January 26, 1996

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - BRAIDWOOD STATION, UNITS 1 AND 2 (TAC NOS. M74382 AND M74383)

Dear Mr. Farrar:

By letter dated June 30, 1994, you submitted the Braidwood Station, Units 1 and 2, Individual Plant Examination (IPE) results for NRC's review. Based on our review of your submittal, we have determined that additional information is required. The enclosed request for additional information (RAI) identifies the information that is needed. The questions are related to the internal event analysis in the IPE, including the human reliability analysis and the containment performance improvement (CPI) program. The RAIs were developed by our contractor, Brookhaven National Laboratory, and reviewed by the IPE "Senior Review Board" (SRB). The SRB is comprised of NRC research and development (RES) staff and RES consultants (Sandia National Laboratories) with probabilistic risk assessment expertise. Please review these questions and provide your written response within 60 days of the date of this letter.

This requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Please contact me should you have any questions regarding this request.

Sincerely,

Original signed by:

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457

Enclosure: Request for Additional Information

cc w/encl: see next page

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REQUEST FOR ADDITIONAL INFORMATION

BRAIDWOOD STATION, UNITS 1 AND 2

1.0 FRONT-END ANALYSIS

We have reviewed the Level 1 (front-end) portion of the Braidwood Station Individual Plant Examination (IPE) and it appears that it contains weaknesses of the same nature as found in IPEs previously submitted by the Commonwealth Edison Company (ComEd) and reviewed by the NRC; specifically the Zion, Dresden and Quad Cities IPEs. These weaknesses were of sufficient magnitude to contribute to the staff's conclusion that the process used in the Zion IPE could mask potential vulnerabilities. The following brief list identifies areas in the front-end for which the staff has expressed concerns. A more detailed discussion of these areas is provided in an NRC letter from Mr. C. Shiraki (NRC) to Mr. D. L. Farrar (ComEd), "Review of Zion Nuclear Power Station, Units 1 and 2, Individual Plant Examination Submittal - Internal Events (TAC Nos. M74492 and M74493)," dated November 22, 1994.

* Use and application of an optimistic quantification process for the reliability of systems and components, including the associated treatment of common cause failure (CCF), without consideration of the inherent uncertainty that is intrinsic to the analysis.

* The use of the MAAP code to determine core cooling success criteria under conditions where it has not been benchmarked, or exercised with due consideration of the associated uncertainties.

* Credit for more combinations of equipment and components of lesser capacity to achieve success.

* Use of success with accident management (SAM) end-states without significant time margin beyond the cutoff criteria (24 hours) for these sequences.

* Use of optimistic criteria for the elimination of the effects of water from spray or impingement.

In responding to the specific following requests for additional information, please address the concerns that the staff has expressed regarding the front-end approach as identified in the above noted concerns and letter to Commonwealth Edison. If in responding to these questions, the approach used for the front-end in the Braidwood IPE is modified, please incorporate the information requested for the modified approach also.

ENCLOSURE

1.1 The frequency of the success with accident management (SAM) sequences are not provided in the Braidwood IPE. In the Zion IPE the SAM end-states occur with a frequency of 1.8E-4. This is substantial given the magnitude of the frequency of such sequences which might otherwise be assigned as core damage. In addition many of the top SAM sequences in the Braidwood IPE are steam generator tube rupture (SGTR) sequences which are usually significant contributors to early releases.

a) Please provide the frequencies of those sequences designated as ending in "Success with Accident Management" (SAM).

b) If accident management actions for any SAM sequences need to be invoked before or shortly after the 24 hour mission cutoff, please provide an explanation for including such sequences in the SAM end-state rather than the core damage end-state.

1.2 The submittal indicates that HVAC systems are independent, each with its own power supply and cooling water connections, and that loss of HVAC was not included as an initiating event.

a) Please describe how the impact of loss of HVAC to the rooms containing safety related equipment was analyzed and provide the rationale for eliminating loss of HVAC as an initiating event. In your description include a discussion of the areas considered in the analysis (including rooms containing AC and DC power equipment and the Control Room), methods of assessment, credit given for operator actions, timing, and temporary equipment.

b) Identify those systems for which the HVAC was included as part of the system model.

1.3 The following question concerns success criteria used in the submittal.

a) Several success criteria seem to deviate from those typically assumed for this type of plant. These success criteria are significantly more optimistic than the FSAR criteria.

i) For large LOCA success, the IPE states that no accumulators are needed if one LPI or two HPI pumps are successful. Also, credit is given for operation of one HPI pump in conjunction with discharging of one accumulator. Most other studies specify that three out of three accumulators must discharge, in addition to LPI injection, while no credit is given to HPI operation. The staff is concerned about the use of the MAAP code to determine core cooling success criteria under conditions where it has not been adequately benchmarked or used without consideration of associated uncertainties. Please address these concerns regarding the use of these optimistic criteria in your search for vulnerabilities.

ii) For small LOCAs, the submittal has a success path with one LPI pump and reactor coolant system (RCS) cooldown accomplished by one AFW pump and one steam generator atmospheric relief valve. The shutoff head for the LPI pump is 200 psi and the cooldown rate specified in the post LOCA cooldown emergency procedure should not exceed 100°F/hr. This means that significant time may pass before the LPI pumps can begin to make up lost inventory. Since there is no success criteria for equipment to maintain RCS inventory, how will the operator maintain and control the pressurizer level during this period and how is it addressed in the analysis?

iii) For small LOCAs, one success path and one SAM path utilize one charging pump without any PORV opening, with no heat removal through the steam generators. It is also stated that if one "SI" pump is operating instead of charging pumps, then two PORVs have to be opened. Typically, small LOCA success criteria require a feed and bleed operation (at least for certain break sizes). Please discuss the basis for this success criteria for small LOCAs without RCS heat removal, including break size and the conditions for use of the SI pump and 2 PORVs or only charging pumps.

iv) Are the above success criteria included in operator guidance (e.g. EOPs) and are the operators trained in them?

b) Since the use of "realistic" success criteria may have had a significant impact on the results of the IPE, sensitivity assessments varying the success criteria would provide valuable information. Have such sensitivity studies been performed? If so, please discuss the sensitivity of the results reported in the submittal to the success criteria used in the analysis for the initiating events. Include a discussion of the impact of the success criteria upon those sequences which are designated as "Success with Accident Management" and which, if designated as "failure," would have resulted in an increase in core damage frequency (CDF).

c) The success criteria for the "Dual Unit Loss of Essential Service Water" initiator are not provided in table 4.1.4-1 of the submittal. Please discuss the changes to the success criteria required by this event if different than for single unit loss of essential service water, and if different provide these criteria.

1.4 In the discussion of results in the submittal a comparison is made with the Zion IPE results, where SGTR sequences are much more dominant than at Braidwood. The explanation is that this is due to a different design of the steam generator tubes (they are narrower at Braidwood than at Zion). How is this difference accounted for in the IPE (e.g. different success criteria, failure rates, initiating event frequencies, etc.)?

1.5 The status 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 probability).

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

1.6 The value of 0.032/yr for the loss of offsite power (LOSP) initiating event frequency (for a single unit) is at the low range of typical LOSP frequency values as found in, for example, industry data from NSAC-147. It is also 1/3 the LOSP frequency of some other CECo plants, namely Dresden and LaSalle. Furthermore, LOSP sequences dominate the risk at Braidwood, contributing 88% to the total core damage frequency from internal events. Therefore the LOSP initiating event frequency will have a major impact on the results. Please explain how the LOSP frequency was estimated (both single and dual unit). Include in your discussion how plant-specific information and data were accounted for, including weather related events.

1.7 It is not clear from the submittal how the cross-tied and shared systems are treated for the unit at power if the other unit is in cold shutdown and some of the shared (or potentially cross-tied) systems are experiencing extended downtime. How does the analysis account for the unavailability of the systems that are capable of being cross-tied during the time the opposite unit is in shutdown? Discuss how the following systems were considered in regard to this question and what was the impact: service water, component cooling water, condensate storage tanks, emergency ac power, dc power, instrument air and nonessential service water?

1.8 There are discrepancies in the definitions of the nodes (top events) shown in the submittal. These discrepancies introduce significant ambiguity when interpreting the sequences. For example, node AFT is defined in Table 4.1.3-4 as Diesel Driven Auxiliary Feedwater Pump (DDAFP) (failure). Its guaranteed failure, AFT=1.0, is defined as failure of the DDAFP due to the failure of a DC bus (Table 4.5.2-1) or due to the failure of the service water system (text, pages 1-31 and 1-33). On the other hand, several indicators point to the fact that service water failure does not necessarily lead to DDAFP failure (see sequences 11, 15, 27 etc. or Note 8 of the success criteria in Table 4.1.4-1, page 4-36).

Further confusion is caused by the AFW system description, which does not show direct DDAFP service water or DC power dependence, and by the dependency matrix in Figure 4.2.2.7 (Volume 2) which does not show this relationship either.

In view of the above, please provide:

a) A precise table of system dependencies (cross-tie dependencies included) utilized in the plant model, and,

b) a list of accurate and detailed node definitions for the dominant sequences.

1.9 This question concerns the treatment of special initiators in the IPE analysis:

a) Please explain why the IPE did not consider the plant response to the loss of a single AC bus (an initiator traditionally considered in PRA analyses) and what would be the impact on the CDF and dominant sequences if this initiator were considered.

b) The "Loss of DC Bus 111" initiator contributed 3% to the total internal CDF. The initiating event frequency is rather small 5E-4/yr. This is at the low end of the range of values reported in NUREG/CR-4550, Vol. 1 (the range is from 5.E-4/yr to 6.E-2/yr. The ASEP mean is 5.E-3/yr, i.e. 10 times higher than used here). (It should be also noted that Zion IPE had the much higher frequency of 1E-2/yr.)

i) How does the Braidwood value compare to plant operating experience?

ii) It appears that a significant change in the initiating event frequency could have a concomitant impact on CDF. Has a sensitivity analysis been performed for the impact of loss of this bus or are there plans to do so? If so please provide the resulting impact on the CDF, the sequences and release frequency. Similarly for DC bus 112, which was not included in the analysis.

c) The fault tree analysis guidelines excluded certain types of low probability events, including passive component failures such as pipe ruptures. This guidance, however, may lead to an underestimation of system unavailabilities. It may also result in initiating event frequencies for loss of certain fluid systems, such as SW or CCW, that are low. For example, in the Zion IPE, the SW and CCW loss initiating event were on the order of 1E-7/yr and 1E-10/yr respectively, without consideration of pipe breaks. When CECo did the reanalysis for Zion and requantified these two initiators including the pipe breaks, the initiating event frequencies went up by 3 orders of magnitude, to 1.E-4/yr and 2.3E-4/yr, respectively. The Braidwood initiating event frequencies are 5.65E-4/yr and 4.36E-5/yr, respectively.

i) Were pipe breaks considered in the Braidwood analysis? If not, please provide the initiating event frequencies for Single and Dual Loss of Essential Service Water and Single and Dual Loss of Component Cooling Water, when passive component failures are also taken into account. Also, provide the impact of these new frequencies on the results (CDF and dominant sequences).

ii) If the pipe break events in the CCW or the SW system have a potential for affecting the results of your flooding analysis, please provide the impact on the flooding analysis and its contribution to CDF (see also, question 13. below).

1.10 The small LOCA frequency (6.3E-3/yr), which also includes spurious RCP seal LOCAs, is several times smaller than NUREG/CR-4550 values. Part of this might be due to the new high temperature seals, but only an oblique reference is made to U.S. Westinghouse experience in the submittal. Please provide sufficient details regarding your consideration of all components of small LOCA frequency (including pipe breaks, reseat failures of PORVs/safety valves and RCP seal failures) for an understanding of the basis for your small LOCA frequency value.

1.11 While the submittal states the LOCA definitions in terms of mitigating equipment, it would help our understanding of the model if these definitions were put in terms of the more traditional LOCA break size ranges. Please provide this information if available.

1.12 The submittal does not consider the initiator: "Excessive LOCA" i.e., Reactor Vessel rupture. Please provide a discussion of your consideration of this initiator and if screened out the basis used.

1.13 Interfacing systems LOCA (ISL) as a type is considered in the analysis, and it is indicated that all likely paths were considered.

a) Identify the ISL paths considered, provide a description of the arrangement and the assessment (calculation or otherwise) of the frequency of each system considered for this type LOCA. Include in the description your consideration of relief valves.

b) Identify the frequency and the source of the value used, and any assumptions used with regard to failure of low pressure pipe subjected to RCS pressure.

c) Discuss any consideration of isolation of the LOCA by operator action, and the consideration of maintenance, test and human error in the interface between systems.

1.14 The submittal indicates that all but two flood zones were eliminated from further consideration through a qualitative analysis but provides no information on the zones, equipment affected, frequency of occurrence nor criteria used for their elimination.

a) Please describe your process addressing the zones with safety related equipment considered, the types of flood initiators in these zones and frequency of the initiators, the source of the data and the specific criteria used to eliminate each zone.

b) Discuss your consideration of drains (including back flooding to other areas and probability of failure, i.e. due to blockage), separation, doors allowing flood propagation to other areas, credit given for actions by operators to stop the flood or to mitigate the consequences.

c) Describe your treatment of water intrusion from spray or impingement from leaks and ruptures, including the criteria used for elimination of the consideration of spray and impingement.

d) Please discuss how maintenance errors were treated in the flooding analysis. Include errors committed while in cold shutdown, which were left undiagnosed until the postulated flood event occurred while the unit was at power.

e) How does your consideration of pipe breaks in the CCW and SW system (see question 10. above) impact the results of the flooding analysis?

1.15 These questions concern the use of failure data in the IPE.

a) Discuss how your use of the "key" component strategy did not miss identifying <u>other</u> components in your plant whose specific failure rate may be higher than the generic rate used and could contribute to potential vulnerabilities.

b) In the submittal, plant specific data were used for most "key component" failures. However, for turbine and diesel driven auxiliary feedwater pumps (failure to run), generic failure data were used. Considering the fact that the AFW system is important (contributing to 75% of the core damage frequency), please justify the use of generic failure rates for these components and discuss what the impact of this assumption is on the CDF and the dominant accident sequences.

c) Braidwood's containment fan cooler failure to run failure rate (which was also used for Byron) does not agree with the number cf failures and number of hours (provided in Table 4.4.1-3 in the Byron IPE report). Please explain.

d) "Check valve fails to close" (failure rate of 2.E-6/hr) seems to really mean "check valve fails to stay closed" in the submittal. Have you modeled failure of check valves to close on demand, and if so what is the failure rate used? If not, then justify omitting this failure mode especially for headered systems where check valve failure to close may compromise more than one train or one system.

1.16 The system description for the Diesel Driven Auxiliary Feedwater Pump (p. 4-44) does not describe the fuel oil supply subsystem. What is the boundary of the DDAFP for its failure data and does it include failures in the fuel supply system and if not why were these failures not included?

1.17 The following question concerns treatment of common cause in the IPE:

a) Please discuss the practice of using the average of the common cause failure factors in the data base for other components (such as batteries) versus specific values from the generic data base, characterize the magnitude of the values used and discuss your assessment of the impact of this approach on the results of the analysis.

b) It was noted that the beta factor values used for some components (diesel generators, MOVs, pumps) in the IPE were significantly lower (1-2 orders of magnitude) than the values estimated in NUREG/CR-4550 (e.g. for diesel generators it is 1.5E-3 in the IPE vs. 3.8E-2 in the NUREG/CR). A similar difference exists when comparing the Braidwood MGL parameter values to those from the ALWR utility requirements document, an EPRI report. The EPRI data are substantiated by tables showing specific instances of common cause occurrences at specific plants. For example, for decay heat removal pumps, the beta factor in the ALWR document is 2.6E-2, vs. 3.2E-3 for Braidwood. For MOVs, EPRI has a beta of 5E-2 vs. CECo's 1E-2 (for a 2 component system). There are also differences in other MGL parameters (gamma, delta).

Please provide a discussion of the process used to arrive at the MGL values for the CCF factors used in the analysis for these components. Provide sufficient supporting documentation for nomenclature and data (or references) to allow for an understanding of the process. Also, provide an estimate of the impact on your results (CDF and dominant accident sequences) if 4550 or EPRI values were used for the CCF factors.

c) The MGL parameters do not distinguish between various failure modes (e.g. diesel generator failure to start and failure to run failure modes). Please discuss the basis for this approach and why separate parameters should not be used.

d) Section 1.4.4 of the submittal states that the generic common cause failure (CCF) had been screened by an "expert panel" to retain only those events that were applicable for the Byron/Braidwood plants. Furthermore, the panel assigned a lesser probability of occurrence to those types of events which were considered to be addressed by Byron/Braidwood maintenance or operational practices. The approach seems to be rather subjective and uncertain. It may neglect certain CCFs that have not yet occurred at the plant or were not identified, resulting in MGL factors whose magnitude tends to be lower than those determined in other analyses (e.g. NUREG/CR-4550, Vol. 1).

Please discuss the screening process used by the expert panel and how the reduction factors described above were calculated. Discuss how, in your search for vulnerabilities, this approach is sufficient to address the uncertainty in your CCF analysis and to understand the sensitivity of the CCF analysis to the possibility of missing a potential vulnerability using this approach.

1.18 It is not clear in the submittal if plant changes due to the Station Blackout rule were credited in the analysis. Please provide the following: a) 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 the specific plant changes are that were credited; b) if available, identify the total 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); c) if available, identify the impact of each individual plant change on the total plant core damage frequency and on the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF); d) identify any other changes to the plant that have been implemented or are planned to be implemented, that are separate from those in response to the station blackout rule, that reduce the station blackout CDF; e) identify whether the changes in d) are implemented or planned; f) identify whether credit was taken for the changes in d) in the IPE; g) if available, identify the impact of the changes in d) to the station blackout CDF; and h) the contribution to CDF from station blackout.

1.19 There is no discussion in the submittal about the PORV block valve position and how it affects various scenarios (feed and bleed, ATWS).

a) What is the fraction of time that either or both block valves are closed during operation and how is this condition accounted for in the model?

b) If the block valves are or will be closed for some time during operation and are not modeled in the IPE, please assess the impact of this alignment on your results, especially early in core life.

BRAIDWOOD STATION, UNITS 1 AND 2

2.0 HUMAN RELIABILITY ANALYSIS

We have reviewed the Human Reliability Analysis (HRA) portion of the Braidwood Station Individual Plant Examination (IPE) and it appears that this HRA contains weaknesses of the same nature as found in IPEs previously submitted by the Commonwealth Edison Company plants and reviewed by the NRC; specifically the Zion, Dresden and Quad Cities IPEs. These weaknesses were of sufficient magnitude for the staff to conclude that human performance was not treated appropriately in the IPEs. The following brief list identifies underlying assumptions of the HRAs for which the staff has expressed concerns. In addition to these the staff has concerns about human performance modeling and the treatment of accident progression on operator performance. A more detailed discussion of these areas is provided in an NRC letter from Mr. C. Shiraki (NRC) to Mr. D. L. Farrar (ComEd), "Review of Zion Nuclear Power Station Units 1 & 2 Individual Plant Examination Submittal - Internal Events (TAC NOS M74492 and M74493)," dated November 22, 1994.

* Symptom based procedures and improved training have eliminated the need for cognitive action (diagnosis and decision making) in response to an accident event. No operator interpretation or diagnosis is required because the operator would initiate an action only through the use of a procedure after an initial response to alarms.

* In addition to the assumed elimination of the need to diagnose, the probability of failure of the crew to respond to the appropriate alarm is reduced due to assumed operating crew experience.

* All post-initiator tasks can be classified as "step-by-step" rather than "dynamic," even for highly complex accident sequences, because the EOPs are symptom-oriented and the operators are highly skilled and extensively trained, thus resulting in the application of lower values of factors applied for the increase of the human error probability (HEP) due to stress.

* Errors of commission are reduced, across the board, because the equipment and controls are properly labeled, symptom-based EOPs are used, and there is an operating philosophy of reading out equipment numbers back and forth between operators.

* Use of a recovery factor for unproceduralized checking (8.1E-2), i.e., for the operators to recover their own error during an <u>accident response</u>, based on a value from the THERP Handbook, that directs the value to be used only for <u>normal operating conditions</u>.

* Blanket application of the "slack time" model. Errors for tasks performed in the control room with a time window more than 5 minutes, have a high probability of recovery due to an "awareness checking" carried out by an operator, a shift foreman, or a shift engineer and are reduced by a factor of 10. For conditions where the time window is between 60 minutes and three hours an additional factor of about 0.21 is applied.

In responding to the specific following requests for additional information, please address the concerns that the staff has expressed regarding the HRA approach as identified in the above noted concerns and letter to Commonwealth Edison. If in responding to these questions, the approach used for the HRA in the Braidwood IPE is modified, please incorporate the information requested for modified approach also.

2.1 The IPE submittal for Braidwood indicates on page 4-111 that the applicable results of the HRA for Byron were used for Braidwood. It goes on to say that, "The HRA for the Byron IPE is described in this section," not the Braidwood HRA. Even though it is stated that both plants use the same procedures, have similar training programs and that detailed procedure comparisons were made, it is not evident from the submittal that in the search for plant specific vulnerabilities it was verified that the plant specific performance shaping factors (PSF) applicable to Byron HEPs, were indeed applicable to Braidwood. It appears that all the operator actions and the associated human error probabilities are exactly the same except for those associated with the cooling tower. While they may be same for Byron units 1 and 2, it is expected that there may be differences between Byron and Braidwood. Please describe the process that was used to verify that all conditions related to the PSFs applicable to the HEPs for Byron, including physical conditions, location of equipment, and man-machine interface (controls, instrumentation and alarms) are applicable to Braidwood for in-control room and ex-control room actions and thus have no impact on the HEPs used for Braidwood.

2.2 Please discuss the process used to assure that key HRA assumptions about operator actions, information available to operators, plant environment, etc., represent the conditions in the as-built, as-operated plant. In particular, please discuss information related to interviews with operators and plant walkdowns performed during the IPE.

2.3 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 4.4.2-1, "HRA Results Summary," lists approximately 6 dozen post-initiator human events without any pre-initiator human events. Please provide the information requested for pre-initiators in the following questions.

a) If pre-initiator human events were not considered please justify this omission.

b) If pre-initiator human actions were considered, but eliminated from further analysis it is important to describe the process used to identify and select the 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 important 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. Please provide a description of the process 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.

c) Factors that are used to modify the generic basic human error probabilities (BHEP) associated with pre-initiator human events can include, post-maintenance or post-calibration tests, daily written checks, independent written verification checks, administrative controls, etc. If they are used, please provide a list of pre-initiator recovery factors considered, their associated values, and provide specific examples illustrating their use. Also, if pre-initiator recovery factors are used, please provide a concise discussion of the justification and process that is used to determine the appropriateness of the recovery factors utilized.

d) Dependencies associated with pre-initiator human errors should be addressed. 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 require 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. Therefore, please provide a concise discussion and examples demonstrating how dependencies are addressed and treated in the pre-initiator HRA such that important accident sequences are not eliminated. If pre-initiator dependencies are not addressed, please justify.

2.4 In looking for vulnerabilities the risk significance of human actions to contribute to, and mitigate the consequences of an accident is an important parameter that provides insight. This information is not available in the submittal.

a) Please provide a list of the most important risk significant post-initiator human actions (including those in the fault trees), and their contribution to the core damage frequency (CDF), include contributions for actions for RWST refill (ORT) and recovery of AC power (OFW).

b) For the top ten important actions, please provide the details of how the associated human error probabilities (HEPs) were quantified in the various sequences in which they appear.

2.5 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 valve (error of commission). No mention of types of human errors was found in the submittal. Please identify what types of human errors were considered for each post-initiator human event identified and provide some examples for illustration.

2.6 The submittal in Section 4.4.2 states "The operator talk-throughs were ... developed to review the selected critical subtasks and to gather plant- specific factors for consideration in the HRA quantification." 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. Please provide a list of the types of plant-specific PSFs considered and their values, provide a discussion and examples of how these PSFs were used to modify the BHEPs of important post-initiator human events. If the PSFs were addressed in previous examples refer to these examples in your discussion.

2.7 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 diagnose and perform the actions. For several of the important post-initiator human events examined, 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.

2.8 The submittal is not clear on the process used to identify and select the response type actions and recovery type actions used. 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.

a) Please identify if the HEPs addressed in the analysis are response or recovery (as discussed above) type and discuss the process used to select them.

b) If recovery actions are used, are they proceduralized and if not, please justify any credit taken for such actions.

2.9 The submittal is not clear on how the method to evaluate the <u>level of dependence</u> between successive operator actions, outlined in Section 4.4.2.1 in the paragraph entitled "Dependency between Operator Actions," is applied. It is indicated that each of the PRTs were reviewed to determine possible dependencies between operator actions. Please elaborate by providing several examples illustrating how important accident sequences were not eliminated. The examples should cover all the dependency categories, including those whose dependency level may have changed for various sequences.

a) What was the criteria used to determine <u>if dependency exists</u> and was this applied prior to entrance into the decision tree?

b) Identify those HEPs that were eliminated prior to entry into the tree and those whose dependencies were addressed using the tree?

c) The last question on the tree involves workload, which the submittal indicates includes stress, complexity, time available and expected to complete the action. How were these factors combined to give the heavy, moderate and light designations?

d) None of the branches in the tree end in complete dependence. Was complete dependence not addressed.

2.10 The submittal indicates that "Dependency between subtasks of an operator action was determined and the HEPs modified accordingly."

a) Discuss how the dependency between subtasks was determined?

b) Identify those that were determined to be dependent?

c) How were they modified? Please provide examples.

2.11 The submittal indicates that the operator actions in the PRTs were assessed for dependency. Discuss how it was ensured that dependencies for those human actions appearing in the fault trees were accounted for, since it is not evident if the actions within one or among various fault trees are independent of those appearing in another for various sequences.

2.12 The submittal (pg 4-117) identifies three classes of operator actions based on their mean human error probability. Explain the purpose of this classification and discuss and provide examples of when and how the three classes were used in the IPE.

2.13 As requested in NUREG-1335, please provide a <u>listing and a discussion</u> of any sequences that drop below the applicable core damage screening criteria because the sequence frequency has been reduced by more than an order of magnitude by credit taken for human recovery actions (not to exceed 50 of the most significant sequences).

2.14 Please provide the basis (or references as appropriate) for the "error recovery" model described in the submittal. Please include consideration of: the dynamic nature of the accident response; the likelihood for detection of errors by the operator committing the error or by other crew members; the specificity of the error indications; the impact on performance of having identified an error in an important task; and any additional factors that are likely to influence recovery from error under a severe accident and the subsequent sequence of events given that an error is made.

2.15 The submittal is not clear if the need to diagnose an event (i.e., to figure out what is to be done in any given situation) was considered in the HRA analysis. The diagnosis "in NUREG/CR-1278 includes the actions to "perceive, discriminate, interpret, diagnose" an event and the operators "first-level of decision making." While using symptom-based emergency operating procedures (EOPs) removes the need to identify the type of accident, such as a LOCA, their use does not remove the need for other aspects of diagnosis. Please discuss how diagnosis was considered in your analysis. If it was not considered please justify this omission.

2.16 Table 4.4.2-1, "HRA Results Summary" of the submittal provides only a brief description of the human events. The table contains several pairs of events which have the same description and initiators but different "case names" and HEPs. For example, both OBLa and OBLb have the same description but different HEPs. The same holds for other pairs including OAI1 and OAI2, ODPa1 and OPDa2, OIR1 and OIR2, and ONCa1 and ONCa2. Provide an event description that would allow an understanding of the differences in the HEPs. Please explain, by way of example, how particular combinations of values for various PSFs were combined to obtain the HEPs for these operator actions.

BRAIDWOOD STATION UNITS 1 AND 2

3.0 LEVEL 2 ANALYSIS

3.1 Temperature-induced steam generator creep rupture, which is considered in other IPEs, is not addressed in the Braidwood IPE. In some IPEs, the probability of induced SGTR increases as the RCP is restarted following the direction given in the procedures. Please discuss the probability of induced SGTR. Please include in the discussion the probability of RCP operation and the effect of RCP operation on the probability of induced SGTR.

3.2 Containment isolation status is one of the PRT top events. It is also indicated by the fourth character in the 4-character PDS designator. The identification of the important containment penetrations are discussed in some detail in Section 4.2.1.10 of the IPE submittal. With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, p2-11) 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 penetrations, (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 all of the above five areas.

3.3 According to the success criteria (Section 4.1.4, Table 4.1.4-1) successful long-term inventory control without long-term heat removal results in SAM end states. According to the IPE, the accident sequences designated as SAM are not core damage sequences for the purposes of the IPE analyses and submittal report, and accident management activities are required to ensure that the plant attains a long-term safe, stable state. Please discuss the containment conditions (e.g., containment pressure and containment integrity) for sequences with successful long-term inventory control but without long-term heat removal, and the effect of these conditions on accident progression and recovery (e.g., whether there would be containment failure and the effect of containment failure on equipment availability).

3.4 It is stated in the IPE submittal in the discussion of Sequences BL4A, SL6A, and ME6K (p4-185) that "The release of core debris, steam and water to the reactor cavity causes a rapid pressure increase there and blows open the sealed cavity exit plate about ten seconds after vessel failure." It is also stated in the discussion that, "The core debris expelled from the vessel is predominantly spread out over the basement floor in a roughly two-inch thick layer, with about 15% retained in the cavity." However, according to Table 4.5.5-1 these sequences have a low

RPV pressure at vessel failure. Please discuss the model used in the IPE for predicting ex-vessel core debris distribution and the effect of RPV pressure on the distribution. For high pressure sequences please also discuss the paths of the missiles and flying debris generated from the blowing-open of the cavity exit plate and the dispersal of the core debris, and discuss whether there is any potential adverse effects on containment integrity and the equipment in the containment. Keeping in mind the assumed maximum coolable debris depth of 25 cm mentioned in Generic Letter 88-20, please discuss the depth of core debris in the cavity (including the cavity sump) and the effect of non-uniform spread of debris on debris coolability if all debris is retained in the reactor cavity.

3.5 In the Braidwood IPE, a constant value for containment pressure of 98 psig is used in the analyses of containment performance (p4-32, containment integrity success criteria). If the analysis indicates that the containment pressure exceeds 98 psig, it is assumed that the containment fails and release of fission products from the containment, beyond that associated with normal leakage, is initiated. Below 98 psig, containment integrity is assumed to be maintained. Based on this approach, many containment failure modes, such as those associated with HPME, are considered as "unlikely failure modes" and do not contribute to containment failure. Since the containment failure pressure used in the analysis (98 psig) is the median failure pressure, the containment failure probability can be almost 50% for a containment pressure load close to, but less than, 98 psig. To address the uncertainties associated with both containment pressure capability and the loads due to DCH and hydrogen burns, please discuss how the "unlikely failure modes" would contribute to containment failure if the containment fragility curve, instead of the median value, is used to assess the failure probability. Please use both best estimate as well as more bounding pressure loads in the discussion.

It is important that the licensee be aware and recognize the uncertainties associated with containment capability and the various containment phenomena and their impact on containment performance and source terms, and consequently, the identification of vulnerabilities and the development of an accident management program. Please include in the above discussion the impact of uncertainties on containment performance and source terms.

3.6 The CDF contribution from bypass sequences is very small for the Braidwood IPE. The CDF contribution is 2.8E-8 (or 0.10% of total CDF) for SGTR sequences and 1.5E-9 (or 0.01% of total CDF) for interfacing system LOCA sequences. Because of their small frequencies, bypass sequences do not appear in the top 100 sequences, and consequently, bypass sequences are not selected for source term calculations. It is noted that the small SGTR CDF is due to the small steam generator tube diameter and the large RWST used in the Braidwood plant, which, according to the submittal, result in core damage at greater than 24 hours. This plant state is assigned a SAM state, instead of a CD state, in the submittal, and accident management actions are required after 24 hours to provide long-term core cooling. Because of the uncertainties associated with accident progression and accident management activities and the potential of significant release that may result from a containment bypass event, please estimate, or discuss, the source terms associated with SGTR for Braidwood.

3.7 It is stated on page 4-201 of the Braidwood IPE submittal that, "A review of plant differences between Braidwood and Byron identified the leak-tight cavity hatch as the major plant difference that would cause variations in the results between Level 2 analyses performed for Braidwood and Byron.", and that "the Braidwood cases generally had nearly identical in-vessel progressions, slightly later containment failure times, and slightly larger volatile fission product releases." It is explained in the IPE submittal that "The delay in containment failure for Braidwood sequences tended to reduce volatile fission product releases in two ways compared to Byron: it increased the time available for naturally-occurring fission product removal mechanisms to reduce the airborne concentrations of volatile fission products in containment prior to failure; and, it reduced the time interval for fission product release between containment failure and the end of the calculations."

It is noted that the above explanation for the difference between Byron and Braidwood in volatile fission product releases (i.e., the delayed containment failure for Braidwood results in a reduced volatile fission product release) is not consistent with the impression obtained from comparing calculational results, as stated in the first quote presented above (i.e., the Braidwood cases have slightly larger volatile fission product releases). Comparison of the calculational results presented in the two IPE submittals shows that the release fractions of the volatile fission products for similar sequences are greater for Braidwood than for Byron. Please clarify the above inconsistency, identify the mechanisms that causes differences in volatile fission product releases, and discuss the relative contributions of these mechanisms to volatile fission product releases.

3.8 The CECo version of the MAAP code includes a model for external vessel cooling, which may delay, or prevent, vessel failure if the vessel lower head is submerged. However, according to the Braidwood IPE, "External vessel cooling was not shown to be possible with the current Braidwood cavity configuration. In all of the Braidwood Level 2 cases analyzed using CECo-MAAP, the cavity exit hatch prevented sufficient water accumulation in the cavity to submerge the lower head prior to vessel failure." On the other hand, external cooling is very likely for Byron. The availability of water to the reactor cavity prior to vessel failure is identified in the IPE as one important difference between Byron and Braidwood. However, comparison of the calculational results presented in the IPE submittals shows that the vessel failure times for similar sequences are practically identical between Braidwood and Byron. This seems to indicate either that the external cooling model was used (or not used) in both Byron and Braidwood calculations or that the effect of external cooling on vessel failure time is minimal (which does not seem likely). Please clarify, and discuss the effect of external cooling on vessel failure time.

3.9 In the base case MAAP analysis, the high hot leg temperature predicted did not last long enough to conclusively determine that the hot leg would experience creep rupture. A sensitivity case, B930311L, was performed to characterize the effects on sequence progression resulting from an induced hot leg rupture before vessel breach. The primary effect of the induced hot leg rupture, according to the IPE submittal, is to depressurize the reactor vessel, which allows the accumulators to inject before vessel failure. The addition of the accumulator inventory delays the time of vessel failure by 2.3 hours relative to the base case. Another effect, as shown in Table 4.5.6-4, is to increase the fraction of volatile fission product release from 0.85% to 6.76%. Please discuss whether this result is representative of other RPV depressurization cases and, if it is, then discuss the benefit and adverse effect of RPV depressurization on containment performance and source terms.

3.10 The plant data that are of interest to the Level 2 analysis is provided in Section 4.3.1 of the IPE submittal. Although this Section provides the essential data for accident progression discussion, it does not contain the detail mentioned in NUREG-1335. Please provide the data in tabular form as described and requested in Table A.1 of NUREG-1335.

3.11 Equipment important for prevention of core damage and/or containment failure was evaluated for survivability during the range of accident conditions postulated in the IPE. To accomplish this task, the Braidwood equipment survivability study was divided into three phases. It is stated in the IPE submittal that "Phase II of the study involved a review of all plant response trees to determine which equipment, including instrumentation, is important in achieving successful end states. The limiting conditions, with respect to the PRTs, were then identified for each piece of equipment and a survivability evaluation was completed. Please provide an example by discussing how this process is applied to the containment heat removal system. Please include in the discussion the environmental conditions, including the effects of aerosol plugging, derived and used in the evaluation.

3.12 According to the sequence designators defined in Table 4.1.3-2, a "K" is used for the fourth designator if the release fraction of volatile fission products is less than 0.1%, and a "L" is used if the release fraction is between 0.1% and 1%. However, from the results presented in Table 4.5.5-3, many sequences with a "K" designator have volatile fission product release fractions greater than 0.1%, which, according to Table 4.1.3-2, should use an "L" designator. Please discuss this apparent inconsistency and its significance for the results reported.

3.13 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 (including accident management procedures).

a) 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:

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) Identity potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided.

c) 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.