Docket Nos. 50-266
and 50-301

April 7, 1994

Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, Wisconsin 53201

Dear Mr. Link:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING NRC REVIEW OF THE POINT BEACH INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NOS. M74452 AND M74453)

While conducting its review of the Point Beach IPE submittal and its associated documentation, the staff has determined that additional information is required to complete the review. The enclosed list of questions are related to the internal event analysis in the IPE, the containment performance improvement (CPI) program, and the proposed resolution of Generic Issue 23, "Reactor Coolant Pump Seal Failur[illegible]

Please provide written responses to the enclosed questions within 60 days of receipt of this letter. We request that you send a copy, as soon as possible, of the phenomenological issue papers developed by Fauske and Associates, Incorporated (FAI) for addressing the treatment of severe accident phenomenology in the Point Beach IPE, in order to expedite our back-end review.

This request for information affects fewer than 10 respondents; therefore, OMB clearance is not required under Public Law 96-511.

Should you have questions, or would like to extend the period needed for your response, please contact me at (301) 504-1373.

Sincerely,
ORIGINAL SIGNED BY

Richard J. Laufer, Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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See next page

Mr. Robert E. Link
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

Ernest L. Blake, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mr. Gregory J. Maxfield, Manager
Point Beach Nuclear Plant
Wisconsin Electric Power Company
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
Two Rivers, Wisconsin 54241

ENCLOSURE

QUESTIONS ON POINT BEACH INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL

Point Beach IPE Review Front-end Questions

1. Point Beach is a two-unit site with several shared systems. For example, the two onsite diesel generators are shared; thus, station blackout (SBO) affects both units simultaneously. While the submittal contains information on the guidelines for modeling dual unit interactions, it does not identify any initiators as dual unit, even though there are shared systems, nor does it identify the success criteria for the station as a whole which it states were considered. Please identify the station success criteria and the impact it has on consideration of dual unit initiators, identify the possible dual unit initiators and the frequency of such dual unit initiators, discuss how they were derived or on what basis they were eliminated, and address their possible impact on core damage frequency (CDF) for both units simultaneously. In addition, please provide the basis for the assumption that suction faults were not considered credible since "there is sufficient amount of water in the forebay after the circulating water pumps trip to supply the service water system for both units for 24 hours," if the traveling screens get plugged and collapse.

2. (a) The use of a system fault tree that includes the support systems, as an initiating event appears to be comprehensive, but is relatively unique. In addition, it is difficult to put in perspective the overall expected frequency of these events. Please provide the total frequencies of the following initiating events which were addressed in this manner but were not provided in the submittal: loss of instrument air, loss of bus D01, loss of bus D02, loss of component cooling water (CCW), and loss of service water (SW).

   (b) In addition the submittal indicated that complete loss of DC power was not included as an initiating event because it was of low frequency. Please provide the frequency of this event and the criteria used to eliminate events of low frequency.

3. Please provide the criteria for core damage in terms of water level and peak cladding temperature, used as the basis for success criteria in your analysis, and the computer code used to determine if core damage had occurred.

4. Please provide the basis for the following two assumptions: (a) Success of auxiliary feedwater (AFW) during an anticipated transient without scram (ATWS) is 200 gpm to each of two steam generators. (b) No containment cooling is required to prevent core damage for any sequence in which energy is released to containment. For assumption (b), specifically address the following two issues: (i) Adequacy of net positive suction head available (NPSHA) for residual heat removal pump (RHR) pump(s) pulling from the containment sump with no containment

cooling. (ii) The temperatures reached for RHR pumps and CCW pumps, relative to design temperatures, with RHR pulling from the sump with only one RHR pump and one RHR heat exchanger in service.

5. (a) The value of 40,000 gallons used for the volume of the condensate storage tank is beyond the technical specification requirement of 13,000 gallons. It is not clear from the data presented in your discussion that 40,000 gallons is always available. Is it considered to be always available, or is it modeled as a probability? Since significant credit is taken for this value, what requirements are in place to assure that the 40,000 gallons will be available to make this critical assumption viable?

   (b) Your submittal indicates that the batteries used to supply power to the instrumentation for operation of the turbine-driven auxiliary feedwater (TDAFW) pump may last for more than 1 hour (possibly 4 hours) subsequent to SBO. Please provide a discussion of the basis (test or calculation) for this critical assumption, or describe why this assumption is not critical.

6. For some of the entries in Table 3.3-1 of your submittal, the failure to run probability is not equal to: (failure to run frequency) * (run exposure time). For example, the failure frequency for an SI pump is listed as 1.6E-5 per hour, and the fault exposure time is listed as 24 hours; (1.6E-5) * 24 = 3.8E-4. However, the probability for an SI pump failing to run for 24 hours is given as 7.1E-4. Please explain the reason for the higher failure probability.

7. Since credit is taken for the air cooling of the charging pumps, describe the impact on the pumps of the loss of heating ventilation and air conditioning (HVAC) to the area where they are located, and its influence on the estimated CDF.

8. Section 3.4.6.2 (USI A-45, "Decay Heat Removal (DHR)") indicates that the DHR insights described in Generic Letter 88-20 were considered in the Point Beach IPE, that the IPE analysis is capable of identifying severe accident vulnerabilities due to loss of DHR and that the reported CDF is due almost entirely to the loss of DHR capability. However NUREG-1335 requests a <u>thorough discussion of the evaluation</u> of the DHR function. Section 3.4.6.2 does not address DHR as an entity nor does it provide insights into the relative contribution to CDF of DHR totally or for its separate constituent systems or for its support systems. Therefore, since significant work has been done on Point Beach, provide a discussion of insights derived and provide the contribution of DHR and its constituent systems (including feed and bleed) to CDF and the relative impact of loss of support systems on the frontline systems that perform the DHR function.

9. In the table presented on page 64 of 121 in Section 3.3 it is indicated that "...at least 10 feet of space is present between systems. Therefore, no credible spray scenario was identified in which more than one safety-related system would be disabled." Please provide the

criteria, including frequency, location, and type of equipment considered used to determine if spray, splashing or dripping was considered for further analysis or eliminated as a concern. The distance of 10 feet specified in the above statement from the submittal appears to present a very limited sphere of influence for the effects of spray from piping, or other equipment. Provide the technical basis for the use of 10 feet as the distance beyond which spray is not a concern, and discuss the sensitivity of your analysis in your use of this criteria.

10. As indicated by NUREG-1335, please identify the components for which plant-specific data was used to differentiate them from those for which only generic data was used.

Point Beach IPE Review Human Reliability Analysis (HRA) Questions

1. Your discussion of the process used to identify potentially important pre-initiator human errors is very general. Please discuss the process used to assure potentially important errors were not missed. For example, what analysis was performed of maintenance and test procedures? Was there a systematic review of support systems focused on identifying human error contributions to unavailability? Were HRA, systems analysts, maintenance, and operations personnel involved in the process?

2. The HRA did not address calibration errors. Other probabilistic safety assessments (PSAs) account for the impact of calibration errors on system unavailability. Please provide the basis for not considering this type of human error in the Point Beach analysis using plant-specific information.

3. Effective implementation of the EPRI decision tree methodology requires a thorough knowledge of both HRA and plant design/operations and an in-depth assessment of human performance *in context*. Please discuss the interaction of HRA specialist(s) and system analyst(s) in the plant-specific analysis of post initiator human errors, including identification of potentially important errors, screening, and qualitative assessment of performance shaping factors, error recoveries, and dependencies. Discuss involvement of operations/training staff in the assessment. Provide examples of documentation - e.g., worksheets, structured interview formats - used in the qualitative and quantitative assessments.

4. Identify the screening values used for post-initiators and the basis for assurance that they were effective values, i.e., that all important human errors were *screened in*.

5. The presentation of quantitative results from the post-initiator analysis is quite thorough and detailed. However, discussion of methodology is presented at a very general level. It is not possible, for example, to trace through the calculation of specific human error probabilities (HEPs) based on information presented in the submittal,

even supplemented by the EPRI reference. Please provide a small set of representative examples of the quantification of specific HEPs using the EPRI decision methodology. In the example discussions, please describe the information sources or other bases used to answer the questions posed as tops in the EPRI decision trees.

6. (a) A number of human error recovery factors are identified in the discussion f post-initiator human errors (page 8 and 9 of 121). Please provide specific references or other basis for each type of human error recovery factor applied to post-initiator HEPs. Please identify which human error recovery factors were applied to Pc (probability of failure to initiate the correct response) and which to Pe (probability of failure to execute the response correctly).

   (b) It is noted in the submittal that recovery factors were applied only when there was sufficient time for the operator to get feedback from the plant and correct the error. Provide the basis for determining sufficient time and the means by which operator response time was estimated.

7. Clarify the discussion of the first type of dependency, i.e., dependency among "elemental HEPs" that make up Pc. Provide examples of the application of these dependency rules in your analysis.

8. Provide specific references or other basis for the dependency guidelines for the second type of dependency, i.e., among different Type C (post-initiating event intera tion) events within the same cutset. Provide specific examples from the HRA of the application of these guidelines. The submittal identifies eleven cutsets (Table 3.3.3-3) for which multipliers were added to account for dependencies among multiple HEPs. How was it determined that these 11 cutsets are the only ones to which the dependency factors apply?

9. HEPs in sequence recovery actions appear to have been estimated based essentially on subjective judgment and plausibility arguments, rather than on application of any identified HRA technique used in the Point Beach analysis. Please provide the basis for this approach in general, and provide specific technical bases for the judgments where possible. In particular, address the three identified sequences which were reduced by more than an order of magnitude due to recovery actions.

10. Your submittal discusses the exception to the general statement that no credit is taken for operator action after core melt, i.e., the SBO sequence in which offsite power is recovered within six hours. What is the impact on release fraction of assuming guaranteed successful operator action to recover fan coolers and SW system subsequent to recovery of power after SBO?

11. The flooding analysis includes estimates of HEPs that appear to be based on judgment or plausibility arguments. What plant-specific assessments

were made to realistically model and quantify the flooding analysis, especially for the large break in the auxiliary room and for the small SW break in the cable spreading room?

12. Your conclusion that the high contribution of human error to CDF is expected because Point Beach is an older plant, with less automation than other plants, is an interesting insight. Please provide specific examples of operator actions other than manual switchover to recirculation that illustrate this conclusion.

13. In Table 3.3.3-2 of the submittal, the value for basic event HEP-RHR-EOP13-23 (failure to align for low head sump recirculation) for large loss of coolant accident (LOCA) is identified as 0.1. Please explain why this value is significantly higher than the nominal value (9.7E-3) used for all other LOCA cases. In your explanation, please discuss important aspects of the HRA and related equipment involved.

14. The model for mitigation of an unisolated steam break assumes that injection of borated water with safety injection (SI) is not required. The discussion of operator actions on page 185 of Section 3.1.2.19.3 indicates that without boration, operator actions must be taken to control cooling with AFW, but it doesn't appear to be modeled. Please discuss how these actions were incorporated in the model and whether they are proceduralized.

15. In your discussion of using the batteries to power the instrumentation for the TDAFW pump operation for more than 1 hour (possibly 4 hours) subsequent to SBO, it was indicated that the operators can likely successfully operate the TDAFW pump "blind." Is the HRA basic event (HEP-AF-ECA00-XX) for this operator action assumed to be with instrumentation or "blind?" Please provide the basis for this critical operator action (e.g., equipment used, procedures, physical conditions such as lighting, etc.).

Point Beach IPE Review Back-end Questions

1. In light of the NRC review of the IDCOR/IPEM methodology (letter dated November 22, 1988, from the NRC to W. Rasin, NUMARC), please discuss how your methodology is different from the IDCOR/IPEM methodology, and how your study addressed the concerns that resulted in the staff finding the IDCOR/IPEM methodology unacceptable.

2. Section 4.5.1, page 32 of Section 4.0, states that, because of the conservatism of 100 percent of the core being expelled from the vessel, "debris coolability can be assumed, if an adequate supply of water exists to ensure that the debris in the reactor cavity is submerged in a water pool throughout the accident."

   Per Generic Letter 88-20, this assumption requires a uniform spread of debris on the cavity floor resulting in a debris depth of less than 25 cm. Have you considered the effects of non-uniform spread of debris?

If you have, please describe your treatment and consideration of non-coolable geometry.

3. Section 1.4. page 9 (of Section 1.0) of the submittal notes that the calculated fission product release fraction (FPRF) is conservative, since (1) all core damage sequences are assumed to go to complete core melting through the reactor vessel lower head, (2) credit is generally not taken for any operator actions or equipment recoveries following the onset of core damage.

   If the vessel did not fail as a result of external cooling (for example, because of cavity flooding), a steam generator tube rupture (SGTR) may occur. The magnitudes and the frequencies of the fission product releases for this event may be higher than the case with vessel breach. In light of this, please explain how this phenomenon was considered in the IPE.

4. Section 2.4.2, page 9 (of Section 2.0) of the submittal, states that your IPE team reviewed eight IPE submittals and four PRA studies. Please provide any significant insights relevant to Point Beach gained from performing these reviews.

5. (a) Please describe what consideration was given to the effect of prolonged high temperatures on penetration elastomer seal materials.

   (b) Section 4.1.1, page 2 (of Section 4.0) of the submittal, notes that all equipment hatches employ non-metallic gaskets as part of their leakage barrier. Please discuss implications of using non-metallic gaskets.

6. Section 4.4.1, page 16 (of Section 4.0) of the submittal, states that, although the mean calculated containment failure pressure was used for this analysis, the result would be unchanged even if the more conservative approach of using the 5-percent lower bound value of 154 psia were applied. Please provide your rationale for this statement.

7. Section 4.4.2, page 18 (of Section 4.0) of the submittal, notes that "MAAP code runs predict that the operation of either a single containment fan cooler, or single train of RHR operating on recirculation, would prevent containment failure due to overpressurization." What is the role of containment sprays as a means to prevent containment failure?

8. Section 4.4.2, page 19 (of Section 4.0) of the submittal, notes that a critical containment failure leading to significant fission product releases occurs through a line "greater than 2 inches in diameter" that penetrates the containment. What is the basis for not selecting lines 2-inch in diameter or less? What source term is associated with a release through a 2-inch line?

9. Section 4.4.3, page 20 (of Section 4.0) of the submittal, notes that "Plant-specific analysis were performed for a station blackout core

damage sequence using the worst case assumptions for hydrogen production." Please list these assumptions.

10. Section 4.5.1, page 28 (of Section 4.0) of the submittal, notes that "HPME will be averted if the operator initiates cooldown and depressurization using the secondary side." Please characterize the credit taken for operator depressurization in the analysis.

11. Section 4.5.1, page 32 (of Section 4.0) of the submittal, states that long-term containment response is not affected, regardless of whether the containment is cooled using the containment spray pumps or the RHR pumps, and regardless of whether the cooling occurs before vessel failure or immediately afterward.

    Please describe the effect of the different means of containment cooling on the Point Beach source term (i.e., discuss any credit taken with respect to source term reduction).

12. Please discuss what source term calculations were performed for interfacing systems LOCA sequences.

13. Please provide the following:

    - Phenomenological issue papers that support important assumptions in the Level 2 analysis.

    - Letter from Dr. Fuller to Mr. Ed Mercier of Wisconsin Electric dated March 18, 1993 (Section 5.2, page 5 (of Section 5.0) of the submittal), which documented the results of an independent technical review of the phenomenological position papers.

    - Letter from Mr. Marc A. Kenton to Mr. Ed Mercier of Wisconsin Electric dated June 8, 1993 (Section 5.2, page 5 (of Section 5.0) of the submittal), which documented the results of an independent review of the Level 2 analysis.

14. With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, page 2-11) states that "the analyses should address the five areas identified in the Generic Letter, i.e., (1) the pathways that could significantly contribute to containment isolation failure, (2) the signals required to automatically isolate the penetration, (3) the potential for generating the signals for all initiating events, (4) the examination of the testing and maintenance procedures, and (5) the quantification of each containment isolation failure mode (including common-mode failure)." Please discuss your findings related to the above five areas.

15. Section 4.6, page 36 of Section 4.0, states that, because the combination of containment event tree (CET) top event success and failure states leading to a particular CET endstates is largely predetermined by the Level 1 sequence definitions, "the split fraction for each CET branch (top event) can readily be assigned as 0's or 1's."

If the CET endstates are predetermined from the front-end analyses, what benefit is achieved by development of CETs?

16. Section 4.4.3, page 20 of Section 4.0, notes that the use of mechanistic models for debris dispersal "show the resulting pressurization to be less than the value necessary to challenge containment integrity."

    How is the term "challenge" defined?

17. Section 4.4.3, page 21 of Section 4.0, notes that the core-concrete interaction model "uses empirical parameters determined from available experimental data." Please identify these parameters and corresponding experiments, and give the parameter values.

18. Section 4.5.1, page 27 of Section 4.0, notes that, based on a review of NRC guidelines and previous work, two CETs for early and late containment failures for Point Beach have been developed. Please cite specific references for these guidelines and the previous work.

19. Section 4.5.1, page 28 of Section 4.0, notes that the determination of high-pressure failure is based on a reactor coolant system (RCS) pressure of 875 psig. This value was determined "based upon the 'cutoff' pressure for direct containment heating (DCH)."

    Please cite a specific reference for the value of this cutoff pressure, and its bases.

20. Section 4.7.2, page 37 of Section 4.0, describes the auxiliary building as a fission product barrier. What DF is associated with this barrier?

21. Section 4.6, page 35, states "PBNP is expected to have a flooded cavity at the time of vessel failure during nearly all accident scenarios analyzed for the PSA. This is due to the fan cooler condensate drains, refueling cavity drains, and general area drains in containment discharging to this region."

    For an accident scenario in which no water from the primary system is released to the reactor cavity, is the reactor cavity still flooded because of the water from the above mentioned sources? If not, how is it considered in your analysis?

22. (a) Provide a concise discussion of how your IPE process treated equipment survivability during a severe accident scenario.

    (b) Have you identified any essential equipment which would fail as a result of severe environmental effects? How is it determined which pieces of equipment (qualified for design basis accident (DBA) environments) will be useable and assumed to operate in severe accidents? How was credit for such equipment taken in your analysis?

23. (a) Have plant walkdowns been performed to determine the probable locations of hydrogen released into the containment? Including the use

of walkdowns, discuss the process used to assure that: (i) local deflagrations would not translate to detonations given an unfavorable nearby geometry, and (ii) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

(b) 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 your assessment of hydrogen pocketing and detonation. Your discussion (including important assumptions) should cover likelihoods of local detonation and potentials for missile generation as a result of local detonations.

(c) Has a calculation been performed to determine whether the containment can withstand a global hydrogen burn given the maximum amount of hydrogen production including the effects of core concrete interaction? Please discuss this aspect of the analysis.

24. Describe briefly how plant-specific insights (including candidates for back-end improvements) were obtained from the back-end analysis, and discuss how the back-end insights were or will be used to enhance plant safety.

REQUEST FOR ADDITIONAL INFORMATION ON GI-23, "REACTOR COOLANT PUMP SEAL FAILURE"

1. Section 3.1.3.1.3 (regarding loss of CCW) states that charging pumps do not require CCW for cooling and that they are adequately cooled by ambient air. Air cooling of charging pumps resolves one of the more significant concerns of GI-23. However, to better understand the dependencies, the staff seeks the following information:

   (a) Is air cooling normal mode or backup mode of cooling?

   (b) Is there, and if so what is the significance of, dependency on room cooling?

   (c) What impact, if any, does automatic containment isolation have on RCP seal cooling due to isolation of CCW as a consequence of this actuation?

   (d) What, if any, instrumentation are available for operator guidance to detect and mitigate RCP seal failures during station blackout?

2. Section 3.1.3.1.4 (regarding loss of CCW) and Section 3.1.3.4.4 (regarding loss of SW) of PBNP IPE submittal state that in case of these transients, even if the seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump. What is the basis of this assumption?

3. Section 3.4.6.3 of the submittal indicated that probabilistic RCP seal LOCA model of W (WCAP-10541, Revision 2, November 1986) was utilized with some modifications to address the NRC concerns due to the "binding" and "popping" modes of failure. It is recognized that the W model has been reviewed and found acceptable by the NRC contractor (AECL). However, the NRC has applied the W model with considerable differences in the probability of occurrence of each failure mode. In your modifications to the RCP seal LOCA model, please provide the probabilities used for the "binding" and "popping" modes of RCP seal failure.

4. In reference to the Section 3.4.6.3 of the submittal, does your RCP seal failure model consider O-ring in the failure event tree? If so, please provide the corresponding probabilities of O-ring failure at failure paths of the event tree.

5. In reference to the Section 3.4.6.3 of the submittal, please provide the basis and the assumptions for various amounts of RCP seal leakage in the failure event tree. If these assumptions are based on some testing, provide the details of those tests.

6. Section 3.4.6.3 also mentioned that the loss of RCP seal cooling is quantified based on final results from WCAP-10541 in an exponential model of core uncovery due to seal failure as a function of time from loss of seal cooling. The model used for non-recovery of off-site power is based on the curve, { $G(t) = 0.61\ e^{-0.391t}$ }. This curve agrees with typical non-recovery data for times of about an hour or later, but it cannot be correct at early times. For example, the curve indicates a probability of non-recovery of 0.61 at time 0; the probability must be 1 at time 0. How do you account for probabilities at times earlier than 1 hour? Please provide the basis for this condition and discuss the effect this condition may have on CDF due to RCP seal failure.

7. In reference to Appendix 3.1.4.A of the submittal, the core uncovery model is based on { $y(t) = ae^{bt}$ }. This is a curve fit with only two data points. Innumerable curve fit can be assumed to fit two points. The deficiency of this curve fit is obvious at the beginning of the curve, where y(t) should be zero at t=0 since the RCP seal LOCA does not occur instantaneously at the time of loss of RCP seal cooling. But according to your equation y(t) is not zero at time 0.

   Provide justification for why the curve fit did not consider at least three points and how do you account for the consequences of the error in the curve. Please provide the rationale for how the use of a simple exponential curve fit, based on only two points is a satisfactory representation of the actual Westinghouse data.

8. The model for RCP seal LOCA presented in Appendix 3.1.4.A of the submittal which provides the probability of a RCP seal LOCA as a function of time from loss of RCP seal cooling was used in modeling the responses to station blackout. Why was it not used in modeling the responses to loss of RCP seal cooling from other events (e.g., loss of CCW, or loss of SW)?

9. The probability of seal LOCA at time t presented in Appendix 3.1.4.A of the submittal is given by equation, { $P(t) = ae^{3} e^{bt}$ }. This is a wrong simplification of an integral function. The correct simplification is { $P(t) = ae^{3b} e^{bt}$ }. This would change the results substantially. Please review your integral function and its simplification and provide the corrected results and the implications of the corrections.

10. Please answer the following questions related to modeling of RCP seal LOCAs:

(a) How did you address a vibration-induced RCP seal LOCA due to loss of CCW motor bearing cooling and failure to trip the operating RCPs?

(b) To derive an equation for the probability of a RCP seal LOCA as a function of time, the model assumes that following a RCP seal LOCA, core uncovery occurs after three hours. If this assumption were consistent with the equations used for y(t), y(t) would have to be zero for at least 3 hours. How consistent is this with Westinghouse data (is it zero for three hours) and discuss the impact this has on the CDF due to RCP seal failure.

(c) In the equation for PROB-SL, the upper limit of integration is 'i-3'; is this supposed to be t_i-3, where t_i is the sequence-specific time for core uncovery without offsite power if no RCP seal LOCA occurs? PROB-SL is based on the assumption that core uncovery from a RCP seal LOCA occurs 3 hours after the RCP seal LOCA; thus, RCP seal LOCAs within the time interval (t_i-3, t_i) do not contribute to core uncovery within time t_i. However, a RCP seal LOCA within the time interval (t_i-3, t_i) does affect the time to core uncovery by increasing the mass loss from the primary system; in effect, the RCP seal LOCA reduces time T. The model does not address this effect since it seems to exclude consideration of RCP seal LOCA within (t_i-3, t_i). One conservative way to address this issue without varying T is to use T instead of (T-3) as the upper limit of integration in PROB-SL. Please address this issue.

(d) The values for PROB-SL given in Appendix 3.1.4.A on page 297 do not agree with the values for PROB-SL given in the data base of Table 3.3.1. Which values are correct?