Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, AR 72801

December 1, 1995

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - UNIT 1 INDIVIDUAL PLANT EXAMINATION (TAC NO. M74376)

Dear Mr. Yellverton:

The Individual Plant Review (IPE) submittal for Arkansas Nuclear One, Unit 1 is under active review by the NRC. Several questions related to your submittal have been generated by our reviewers. These questions are included in the enclosed request for additional information. Please respond to these questions within 60 days if possible. Communicate with your NRC project manager to develop an alternate response date if it is not possible to meet the requested schedule. This requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

ORIGINAL SIGNED BY:
George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure: For Additional Information

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

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Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

cc:

Mr. Harry W. Keiser, Executive Vice
President & Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Ms. Greta Dicus, Director
Division of Radiation Control
and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205-3867

Mr. Nicholas S. Reynolds
Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Mr. Robert B. Borsum, Manager
Rockville Nuclear Licensing
B&W Nuclear Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Mr. Jerrold G. Dewease
Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Mr. Robert B. McGehee
Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

Request for Additional Information
Arkansas Nuclear One, Unit 1
Individual Plant Examination Submittal

1. The following question concerns the treatment of initiating events in the submittal:

a) The plant model does not include the "Loss of Vital 120V AC Buses" as a potential initiating event. Please provide the basis for screening out this initiator, including any plant specific data and models used.

b) Your initiating event frequencies for "Loss of (125V) DC Bus D01 or D02" and for the "Loss of (4.16kV) AC Buses A3 or A4" are equal (IE=3.94E-4 per year), but they are about 10 times lower than the NUREG/CR-4550 recommended value of 5E-3/yr. In addition, the table of initiating event frequencies indicates that the frequencies for these initiators were taken from generic data. Please explain how these numbers were calculated, what was the source of data, and if plant specific fault trees were constructed for that purpose. Please provide justification for your assumptions and the data used. If available, provide an estimate of the impact on the results of your assumptions and data.

c) According to the submittal, small LOCAs are important contributors to the ANO-1 core damage frequency, contributing about a third to the total CDF. Therefore, the number used for the small LOCA initiating event frequency is important.

Your small LOCA initiating event frequency, which also includes spontaneous RCP seal failures, appears low (5E-3/yr). An explanation is provided which states that the Byron Jackson design is different than that of the Westinghouse RCP seals, that there have been changes in procedures and maintenance practices, and that the events that did occur (including the one at ANO-1 in 1980) are not indicative of seal failure frequency today. It is also stated that the ANO-1 event was too small to be counted as a LOCA. However NUREG/CR-4550 Vol. 3 Pt. 2 App. D quotes a leak rate of 400 gpm for that event, clearly above the normal makeup rate. While giving credit for improvement of the seals over the years, and including very small pipe and other component failures, NUREG/CR-4550 arrives at a generic "very small break" LOCA frequency of 1.3E-2/yr. In addition, the small LOCA frequency quoted in that document is on the order of 1E-3. Thus, the frequency used in the submittal is about 1/3 of that recommended in the NUREG/CR-4550.

Therefore, please provide details of your consideration of small LOCA frequency, including its constituent parts: pipe breaks, component leakages, RCP seal failures and inadvertent open relief valve failures. Include sufficient detail for an understanding of the basis for your initiating event frequency number. Also, please provide a more detailed justification as to why the ANO-1 event of 1980 should not be counted as a plant-specific event.

d) In addition, the large LOCA frequency appears low (1E-4), which also includes the "medium LOCA." NUREG/CR-4550 gives 5E-4 as the recommended frequency for large LOCAs, and 1.E-3 for medium LOCAs. Please describe how you arrived at the large LOCA initiating frequency.

e) The LOSP (loss of offsite power) frequency of 0.036/yr is at the low range of typical LOSP frequency values in, for example, industry data from NSAC-147. It should be also noted that ANO-1 is situated where severe weather phenomena are relatively frequent. Furthermore, LOSP sequences are important at ANO-1, according to the submittal, contributing more than 1/3 to the total CDF from internal events. Therefore, the LOSP frequency will directly influence a major portion of your results. Please explain how the LOSP frequency was estimated. Include in your discussion how plant-specific information and data were accounted for, including the weather related events.

f) Pipe breaks in the CCW and SW systems have been shown in other studies (e.g. NUREG-1150) to be important contributors to the SW and CCW loss frequencies. Were pipe breaks modeled as part of your initiating event analysis for loss of service water and component cooling water? How were they modeled and what was the effect on your results considering that these would be potentially irrecoverable events? Also, how were they included in, and what was the impact on, your flooding analysis? If pipe breaks were not modeled, please justify this omission.

2. The ANO switchyard is shared between Units 1 and 2. Please identify any maintenance activities performed during shutdown of ANO-2, in the switchyard, that may impact ANO-1 initiating events frequencies, e.g., increased LOOP frequency for ANO-1 due to ANO-2 switchyard fault, and the potential unavailability of the AC backup provided by ANO-2 diesel generators. If such activities were considered, please discuss how they were modeled. If they were not included please provide the basis for not considering them, and, if available, provide an estimate of the impact on your results.

3. The common-cause failure data used in the plant model is listed in Table 3.3-5 of the submittal. A review of the listed components indicates that the list may not be comprehensive; e.g., the following types of components are missing:

Circuit breakers, relays (ESFAS)
Electrical switchgear, transmitters
Air operated valves, switches
Check valves
Fan cooler units
Ventilation fans
Air compressors
Inverters

4. The common cause table (Table 3.3-5) does not include the "failure of all service water pumps to start" (e.g., following the loss of normal power) failure mode. Provide the basis for omitting start failure events of these pumps (or of other pumps) in the failure and common cause analysis.

5. This question concerns the failure data reported in the submittal:

   a) Please explain the rationale for not distinguishing among various types of pumps or valves in the development of individual failure data, while this distinction is made in the common cause failure data.

   b) Please explain how the plant specific data were derived. Provide enough details for understanding the process (e.g., was Bayesian updating used?). The error factors for the plant specific data are very small (on the order of 1-2 for many of the failure rate data, even for failure rates that are on the order of 1E-6/hr). How were such small error factors derived from plant specific data, considering that there could not have been many failures to give a sharp peak?

6. The following question deals with your consideration of internal flooding at ANO-1:

   a) The ANO-1 flood analysis has identified that a flood in the control room (RAB 129-F discussed in p. 3.6-12 of the submittal) has the capability to degrade the control room function. The submittal also identified the Control Room as a shared facility with ANO-2 (See p. 1.2-2).

   Please provide a description of how a flood scenario in the ANO-2 control room would affect the operation of ANO-1's control room (and vice-versa). What is the frequency of such a (dual) initiator and what is the value of the associated (dual) CDF?

   b) Please discuss how you considered failure of flood mitigating equipment (drain plugging, back propagation through drains, intercompartment door failures) in developing your flood scenarios. Provide the basis for data or assumptions used.

   c) Please provide the basis for your assumption that scenarios are limited to locations reached within 20 minutes by the flood waters.

   d) Please clarify whether spray from inadvertent actuation of fire suppression equipment is included in your flood scenarios, and if not, provide the basis for excluding this possibility.

7. The ANO-1 IPE's cut-off value for the cut-sets is 1E-8 per year (1E-9 per year for containment bypass sequences). Often PRAs use 1E-9 per year or smaller values for cut-off. Please provide assurance that the ANO-1 IPE has captured the majority (i.e., 95%) of the total CDF despite the 1E-8/yr cutoff value.

8. In the description of the system, "Reactor Coolant System (RCS) Pressure Control," the modeling conditions are discussed for the "electromatic relief valve," ERV (i.e., PORV) (p. A-70). The submittal states: "Local control of the ERV block valve is provided in the event of an emergency." It is not clear how the pressurizer ERV and the block valve are modeled. Please discuss:

   a) What fraction of time is the block valve closed?

   b) How is this modeled? What is the impact on the results if not modeled?

   c) Is it considered in the separate ATWS analysis and what is the estimated impact if not considered?

   d) Discuss the operator actions required to open the block valve and the ERV when needed.

9. The IPE initiating event analysis dismisses HVAC failures as being too slow to evolve into a plant trip and states that these failures are bounded by other initiators. Your system dependency table does not show any dependency on HVAC.

   a) Does your initiating event analysis consider that HVAC loss can eventually lead to multiple failures and/or irrecoverable failures of equipment, which therefore would not be bounded by other initiators? If not, please provide the basis for discounting this possibility.

   b) Are HVAC failures subsequent to initiators included in your fault trees? If yes, please provide the updated dependency table; if not, please provide the basis for discounting these failures.

   c) Please provide a more detailed discussion of your investigation into the impact of loss of HVAC in rooms containing safety related equipment, including electrical switchgear and relays, pumps and in the control room. Your discussion should include the following: systems in the areas considered, basis for elimination (i.e., review against the rooms containing electrical equipment), description of the method of assessment, including calculation and tests, credit for operator actions, alarms, procedures, and staged equipment.

   Please consider that in some instances equipment may be isolated at a temperatures lower than that necessary to cause damage to that equipment.

10. It is not clear in the submittal if plant changes due to the Station Blackout rule were credited in the analysis. Alternate AC power is considered only as part of the sensitivity analyses (p. 6.18). Please provide the following: (1) identify whether plant changes (e.g., procedures for load shedding, alternate AC power) made in response to the blackout rule were credited in the IPE and what are the specific plant changes that were credited; (2) if available, identify the station blackout CDF and the impact of these plant changes to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF); (3) if available, identify the impact of each individual plant change to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF); (4) identify any other changes to the plant that have been implemented or planned to the implemented that are separate from those in response to the station blackout rule, that reduce the station blackout CDF; (5) identify whether the changes in #4 are implemented or planned; (6) identify whether credit was taken for the changes in #4 in the IPE; and (7) if available, identify the impact of the changes in #4 to the station blackout CDF.

11. The status of some of the potential plant improvements to reduce the likelihood of core damage and/or improve containment performance discussed in the submittal is not clear. Please clarify the submittal information by providing 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 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 was to be credited in the reported CDF (or containment failure probability), or the increase in the CDF or the conditional containment failure probability if the credited improvement was to be removed from the reported CDF (or containment failure prob.)

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.

12. NUREG-1335, Section 2.1.6 part 4 asks for "a thorough discussion of the evaluation of the decay heat removal function." Section 3.7.3 of the IPE, "Decay Heat Removal Evaluation," does not provide specifics or 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 that function.

13. In many PRAs RCP seal LOCA is a significant contributor to the CDF either as an initiating event or as a system failure consequential to another initiator. While the submittal discusses RCP seal LOCA consideration, please provide the additional information requested:

    a) Please provide a discussion of the RCP seal LOCA model used, including the support system needed and their function (e.g. seal and motor cooling, ICW, seal injection, etc.). Include the probability vs. leakage rate vs. time data and any specific test results. Also, please explain how the seal cavity recirculation pump works and how it is modeled.

    b) Provide a discussion of operator actions which are proceduralized and their timing in the event of a loss of one or the other method of seal cooling.

    c) On page 3.1-25, it is stated that tests with loss of CCW showed no seal degradation for a 40 minute run of the tests. The submittal goes on to say that therefore, there will be no loss of seal integrity for loss of seal injection <u>and</u> ICW cooling for up to 40 minutes. Please provide the basis for this extrapolation.

    d) On page 3.1-24, it is stated that there will be no RCP seal LOCA in case of SBO at ANO-1 (while noting NRC's disagreement with using St. Lucie and San Onofre test results for this purpose, and noting Entergy's disagreement with the NRC on this issue). Notwithstanding these disagreements, this position seems to contradict the position explained in part "c" above. Please explain these seemingly contradictory positions.

    e) Please provide an estimate of the impact of your assumptions regarding the RCP seal LOCA model on your results (core damage frequency, significant sequences, system importance measures, important operator actions).

    f) Have you given credit to the "near future installation of an AC source for seal cooling" in your model? If so, provide an estimate of the effect of this credit on the results.

14. On page 1.4-1 of the submittal it is stated that "conservatisms" were relied upon in the study.

    a) Please provide a brief summary of the major conservatisms in your analysis and their estimated impact on your results.

    b) Were the HRA values conservatively derived?

c) Discuss how do the conservatisms impact your results on a relative basis, i.e., with respect to relative importance of sequences, initiators, component and HRA failures.

15. It is not clear from the submittal whether the impact of the human to cause an accident was considered. Identification of the pre-initiator human events that can disable a system, such as failure to properly restore after test or maintenance or miscalibration of instrumentation, are essential to the human reliability analysis. Table 3.4-1, "Human Failure and Recovery Event Data" and Table 3.4-2, "Major HRA Event Input Data" in Section 3.4, "Human Reliability Analysis" of the submittal may contain this information but are difficult to understand. Please provide a list of the types of pre-initiator human events, in order of importance, that were considered in the analysis.

16. If the submittal does include pre-initiator human actions, it is important to describe the process used to identify and select the important pre-initiators involving miscalibration of instrumentation and the failure to properly restore equipment to service after test or maintenance. The process used to identify and select the instrumentation calibration related human action events may include the review of procedures, and discussions with appropriate plant personnel on interpretation and implementation of the plant's calibration procedures. For assessing the failure to restore important equipment to service after test or maintenance, the process may include the review of maintenance and test procedures, and discussions with appropriate plant personnel on the interpretation and implementation of the plant's test and maintenance procedures. Section 3.4, "Human Reliability Analysis" of the submittal and its associated Tables 3.4-1, "Human Failure and Recovery Event Data" and 3.4-2, "Major HRA Event Input Data" do not appear to contain a description of the process. Please provide a description of the process that was used to identify pre-initiator human actions involving miscalibration of instrumentation and failure to restore equipment to service after test or maintenance. In addition, please provide examples illustrating the processes using several relatively important pre-initiator human actions.

17. The submittal does not clearly identify the actual recovery factors applied in quantifying the pre-initiator human events. Factors that are used to modify the generic basic human error probabilities (BHEP) can include, for example, post-maintenance or post-calibration tests, daily written checks, independent written verification checks, administrative controls, etc. In Section 3.3.3, "Human Failure Data," of the submittal, there were no pre-initiator recovery factors mentioned. If they are used, please provide a list of recovery factors considered, their associated values, and provide specific examples illustrating their use. Also, if used, please provide a concise discussion of the justification and process that was used to determine the appropriateness of the recovery factors utilized.

18. It is not clear from the submittal how dependencies associated with pre-initiator human errors were addressed and treated. There are several

ways dependencies can be treated. In the first example, the probability of the subsequent human events is influenced by the probability of the first event. For example, in the restoration of several valves, a bolt is required to be "tightened." It is judged that if the operator fails to "tighten" the bolt on the first valve, he will subsequently fail on the remaining valves. In this example, subsequent HEPs in the model (i.e. representing the second valve) will be adjusted to reflect this dependence. In the second example, poor lighting can result in increasing the likelihood of unrelated human events; that is, the poor lighting condition can affect different operators' abilities to properly calibrate or to properly restore a component to service, although these events are governed by different procedures and performed by different personnel. This type of dependency is typically incorporated in the HRA model by "grouping" the components so they fail simultaneously. In the third example, pressure sensor "x" and "y" may be calibrated using different procedures. However, if the procedures are poorly written such that miscalibration is likely on both sensor "x" and "y", then each individual HEP in the model representing calibration of the pressure sensors can be adjusted individually to reflect the quality of the procedures. Section 3.4, "Human Reliability Analysis," of submittal states that for "... a pre-initiator the interdependency model is used." but does not appear to contain a description of the process. Tables 3.4-1, "Human Failure and Recovery Event Data" and 3.4-2, "Major HRA Event Input Data" seem to indicate possibly dependent human actions. Please provide a concise discussion of how dependencies were addressed by the interdependency model and treated in the pre-initiator HRA such that important accident sequences were not eliminated.

19. The submittal is not clear about the risk significance of human actions to contribute to, and mitigate the consequences of an accident. Section 3.4, "Human Reliability Analysis" of the submittal and its associated Tables 3.4-1, "Human Failure and Recovery Event Data" and 3.4-2, "Major HRA Event Input Data" do not appear to provide this information. Please provide a list of the most important risk significant post-initiator human actions, such as the requirement for the operator to manually open valves CV-1405/06 upon failure to deliver flow from the sump (MANSUMP), and the operator to trip the RCPs following a loss of seal cooling (QHFIRCPTRP). For the important actions, please provide the details of how the associated human error probabilities (HEPs) were quantified in the most important sequences in which they appear.

20. The submittal does not clearly describe the type of human errors considered for each post-initiator human event identified. For example, a human event identified may be the failure to feed and bleed, while the types of human errors considered may involve failure to open the correct valve (error of omission), or opening incorrect valves (error of commission). No mention of types of human errors was found in the submittal's Section 3.4, "Human Reliability Analysis." Please identify what types of human errors were considered for each post-initiator human events identified.

21. The submittal does not clearly describe the method used to identify and select response type actions and recovery type actions for analysis. The method utilized should confirm the plant emergency procedures, design, operations, and maintenance and surveillance procedures were examined and understood to identify potential severe accident sequences. The submittal's Section 3.4, "Human Reliability Analysis" and associated Tables 3.4-1, "Human Failure and Recovery Event Data" and 3.4-2, "Major HRA Event Input Data" are not clear as to whether the submittal's HFEs and recovery events correspond entirely to the response type actions and recovery type actions classifications. Also, the method used was not addressed. Please provide a description of the process that was used for identifying and selecting the response and recovery type actions evaluated.

22. The submittal does not clearly indicate what screening process was utilized to help differentiate the more important post-initiator human events. The submittal's Section 3.4, "Human Reliability Analysis" does not provide the basis for the screening value(s) used. Please provide the screening value(s) basis and a list of all post-initiator human actions initially considered and those screened.

23. In applying performance shaping factors (PSFs), the consideration of time is important. The submittal is not clear on how available time and "required" time were calculated for the various post-initiator human events. "Required" time is the time needed for the operator to detect, diagnose and perform the actions. In Table 3.4-2, "Major HRA Event Input Data" of the submittal, under the "Ex-Control Room Model" are included "Available Time" and "Response Time." Section 3.4, "Human Reliability Analysis" contains no explanation about the "Response Time." Please confirm that "Response Time" is the same as "required" time. If it is not, please define "Response Time" as used in the IPE, and provide all the "required" times for Table 3.4-2 human events. For several of the important post-initiator human events, provide the available and "required" times estimated for the operator action and the bases (e.g., calculated from simulator exercises, estimated from walkdowns) for the time chosen. Also provide illustrations of how different times were calculated for the same task but in different sequences.

24. It is not clear from the submittal what plant-specific performance shaping factors (PSFs) were used to modify the basic human error probability (BHEP) and what the bases were for reducing HEPs through their application. The plant-specific information could include the size of crew, availability of procedures, time available and time required etc. The process could include examination of procedures, training, human engineering, staffing, communication, and administrative controls. The submittal, in Section 3.4, "Human Failure Data," briefly states that "... a complete analysis are handled by PSFs ..." However, neither Table 3.4-2, "Major HRA Event Input Data," nor Table 3.4-1, "Human Failure and Recovery Event Data" contain the PSFs used. Please provide a list of the types of plant-specific PSFs considered and their

values, and discuss by way of example how these PSFs were used to modify the BHEPs of important post-initiator human events.

25. The submittal is not clear whether response type actions and recovery type actions were considered. Response type actions include human actions performed in response to the first level directive of the EOPs. For example, suppose the EOP directive instructs the operator to determine reactor water level status, and another directive instructs the operator to maintain reactor water level with system X. These actions - reading instrumentation to determine level and actuating system X to maintain level - are response type actions. Recovery type actions include those performed to recover a specific failure or fault and may not be "proceduralized." For example, suppose the EOP directive instructs the operator to maintain level using system x, but the system fails to function and the operator then attempts to recover it. This action - diagnosing the failure and then deciding on a course of action to "recover" the failed system - is a recovery type action. Section 3.4, "Human Failure Data," of the submittal states that "Human Failure Events (HFEs) were considered to be events involving failure of human actions that are of a manual actuation nature rather than of a corrective or recovery nature. Recovery events, on the other hand, were considered to be events involving failure of human actions of a corrective or recovery nature. ... Recovery type actions include those performed to recover a specific failure or fault and may not be proceduralized." It is not clear whether the submittal's HFEs and recovery events correspond entirely to the response type actions and recovery type action classifications described above. Please provide separate lists of the response and recovery type post-initiator actions considered in the analysis. If response or recovery type actions were not considered, please justify. If response type actions were credited, were they proceduralized? If not, please justify.

26. It is not clear from the submittal how dependencies were addressed and treated in the post-initiator HRA. The performance of the operator is both dependent on the accident under progression and the past performance of the operator during the accident of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant accident sequences and, therefore, the identification of significant events. The submittal in Section 3.4, "Human Reliability Analysis," states "Interpersonal dependencies were modeled explicitly in the ANO-1 HRA for slips" but does not appear to contain a description of the process. Tables 3.4-1, "Human Failure and Recovery Event Data" and 3.4-2, "Major HRA Event Input Data" identify four dependent human actions. Please provide a concise discussion of how dependencies were treated and examples illustrating how dependencies were addressed in the post-initiator HRA such that important accident sequences were not eliminated. The discussion should address the two points below:

- Human events are modeled in the fault trees as basic events such as failure to manually actuate. The probability of the operator to perform this function is dependent on the accident in progression, what symptoms are occurring, what other activities are being performed (successfully and unsuccessfully), etc. When the sequences are quantified, this basic event can appear, not only in different sequences, but in different combinations with different systems failures. In addition, the basic event can potentially be multiplied by other human events when the sequences are quantified which should be evaluated for dependent effects.

- Human events are modeled in the event trees as top events. The probability of the operator to perform this function is still dependent on the accident progression. The quantification of the human events needs to consider the different sequences and the effects of other human events.

27. Table 3.4-2, "Major HRA Event Input Data" under the heading "In-Control Room Model" has three columns labeled "Burden," "Response Time," and "Available Time." Please explain what is meant by "Burden" with its provided options of "yes" or "no." Also, please explain the "Response Time" options of "1," "2," "3," or "default." In addition, please provide the missing units for "Available Time," which most probably is "minutes."

28. Table 3.4-1, "Human Failure and Recovery Event Data" is the database of 16 detailed analyzed HFEs and 48 recovery events for the ANO-1 PRA. For the events in the table, please provide the event's mean, 95th, and 5th percentile [Error Factor] estimates. For a number of the more important events please provide the "complete, step-by-step hand computation" of these values, referred to in the submittal.

29. Since Table 3.4-1, "Human Failure and Recovery Event Data" of the submittal provides a brief (less than 20 words) "Event Description" of the detailed analyzed HFEs and recovery events, it is sometimes difficult to distinguish between events. The table contains several pairs of events which have the same description but different "event names" and mean HEPs. For example, both DGCRANK and DGCRANK2 have the same description but different HEPs. The same holds for other pairs including DHMAN and DHMANR as well as SWECPREC and SWECPREC2. Please provide enough of an event description so that the different events can be easily distinguished from each other and so that it is possible to understand the differences between similar events.

30. The ANO-1 submittal assumes that a catastrophic reactor building failure is not likely since liner failure and reactor building leakage are expected to occur prior to structural reactor building failure. From the few examples of quantified event trees provided in the submittal, it seems that the highest probability assigned to structural failure is 0.5, and the typical value is 0.05. The basis for this assumption rests on two scoping calculations. The first, a simple hoop stress calculation, concludes that the ultimate failure pressure, i.e., the pressure at which structural failure occurs, is 162 psig. The second

calculates leakage failures due to liner tear at five different locations to be 154.3, 154.6, 154.8, 159.6, and 163.1 psig. The submittal also states that while these liner tear pressures are "conservatively" assumed as failure pressures, they are actually pressures for onset of tearing and additional pressure would be needed to propagate the tear. Given that the calculations are simple scoping calculations, i.e., expected to have relatively high uncertainty, and that the lowest liner tear value is within 5% of the structural failure value please provide additional justification for your assumption that the structural containment failure is unlikely.

31. Please provide the actual ultimate containment pressure used in the ANO-1 analysis. Several values appear in the submittal. For example, page 4.4-7 states 162 psig, page 4.4-13 states 154 psig, while the Figure on page 4.6-32 seems to imply a pressure around 170 psig.

32. Please clarify Figure 4.6-4 on page 4.6-32. The third (or bottom) figure, which shows the adjustment of the ANO-1 failure pressure relative to the reference plant, appears to be drawn incorrectly. As currently shown, the probability of surviving pressure P increases as P increases.

33. For the calculation of the personnel hatch failure modes the submittal states that the glass viewport, which has a 5 inch diameter, is made of Corning 7740 hardened glass and is designed to withstand 315°F and 70 psi pressure. The submittal then goes on to state "The ANO-1 calculation of the stresses modeled a punching type failure mode. Based on these calculations, the viewport is judged capable of withstanding the severe accident conditions and retain its structural integrity before ultimate pressure is reached." Since ultimate pressure is 162 psig and temperatures could be higher than 315°F, please clarify your calculations and your conclusion that the viewport will retain its structural integrity until ultimate pressure is reached.

34. When discussing reactor building loads related to high pressure melt ejection (HPME) for ANO-1 the submittal notes that for Zion and Surry (the reference plants) much of the core debris will remain in the cavity or instrument tunnel after vessel failure, even for high RCS pressure conditions. The submittal then states (page 4.6-11) "While ANO-1's cavity differs significantly, the amount of debris remaining in the cavity is expected to be similar. Therefore debris dispersal and pressure loads associated with the dispersal will likely be similar for ANO-1 and the reference plants." On the next page (4.6-12), when discussing the formation of coolable debris ex-vessel, the submittal states that "For high pressure vessel failure scenarios, much of the core debris may be transported out of the cavity to the reactor building." In addition the drawings of the cavity area provided in the submittal show a relatively open path from the cavity to the containment.

a) Please justify your assumption that much of the core debris will remain in the cavity or instrument tunnel during HPME in ANO-1.

b) Please comment on the effect increased dispersal would have on the early containment failure probability in light of the results of the sensitivity study of Section 4.8, which shows that increased loads at high pressure vessel failure could increase the early failure probability by up to a factcr of seven.

35. The submittal recognizes the due to ANO-1's unique cavity area geometry containment liner impingement is a possible failure mode.

a) The submittal states that liner impingement is more likely when the RCS fails at low pressures than at high pressures. As a result early failures are much more likely in ANO-1 for cases when the RCS has an induced depressurization than if it stays at high pressure. Why is liner impingement unlikely when the RCS fails at high pressure? If much of the debris is retained in the cavity and surrounding area, as is assumed in the ANO-1 IPE, it would seem that a likely place for it to collect would be at the exposed liner at the end of the instrument tunnel. Please justify your assumption that liner impingement is unlikely with RCS failures at high pressures.

b) Please also comment on what effect a higher likelihood of liner impingement would have on early failure frequency in light of the fact that your sensitivity study in Section 4.8 showed that early failures could increase by up to 14 times with higher liner impingement probabilities.

36. The submittal also assumes a very low likelihood of failure by liner impingement if there is water in the cavity area during low pressure RCS failures (5E-05 for wet versus 0.2 for dry conditions). However, no discussion on the amount of water or debris depth are provided. Please summarize your calculations, including an estimate of the range of debris depth expected and the amount and depth of water expected, which justify the low likelihood assigned. Please also comment on what effect a higher likelihood of liner impingement would have on early failure frequency in light of the fact that your sensitivity study in Section 4.8 showed that early failures could increase by up to 14 times with higher liner impingement probabilities.

37. Because of ANO-1's unique geometry, the submittal considered the possibility of containment failure resulting from ex-vessel steam explosions which could cause a large impulse load to be propagated down the instrument tunnel to the reactor building wall. The IPE assigns a probability of 0.2 to this failure if the RCS fails at high pressure. For a low pressure failure the probability is dropped to 5E-5 (although Table 4.8-1 shows a .005 value). Please explain why the likelihood of an ex-vessel steam explosion of sufficient energy to fail the reactor building wall is assigned such a low likelihood with a low pressure RCS failure.

38. The ANO-1 IPE considers induced surge line and hot leg failures but does not consider induced steam generator tube rupture. Other B&W plant IPEs have stated that natural convection paths through the heat exchanger tubes cannot develop in a straight tube heat exchanger, such as the B&W OTSG, as long as loop seals remain in place. 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 steam generator and could induce SGTR. Please discuss whether there is any mechanism, such a- the restart of the RCPs, that may cause clearing of the loop seal and a creep rupture of the steam generator tubes for ANO-1.

39. On page 4.1-4 of the submittal the survivability of reactor building systems under potentially severe accident conditions is briefly discussed. However, the discussion only considers the impact of reactor building failure on the sprays and the fan coolers. While the containment is still intact there can be significant adverse conditions on these systems. Please indicate how the effects of high containment pressures, temperatures, moisture content and aerosols were considered. If they were not considered, please justify this omission.

40. The Generic Letter CPI recommendation for PWR dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements.

    Please discuss whether plant walkdowns have been performed to determine the probable locations of hydrogen releases into the containment. Discuss the process used to assure that: (1) local deflagrations would not translate to detonations given an unfavorable nearby geometry, and (2) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

    Please identity 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 likelihood of local detonation and potentials for missile generation as a result of local detonation.

41. The submittal provides the sensitivity studies which were carried out with respect to event tree quantification. However, no sensitivities of the radionuclide release calculations, i.e., variation in the release and removal terms of the calculations, were provided. Were the input constants of Table 4.7-2 used for release calculations varied to obtain a range of results? If so please provide such results. If not, please justify this omission.

42. Section 4.9.7 of the submittal states that a plant specific vulnerability has been defined for ANO-1 IPE as any condition satisfying the top evaluation category of the tables adopted from NUMARC 91-04 *that*

is not artificially increased by conservative assumptions regarding uncertain plant response or phenomena. How are the terms "artificially" and "conservative" defined when this statement is applied?

43. Section 4.9.7 of the submittal, "Back-end Vulnerability Screening Results", mentions that although no back-end vulnerabilities were identified for ANO-1, several issues relating to ANO-1 reactor building performance during a severe accident were noted as appropriate for additional investigation. These include several of the most critical items, such as liner impingement and ex-vessel steam explosions. What is the status of these additional investigations and what results have they produced?

44. The submittal states in Section 4.2.6 that a MAAP input deck was developed for ANO-1 and a limited number of plant specific MAAP calculations were carried out. Results were not obtained until after most of the scoping work was completed. The MAAP calculations were "used to confirm the scoping calculations and gain additional insights into the ANO-1 severe accident response." No further information is given. Please summarize the results of the MAAP calculations carried out and comment on their agreement or disagreement with the scoping calculations used in the IPE.