NSW pumps are different, it is indicated that instantaneous common cause failure of all the pump are eliminated from consideration. However since they are different, it is possible upon failure of the operating NSW pump to have the standby NSW pump fail independently (to start or run)

and for the 2 ESW pumps to fail due to common cause unique to these pumps.

Based on data for common cause and independent failure to start of the ESW pumps provided in the submittal, the following sequence could be an important contributor to core damage.

Loss of NSW X {CCF of both ESW pumps to start} = CDF

0.11/yr X {0.064 (beta fact.) x 0.002 (fail to start)}= 1.4E-5/yr

This is the CDF due to RCP seal LOCA from this sequence, since loss of SW causes loss of charging (CSIP) and loss of CCW. RCP seal LOCA cannot be mitigated due to loss of HHSI. It appears that this could be a significant contribution, since the current reported CDF is 7E-5. Other configurations not considered here include: either one of the ESW pumps out for maintenance and independent failure of the standby pump to start, or some additional common cause or independent failure of other equipment such as valves. Discuss this sequence and address why it should not be considered in the submittal. Include other failures as discussed above which would increase the frequency of this sequence.

F.E.10 The submittal implies that following a transient only one pressurizer PORV will be challenged to open, thus only one PORV is modeled as failing to close leading to a small LOCA. What percentage of transient initiating events require a PORV to open? What is the basis for assuming that only one PORV and not more will be challenged to open, given a common setpoint for the PORVs? What is the impact of this assumption on CDF?

F.E.11 How were failures in the reactor protection system and the control rod drive system modeled, and what is the probability of reactor trip failure following a transient?

F.E.12 Did the IPE address a break in the steam supply line to the turbine driven auxiliary feedwater pump as an initiating event? In some plants, this break renders the AFW pumps in the room with the break unavailable due to EQ failures.

F.E.13 The submittal identifies that plant specific failure and maintenance and test data were used for only the pumps for seven specific systems, and the EDG. " Other PRAs" and LERs were used to determine if plant specific data should be collected. The importance of components is highly plant specific (based on configuration, operation and data) and other PRAs can only be indicative, not definitive. Discuss the basis for excluding the use of plant specific failure and maintenance data for possible important components (valves in the suction and discharge of pumps, check valves, PORVs, air compressors, strainers in the ESW discharge, breakers, inverters, batteries, fans) in your search for vulnerabilities, for which operational and or test data exists.

F.E.14 The submittal indicates that there were no cost effective hardware or procedural changes, other than the changes for non-vital dc power, based on the criteria for the accident sequences or classes identified by the IPE. Therefore, any and all sequences with a frequency greater than 1E-6/yr are to be addressed in the severe accident management guidance. Is this correct? Since there was no discussion of the criteria for vulnerability as requested by NUREG-1335, please address the criteria used to define "cost effective" and discuss its application to the core damage sequences identified on pages 3-243 and 3-244 of the submittal.

F.E.15 The submittal indicates that loss of the non-vital 125 V dc power was a dominant contributor before procedural changes were made, and that it contributed approximately 50% of the total CDF. Before the changes, did CP&L consider this a vulnerability, and what criteria was used to make the determination? Were any other options considered?

F.E.16 The IPE submittal presents the contribution to CDF from the initiating event groups, but does not present other methods of identifying the contributors which may provide insights or identify concerns in your search for vulnerabilities. Please discuss any insights gained and provide the contribution to CDF identified for the following:

a. All the initiating events (including individual transients such as loss of vital and non-vital dc power)
b. Loss of Offsite Power(non-station blackout), RX coolant pump seal LOCA, loss of high pressure injection.
c. Systems (top 10) contained in the cutsets for the sequences leading to core damage (including support systems)
d. Common cause failure
e. Maintenance

F.E.17 There is no description of the LOSP and SBO events provided in the submittal. Please provide a discussion of these important events including the success criteria, timing, load shedding, and effect on the battery life.

F.E.18 Section 3.4.3 (DHR Evaluation) indicates that the DHR contribution to CDF has been considered with all other insights in Section 3.4.2. While the submittal indicates that the contribution for loss of DHR for transients is approximately 11%, it does not address the contribution from loss of DHR in total, nor does it provide insights into the relative contribution to CDF for its separate constituent systems. In addition, Generic Letter 88-20, Appendix 5 indicates that support systems are important to the DHR function and suggests that they be considered in the search for DHR-related vulnerabilities.

Therefore, provide a discussion of insights derived 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 front line systems that perform

the DHR function.

F.E.19 The submittal identifies operator refill of the RWST for continued injection as a success criteria for SGTR, however no system and no operator action is identified for the refill. What system is to be used and how is it and the operator action modeled in the IPE?

B.E.1 The combustion of CO is not addressed because the SHNPP is composed of quartz aggregate concrete, which is said to produce very little CO. However, review of results from other plants with a basaltic concrete basemat show the generation of CO is not negligible. Does the quartz-based aggregate in the Shearon Harris plant lead to even smaller non-condensible gas generation than basaltic concrete? Please provide additional details of results from the calculations regarding the generation of CO, including the results from sensitivity calculations, together with the concrete composition (including cement type).

B.E.2 The submittal does not explicitly address the recommendations of the Containment Performance Improvements (CPI) Program as they pertain to large dry containments. Please address these recommendations, including:

a. Equipment vulnerable to the effects of hydrogen combustion, including combustion from the effects of localized pocketing. Include discussions of key systems such as fan coolers and containment sprays (local H_2 burns were neglected because it is believed that the cavity gas temperatures would be about 500°F, which is below the MAAP auto-ignition temperature of 1310°F);

b. The effects of local detonations which may also result from pocketing; and

c. The extent to which walkdowns were made to determine that hydrogen would not concentrate in localized sub-compartment.

B.E.3 For MCCI calculations it is assumed that the debris will uniformly spread. However, it is possible that a large coherent mass released from the RPV will stay in a clump, spread slowly and provide a large amount of localized MCCI as compared to the IPE analysis. The submittal acknowledged that gravity pours can lead to melt accumulation under the RPV. Please discuss the extent to which calculations were performed to study the resulting MCCI and non-condensible gas generation.

B.E.4 The models for MCCI seem to consider pool decay heat as an energy source, neglecting the ex-vessel condensed phase reaction of molten metals with Zr, which can produce large exothermic energy (according to your description of basic event HTSTUFF in Table 4-25 and the associated fault tree). To what extent were the exothermic reactions considered in the MCCI models and in the fault trees?

B.E.5 Containment structural capacities are said to be weakly dependent on temperature, however no analysis was provided to support this assertion. What was the effect of temperature on containment structural capacity? To what extent was the fragility curve generated for higher temperatures used in the analysis?

B.E.6 Listed below are several apparent discrepancies. Please clarify the following inconsistencies.

- Discrepancies exist between the results reported in Table 4-26, and the results given in the discussion of CET quantification results. For example, in discussing late containment failure, a frequency of 6.7×10^{-7}/year is given. The discussion states the PDS 1P "represents about 92% of the total frequency". However, using the value given in Table 4-26 for damage state 1P, this probability is actually $6.4 \times 10^{-7}/6.7 \times 10^{-7} = 95.5\%$. Results for the early containment failure show similar discrepancies, with a stated frequency for PDS 15A of 1.8×10^{-7} providing about 92% of the contribution. Using the value given in Table 4-26, the contribution is 88.9% ($1.6 \times 10^{-7}/1.8 \times 10^{-7}$).

- In Table 3-36 the total value for Small-2 LOCAs is given as 3.3×10^{-5}, but the values in the table sum to 2.3×10^{-5}.

- On page 7-7 of the submittal, it is stated that RC-4 is the dominant release mode, and radionuclide scrubbing does not occur because the secondary side of the steam generators do not contain water. However, Table 4-28 (page 4-186 of the submittal) states that RC-4 is scrubbed.

- In the executive summary (page 1-13), it is stated that the probability of early containment failure conditional probability is 0.10, but the overall conditional containment failure probability is 0.15. Also in the executive summary, Fig. 1-3 indicates that isolation failures and early failures are both 0.

- Figure 1-1 in the submittal states that internal flooding contributes 5% to overall CDF. However, the CDF reported due to internal flooding is 5E-6/R yr and the total CDF is 7E-5/R yr, implying that internal flooding is 7% of the total CDF.

B.E.7 This question deals with the quantification of the basic events in the submittal. The probabilities for the basic events in the fault tree are listed in Table 4-25 (page 4-170) of the submittal. The basis for the probabilities assigned for many basic events is qualitative, and the qualitative values assigned varied from 0.001 to 0.9.

(a) Discuss why the same value for failure of the hot leg due to thermal effects was assigned for station blackout, seal LOCA and small break LOCA sequences.

(b) Please provide a justification for the value of 0.1 assigned to "PRZFAILS", the basic event describing the failure of the surge line at elevated temperatures. Shouldn't the probability be comparable to that for the failure of the hot leg?

(c) The IPE assigned a value of 0.1 to basic event "INGEOMOP", failure of in-vessel cooling due to debris geometry. Shouldn't this value be dependent on the extent of accident progression and time at which ECCS was recovered?

(d) In connection with the calculations performed to determine the value of 0.003 obtained for CFAIL1 and CFAIL2, the probabilities of containment failure calculated for hydrogen burn, why was the same value assigned for both low and high steam concentrations?

(e) Please discuss how the value of 0.5 was determined for "SGTUBEF2", the basic event describing the probability of steam generator tube rupture after RCP pumps are turned on. This is an important quantification in terms of its effect on source terms.

(f) It is stated in the submittal that there is a higher probability of U-tube rupture when the steam generator secondary side is depressurized. Why does the submittal assign an apparently low value of 0.1 to "SGTUBEFD", the basic event describing the probability of SGTR with RCP pumps on and the secondary depressurized?

(g) The submittal assigns a value of 0.1 for SRV failure in a ruptured steam generator. Since this is an important basic event with a large impact on source terms, discuss how either data from the open literature or plant-specific data were used in determining this value.

B.E.8 Please expand Table 4-26 (page 4-174 of the submittal) to list the outcomes (containment failure or intact containment) for all PDSs, so that a complete C-Matrix can be generated for this plant.

B.E.9 CP&L identified an important mode of radiological release resulting from induced steam generator tube rupture caused by RCP restart procedures following core damage (pg. 1-13). Owing to the relatively high consequences of this action and the relatively small benefits gained by it, has the licensee considered or identified any procedural changes to inhibit RCP restart and thus prevent SGTRs?

B.E.10 The submittal does not discuss the effects of elevated temperatures upon the performance of containment penetrations, including those with elastomer seals. Please discuss the analyses performed to evaluate these temperature effects upon seals, electrical penetrations, and other containment penetrations.

B.E.11 Although fault trees were developed in order to quantify the probability of containment isolation failure, none were provided as part of the submittal. Please provide and discuss a representative fault tree used in this process.

B.E.12 The IPE provides the six PDSs with the highest frequencies and lists the corresponding breakdown of containment failure modes. However, the breakdown for all PDSs according to release classes could not be determined. If available, please provide the PDSs for each sequence in Table 3-36, the conditional probabilities of the release classes associated with each PDS, and the frequencies of these release classes.

H.R.A.1 The treatment of pre-initiator human errors appears to be reasonably comprehensive in scope, in that both restoration and miscalibration errors are addressed, and fifteen errors were quantified and retained in the model. However, the submittal discussion of the quantification process is very abbreviated. Were there pre-initiator errors screened out via a numerical screening process? If so, what was the numerical screening process used, or was it the ASEP pre-initiator screening model? Please identify which pre-initiator HEPs were quantified using ASEP, and which via THERP. Select several examples covering both miscalibration and misalignment, quantified by each method and provide details of the quantification including: the basic HEP selected and the basis for its selection; the plant-specific performance shaping factors evaluated and how they were evaluated (e.g., plant-walkdowns, interviews with operators/maintainers, procedure reviews, etc.); and evaluation of potential dependencies in restoration or calibration errors due to such factors as performance by the same crew, performance at the same time, training or procedure deficiencies, common cause miscalibration of instruments.

H.R.A.2 NUREG-1335 requests information on plant specific factors used to estimate HEPs, including time for operator response. The submittal notes that for the SHNPP analysis, a time line was developed for each cognitive action assessed, and that the time lines were used to verify that the action could be performed and to justify credit taken for recovery actions. The submittal also notes that simulator exercises performed specifically for the SHNPP HRA "provided information on the amount of time needed to go through various procedures". However it does not state clearly the basis for specific time estimates, and provides no specific results or insights from the simulator exercises. The time available for each action is tabulated in the submittal, but estimated actual performance time is not provided. Please identify the estimated operator response times, the method (e.g., timed walkdowns, simulator runs, etc.) used to obtain the estimates, and the point (e.g., receipt of alarm) at which the estimated response time and available time starts.

H.R.A.3 Simulator exercises can be a powerful source of information to support the HRA. Please identify the human interactions for which

simulator exercises were used to support the quantification of the HEPs. Provide a brief summary of the structure for conducting those exercises (e.g., did they involve currently licensed operators in unrehearsed scenarios, what data collection means were used); and the results and insights from the simulator exercises conducted to support the SHNPP HRA.

H.R.A.4 In quantifying post-initiator human errors the values for p_e were obtained using tables from the THERP Handbook. The submittal states that "The most commonly used were Table 20-7 for errors of omission and Table 20-12 for errors of commission. Recovery factors were applied where appropriate, and THERP or fault trees were employed as necessary to determine the overall HEP." The submittal states that once risk significant human interactions had been identified walkdowns were performed, including verification of accessibility of equipment and transit times for local actions outside the control room. No other details are provided regarding the consideration of plant-specific or sequence-specific performance shaping factors or recovery factors, nor regarding the actions taken to support the quantitative values selected. Therefore, it is difficult to assess the plant-specific application of the THERP technique. Similarly, the summary discussion of the implementation of the EPRI methodology to estimate p_c does not provide enough detail to assess the plant-specific application of the methodology.

Using the example HEPs listed below:

a) discuss the application of THERP to estimate p_e; include in the discussion 1) the basic HEP values selected and the basis for the selection, 2) the plant-specific performance shaping factors applied, including how potential dependencies were addressed 3) the plant specific assessment performed to evaluate/confirm credit for reducing the value of the basic HEPs, and 4) the THERP trees or other representation used to breakdown and assess the overall action.

b) discuss the EPRI methodology to estimate p_c; include in the discussion 1) the causal mechanisms considered for each example HEP and 2) a brief summary of the plant specific and sequence specific evaluations supporting the implementation of the respective EPRI decision trees tree in the EPRI cause-based model were evaluated on a plant-specific and sequence-specific basis.

<u>Example operator actions:</u>

a. OPER-1 Operator fails to establish recirculation (LHSI, Large LOCA).
b. OPER-3 Operator fails to implement feed-and-bleed.
c. OPER- 9 Operator fails to initiate RCS cooldown to use LPSI (SB LOCA).
d. OPER-17 Operator fails to establish recirculation (HHSI).
e. OPER-21 Operator fails to establish shutdown cooling.

f. OPER-42 Operator fails to align CSIP suction for SI
g. OPER-43 Operator fails to align offsite AC (NV BAT no output).
h. OPER-R1 Operator fails to locally align offsite power (late, no DC).
i. OPER-15 Operator fails to trip RCPs on loss of seal cooling.

H.R.A.5 The submittal states that credit for recovery of offsite power was based on "industry data" but provides no details on the sources of data, or the modeling of equipment recovery or human actions to effect this recovery. Please discuss the model/data used for recovery of offsite power as they were applied to the SHNPP analysis.

H.R.A.6 We assume that walkdowns of the "risk significant" actions outside of the control room included recovery actions. Please confirm and provide a brief discussion of the conduct and the results of these walkdowns.

H.R.A.7 Section 3.1.1 of the submittal includes a brief discussion of the internal flood initiating sequences. Several important operator actions are identified. The submittal indicates that flood sequences included "full consideration" of human errors, but provides very few details. Please identify specific operator actions credited in the internal flood sequences, the procedures in place to effect the necessary actions to mitigate the flood, and provide a complete, but concise, discussion of the HRA method used to quantify those actions.

H.R.A.8 Section 3.4 of the submittal discusses the sequences which would have been above the 1.0E-06 cutoff for reporting were it not for credit taken for operator actions, and indicates that individual sequences were reduced by credit for operator actions. a) Discuss the process by which these results were generated (i.e., were all the operator actions HEPs set to 0.1 simultaneously and the whole model rerun, was only one operator action HEP value at a time set to 0.1 and the model rerun, was a reduced model used, based on the results of the initial results, etc.). b) The discussion of each sequence identifies the dominant human action; failure to initiate cooldown and depressurize the RCS, and failure to provide RWST makeup, being two of them. What consideration was given to potential human performance related enhancements for these actions, to reduce the value of the HEPs, or to ensure that the original lower estimates are valid, or to perform a more rigorous analysis to reduce uncertainties associated with the HEPs? c) It is not clear that the recovery actions listed were included in this re-analysis. Were they? If not, why were they excluded?

H.R.A.9 The submittal lists the post-initiator response HEPs in order of their risk significance, but it does not identify the basis for this significance, nor does it provide the ranking based on risk significance of the pre-initiators or the recovery actions. It would be of significant insight to know what the contribution to CDF of the HEPs is, such that procedural or other enhancements for

important operator actions could be identified. Please: a) identify the basis for the risk significance (i.e., importance, risk reduction worth, risk achievement) and the values used to order the HEPs listed, b) provide similar values for the pre-initiator and recovery actions identified in the submittal, and c) What conclusions or insights were obtained from the IPE regarding the importance of human error to plant risk? For example, are there sequences in the results in which human error is a significant or dominant contributor? If so please identify which sequences and which HEPs.

H.R.A.10 Our review identified at least two potentially significant operator actions credited in the CETs: 1) OPER-IV, Operator fails to manually depressurize RCS which allows LPI flooding, and 2) OP-H2REC, Operators preclude hydrogen burn following recovery of containment heat removal. Please identify any other operator actions credited in the CETs. Identify the HEPs and provide a concise discussion of the quantitative process. If the same values were used in the CETs as were used for that action in the front-end analysis, please discuss the potential impact of the post-core melt recovery situation vs. pre-core melt response situation as it impacts operator performance.