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Docket Nos. 50-352/353



Mr. Edward G. Bauer, Jr.  
Vice President & General Counsel  
Philadelphia Electric Company  
2301 Market Street  
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Dear Mr. Bauer:

Subject: Request for Additional Information - Limerick

The Reliability and Risk Assessment Branch has reviewed the Probabilistic Risk Assessment you prepared for the Limerick Generating Station. This review has indicated a need for the additional information delineated in Enclosure 1.

We look forward to receiving your response no later than June 11, 1982.

Sincerely,

Original signed by:

A. Schwencer, Chief  
Licensing Branch No. 2  
Division of Licensing

Enclosure:  
As stated

cc: See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION AND CLARIFICATION ON  
LIMERICK PROBABILISTIC RISK ASSESSMENT (PRA)

LIMERICK GENERATING STATION

Note: all items contained in this enclosure are grouped according to the Chapters and Appendices of the PRA.

CHAPTER 1

- PRA 1.01 The text conveys the notion that no cross-ties between Unit 1 and Unit 2 were taken into account (p. 1-18). Cross-ties could be sources of redundancy as well as additional failure causes. Are there cross-ties between units (e.g., RHRSW, RHRHX)? If yes, provide rationale for not considering them in the analysis.
- PRA 1.02 The ultimate containment capability is calculated to be in excess of 140 psig which is approximately 2.5 times the design pressure (p. 1-19). Given the fact that the containment exhibits leakage under design conditions, what is the increased leakage rate of the containment prior to reaching 140 psig? Provide a description of how the leakage between the primary and secondary containment was modeled.
- PRA 1.03 Based on the design leakage of the containment (p. 1-20), it is expected that some amount of containment environment constituents will escape into the reactor building; this may become more pronounced when the containment is at an elevated pressure. Given the long-time nature of some transients, what is the probability of hydrogen combustion inside the reactor building? In the event that there is containment failure prior to core melt, what is the likelihood of hydrogen combustion in the reactor building? Does hydrogen combustion inside the reactor building further aggravate radioactive releases, and if so, in what way?
- PRA 1.04 The sources of data used for the Limerick study were summarized into four categories. Please provide criteria for the selection of one data source over the other? What was the rationale for combining several data sources in some instances and not others? What are the guidelines used to determine whether or not the data base should be integrated (p. 1-23)?
- PRA 1.05 It is stated that for the purposes of the analysis, the Technical Specifications of the Peach Bottom Station and the test frequencies from the Susquehanna Station were used (p. 1-24, 1-25). Provide rationale for this combination as representative of LGS.

PRA 1.06 What is the rationale for Guideline No. 11 (p. 1-32)?  
Provide the reference for the "improved chronic-health-effects model"  
referred to (p. 1-10).

PRA 1.07 "Table 1.2 Summary Of Success Criteria For The Mitigating Systems  
Tabulated As A Function Of Accident Initiators (p. 1-26)." For each  
initiator, including all 5 transients, reference the section of the  
FSAR that describes the adequacy of the selected success criteria.  
For those initiators, including all 5 transients with success  
criteria not described in the FSAR, provide the reference that  
justifies the adequacy of the selected success criteria.

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### CHAPTER 3

- PRA 3.01 For the event trees shown in Chapter III, provide the reference for the probabilities assigned, to each system success or failure and/or frequency of initiators. Provide rationale and method used whenever different probability values are used for the same event.
- Identify values obtained from fault trees and provide cross reference to corresponding fault tree figure.
- PRA 3.02 The Limerick FSAR reported that the vapor suppression system reliability and effectiveness varies as a function of the LOCA size. However, in the Limerick PRA study, it does not appear that this particular aspect of the system has been incorporated into the containment event trees. If it was neglected, what is the justification? If it was included, provide additional details on how the system was modeled.
- PRA 3.03 In addressing manual shutdown as an initiating event, there are situations in which the reactor operator is required to shutdown the reactor in order to be in compliance with technical specifications due to the unavailability of certain safety systems. Provide a summary of how these types of manual shutdowns were included in the event tree depicted in Figure 3.4.2?
- PRA 3.04 Plateout and settling is assumed to "remove" radioactivity.  
Can the radioactivity be released back to the environment by some physical means, for instance water flash (p. 3-125)?
- PRA 3.05 Isn't the  $\delta'$  sequence a drywell overpressure and not a wetwell overpressure as labeled in the far right column of the containment event trees (see for example p. 3-82)? Explain the difference between  $\delta$ ,  $\delta'$  and  $\delta''$ . The definition of  $\delta$  is confusing. In the containment event trees it means containment overpressure either drywell or wetwell. The definition on the top of page 3-133 indicates it is a drywell failure.
- PRA 3.06 In Section 3.2, the text states that one of the most important aspects of the event tree technique is that it ensures that all of the key accident initiators are identified. How does the event tree technique identify key initiators, and how does it identify all key initiators?
- PRA 3.07 Further justify the statement in Section 3.4.3.2 that potential failures of the reactor pressure vessel as an initiating event have a very low probability of expected occurrence. What is the effect on the overall consequences by omission of this event?
- PRA 3.08 According to p. 1-14, Section 3.2 will discuss the subject of completeness. Further details as to why those events noted in the section satisfy the completeness requirements are needed. Have events like, RCP seal failure, loss of instrument and control air, loss of DC power, etc. been examined in the Limerick study?

PRA 3.09 In order to successfully operate the ADS, it must be manually initiated in a timely fashion (p. 3-17).

Provide the basis for the time limit on how soon the depressurization should begin.

Is there a time limit beyond which depressurization is not possible?

Is there any requirement on the rate of depressurization?

It is stated (p. 3-18) that the alternate methods of depressurization are given low probability for success, since they involve "creative operator actions under potential stressful conditions". Will there be approved procedures delineating steps required to implement these alternate methods? Are these methods included in the quantification of the sequence?

PRA 3.10 Why does a "controlled" manual shutdown require SRV actuation? Isn't the plant scrammed from a power level below the bypass valve capacity? Explain the major sequence of events which are expected for a normal reactor shutdown.

PRA 3.11 Explain why a value of  $1.1 \times 10^{-4}$  was used for the unavailability of RHR/RHRSW or PCS, given a failure of the SRV's to reclose, in Figure 3.4.3. This is the same as for turbine trip or manual scram event trees. The additional problem of recovering feedwater (the initiating event) should increase the unavailability as stated on p. 3-27 under event W description.

PRA 3.12 Provide supporting documentation and/or calculations showing that a feedwater pump can add water to the reactor vessel following a scram and a subsequent stuck open SRV. The event trees for turbine trip and MSIV closure show the feedwater availability to be the same, independent of the condition of the SRV. It is realized that, should the feedwater pump not be able to continue running, due to low steam pressure, the condensate pump would take over at approximately 600-700 psig pressure. However; operator actions and additional valve operation would seem to reduce the probability of successful operation. Have these items been considered?

PRA 3.13 The event tree for manual shutdown has different feedwater system unavailabilities depending upon the condition of the SRV's. Why does the difference exist in this case?

The statement at the top of p. 3-20 discusses overriding of the low vacuum interlocks for the turbine bypass valves. Have the operator actions required to bypass the MSIV low vacuum interlocks been considered in calculating the unavailability of the power conversion system?



- PRA 3.14 For the MSIV transient, the report indicated that the RCIC steam condensing mode was not evaluated (p. 3-28). On p. 3-20 it is stated that the RCIC steaming mode was included in the turbine trip event tree. Why is this decay heat removal method not consistently included in the analysis?
- PRA 3.15 In order to establish natural ventilation in the HPCI and RCIC rooms, operator action is required. Is this going to be part of the emergency procedures?
- PRA 3.16 Further elaboration on the removal of the emergency core cooling functionality from the event tree is required (p. 3-40).
- PRA 3.17 Are there any erroneous actions expected upon a plant scram condition, i.e., containment isolation due to level shrink or turbine trips (main and RFPT) due to actual or sensed level swell? If so, how has this been taken into account in the accident sequences? Is MSI closure trip point at Level 2 or Level 1?
- PRA 3.18 Please explain the basis for assigning a reactor scram failure of  $1 \times 10^{-5}$  for a large LOCA (p. 3-42) and  $3 \times 10^{-5}$  for medium and small LOCAs (p. 3-45 & 3-47).
- PRA 3.19 The text indicates (p. 3-43) that the success criteria and calculated probability of long term coolant recirculation and short term coolant injection are similar. Why are the success criteria for long term and short term demands the same? What is the difference in system configuration between coolant injection and coolant recirculation?
- Given the long time nature of some of the accident scenarios - in the order of twenty to thirty hours, was failure subsequent to successful system actuation addressed in the Limerick study (failure to run)? If yes, were the degraded environmental conditions under which the systems must operate taken into consideration?
- PRA 3.20 What are the set points for the high radiation interlock for the COR? Page 3-46 states that the COR is assumed available for a medium LOCA. What would the containment radiation level be from this event and would the COR actually be available?
- PRA 3.21 Some sequences on the Turbine Trips ATWS event tree (p. 3-53) are designated as negligible. The  $T_T C_M C_{12} U$  sequence is  $8.3 \times 10^{-8}$  which is not negligible when compared to other sequences on the same event tree which are assigned probability values such as  $6.4 \times 10^{-9}$  for  $T_T C_{MD}$ . This discrepancy is present in other sequences and for other events. What is your criterion for assigning sequence path probability as negligible?

PRA 3.22 What is the probability of failure for the secondary containment (p. 3-50)?

Given the unity probability for a number of the branches with MSIV not open, what do TW, TWE, TA, TAE, TQ, and TQE signify?

PRA 3.23 The report states (p. 3-56) that with multiple relief valves failed open, the RHR is required to operate successfully.

Is there a time limit on how long multiple relief valves could stay open before exceeding the capability of the RHR system? Has this been accounted for in the PRA?

PRA 3.24 The  $T_{TCM2}$  sequence (p. 3-57) does not use COR due to "high radiation associated with incipient fuel failure". Why is there no incipient fuel failure with the  $T_{TCMR}$  sequence on that same page? A related question is to give the basis of the 90% MSIV isolation assumption for the  $T_{TCM2}$  sequence.

PRA 3.25 The  $T_{TCMR}$  sequence (p. 3-57) states that it is "assumed" that RPT and FW runback are tripped from the same set of logic and sensors. Are they in fact tripped from the same logic and sensors? What flow rate does the FW run back to? Has the case been investigated in which the FW runback does occur, but the RPT does not? This would seem to be a more limiting case, since vessel inventory would be rapidly decreasing.

PRA 3.26 Page 3-69 states that ARI is successful if, and only if RPT is successful. Provide detailed information on ARI.

PRA 3.27 Page 3-65 shows ARI either working or failing independent of a success or failure of RPT. Is this consistent with the requirement on page 3-69 that ARI is successful if, and only if RPT is successful?

PRA 3.28 (Top Paragraph, p. 3-86) The statement is made that the diaphragm floor is drained into a sump and the downcomer pipes. This drainage capability eliminates the possibility of a molten core dropping in one large mass from the vessel directly into a pool of water. How does this statement apply if containment spray is used? The downcomers are approximately one foot above the floor level so a large amount of water can accumulate on the floor prior to the molten core dropping. It is realized that no credit for containment spray has been assumed, but have negative effects, such as the above or excessive steam production, been accounted for?

- PRA 3.29 Limerick takes a  $10^{-2}$  failure rate per demand for COR (see bottom of p. 3-102). A Sandia Report on "Risk Assessment of Filtered-Vented Containment Options for a Mark-I Containment" shows a maximum reduction in core melt probability of a factor of 10. Limerick's value looks to be optimistic. (See also the bottom of p. 3-21).  
Explain why Limerick COR has been given an unavailability of  $10^{-2}$  per demand.
- PRA 3.30 What are the bases for the selection of the probabilities on the containment event tree? Address each containment failure mode in detail.
- PRA 3.31 Why was an average value of  $10^{-3}$  per event used for a coherent in-vessel steam explosion when more detailed values of  $10^{-2}$  for a steam explosion during a LOCA event and  $10^{-4}$  for a steam explosion during non-LOCA events were stated on p. 3-114?
- PRA 3.32 Provide supporting analysis and/or calculations to show that RCIC (as stated on the bottom of p. 3-104) alone or HPCI alone is adequate for coolant inventory makeup during an ATWS condition and does not result in core meltdown.
- PRA 3.33 Why are transients in which the SRV's fail to open transferred to only the large LOCA event tree?
- PRA 3.34 The Emergency Operator Guidelines (Rev. 0) state in Steps LC-2.5 and SD-2.2 that if the SRV's are cycling, the operator is supposed to manually open one SRV to reduce pressure to 150 psi below the SRV set points. Page C-29 shows the reactor pressure to be cycling about the SRV set points indicating that this action has not been included in the analysis. Has this operator action been included in the event tree and fault tree quantification and if not, why hasn't it?
- PRA 3.35 The containment event tree assumes an equal probability of containment failure occurring in the wetwell or drywell. Interpretation of the information in Appendix J, and the information presented at the February meeting results in our assumption that the containment failure always starts at the midwall of the wetwell and rapidly progresses upward to the drywell. Thus, both the wetwell and drywell will be failed simultaneously. This would remove the distinction of whether or not the failure was in the drywell or wetwell and whether or not the suppression pool has been lost, i.e., all failures rapidly lead to drywell failure and loss of the suppression pool scrubbing is irrelevant. What is the rationale for using the containment event tree sequences as presented in Figure 3.4.14?

PRA 3.36 Where in the report is the propagation of uncertainties for the dominant sequences T<sub>F</sub>QUX, ATWS and LOCA documented?

PRA 3.37 In the table in page 3-122, Iodine released to the environment is typed as vapor. Clarify whether its deposition to the ground was considered in the subsequent CRAC analysis or not.

PRA 3.38 Provide discussions as to how the following parameters in Table 3.6.5, which were part of the inputs to CRAC, were determined.

- Time of release
- Duration of release
- Warning time
- Elevation of release, and
- Energy release

What were the values of the above mentioned parameters for the sequences C<sub>2</sub>Y' and C<sub>3</sub>Y' in the same table?

PRA 3.39 Provide the description of those failures that were identified by the study that could disable more than one ADS valve.

CHAPTER 4

PRA 4.01. Explain in detail how Figure 4.3 was generated. What exactly was used to compute the risk of Limerick at WASH-1400 composite site with WASH-1400 data and methods?

How was Figure 4.2 "WASH-1400 BWR with updated methods and data" obtained?

APPENDIX A

- PRA A.01 Tables A.1.2 and A.1.3 list the anticipated transients considered in the EPRI-SAI study and GE assessment. Provide clarification as to why transients #14-19 and #22 of Table A.1.2 do not appear in Table A.1.3.
- In Table A.1.3, under Turbine trip with bypass, transients #36 and #37 are indicated; there are no corresponding numbers 36 and 37 in Table A.1.2.
- In view of the EPRI survey and the GE assessment, discuss the major differences noted in Table A.1.3, e.g., loss of condenser, inadvertent opening of bypass, turbine trip with bypass...etc.
- PRA A.02 The electric load rejection with bypass valve failure in Table A.1.3 is listed under turbine trip. Shouldn't this be listed under MSIV closure so as to be consistent with the statement on p. 3-15? This would change the transient initiator frequencies used in the PRA for MSIV closure and turbine trip.
- PRA A.03 In the footnote on page A.12, a statement is made to the effect that due to the controlled nature of manual shutdown, there is an increased reliability of feedwater to maintain reactor inventory. What is the qualitative and quantitative basis for such a statement?
- Provide clarification for the statement: "However, coolant injection functions, ATWS, and LOCA sequences are not affected by these initiators when they are quantified (p. A-12)."
- PRA A.04 On the discussion of reported failures for all sizes of piping, the sum total of all the failure percentages comes to 64.4% (p. A-13); furnish information on the remaining 35.6%. Given the magnitude of the balance in failures (35.6%) versus the largest failure category (25.1%), how does one justify the accuracy of the data if the 35.6% is not included in the data base?
- PRA A.05 In addition to pipe rupture, there are other causes which could lead to LOCA, for instance, valves failed open, failure of recirculation pump seals. Were they addressed and properly included in the analysis as LOCA initiators?
- PRA A.06 The use of a 10% reduction of the probability of pipe rupture for the probability of LOCA seems to be a rough estimate (p. A-16). Are pipe failure probability data given per unit length? Are there any data on primary piping ruptures? If yes, have they been compared to the 10% estimate?
- PRA A.07 Table A.1.6 gives the probabilities of a LOCA for various cases. Discuss the method, analysis and criteria used in the selection of the Limerick values.

APPENDIX A (Cont.)

- PRA A.08 Table A.2.1 compares median and mean values. It is further stated on p. A-22 that "mean values of failure rates used in WASH-1400 appear lower than mean values reported in other sources". Where is this shown?
- PRA A.09 Provide more detail on the modeling of how a component could fail to run for the duration of the accident. Also, explain the basis or justification on how 20 hours was selected (p. A-29).
- PRA A.10 The fact that values in Table A.2.4 agree does not necessarily mean that the 4 cases can always be indiscriminately used. What criteria was used in the selection of cases in the LGS/PRA?
- PRA A.11 (p. A-62, top paragraph, last sentence). This sentence states that "components involved in the room cooling and ventilation are not included in the estimate of maintenance unavailability". Page B-5 (bottom) states that "room cooling must be available to maintain acceptable temperatures in the HPCI compartment" for long term operation. Is there an inconsistency in ignoring the cooling system? The same comment applies to RCIC (see p. B-8).
- PRA A.12 In the fault tree model of the diesels (p. A-72), not all the dependencies are shown, for instance, based on Table A.4.1 if one diesel is out of service, the LPCI, both core sprays, remaining diesel generators and the containment cooling systems have to be operational. How are these dependencies accounted for in the fault tree model?
- PRA A.13 The average demand of 65.4/diesel-year seems to be low, compared to the data given for Zion and Cook. We could not verify this number because of the lack of necessary data regarding Plant X. Provide additional information (p. A-91).

## APPENDIX B

- PRA B.01 A statement is made (top paragraph, p. P-21) about the large uncertainty of bringing the reactor from full power to cold shutdown. Is this consistent with the statement at the bottom of page 1-17 which states that the operation is of a routine nature?
- PRA B.02 Explain why ADS pressure sensor is not included in Table B.5.5 (p. B-53).
- PRA B.03 Figure B.9.2 depicts a generic fault tree of a MOV. How are redundant demands on a MOV modeled? If one assumes a situation in which the first demand is to close the valve and the second demand is to open the valve, how is this modeled in the study?
- PRA B.04 Have DC failures been considered and if so, did they include operator and maintenance contributions?
- PRA B.05 There are more than 600 penetrations in the containment. Has their ability to withstand pressure up to 145 psi been evaluated? Would some of these penetrations, for instance the electrical penetrations, yield to excessive leakage under elevated temperature and pressure environments?



APPENDIX C

- PRA C.01 How was the metal water reaction of the fuel bundle zirconium channels considered?  
Which core melt model and metal-water reaction model is assumed?
- PRA C.02 The statement is made (top paragraph of p. C-16) that HPCI is allowed to stay on even after high exhaust pressure trip point is reached (i.e., the operator overrides the interlock). Does the operator have enough time to do this since the containment fails in less than 50 minutes? What were your assumptions?
- PRA C.03 What is the basis for assuming that the diaphragm floor fails at 2/3 of the floor penetration (70 cm) (p. C-19)? What happens to the core after floor failure?
- PRA C.04 Please provide the modification that was done to INCOR which tracks the water level in the vessel and assigns 30% power to covered nodes and decay heat power to uncovered nodes (p. C-15).
- PRA C.05 Provide all data for input decks of RACAP including documentation of calculations performed in order to obtain the required inputs.

## APPENDIX D

- PRA D.01 Provide the basis for the assumption that 98% of the secondary containment building air flow is filtered and 2% is not (p. D-13)?
- PRA D.02 Justify the use of the CORRAL value instead of the REACT value for the Tellurium release fraction. How sensitive are the consequences to this assumption (p. D-28)?
- PRA D.03
- (a) In the tabulation of nuclide species, iodine is listed as elemental and/or organic. In the fission product transport calculations, however, iodine is assumed to be CsI (Appendix D). In the estimation of SGTS effectiveness, different DF values for elemental and organic forms are quoted (Appendix D). Please identify what forms of iodine were assumed in what proportions, and why; then determine decontamination factors consistently for this form(s).
- (b) Indicate the applicability of the decontamination factors of Table 3.6.4 with respect to fission product element and physical/chemical form.
- (c) Section D.2.3.1 states that the SGTS was assumed to achieve certain decontamination factors independent of the accident sequence. The evaluation of filtration systems as ESFs in NUREG-0772, in contrast, indicates susceptibility of these systems to plugging as a result of high aerosol loading for some sequences. Discuss the particulate loading capability of the SGTS, and compare with the expected aerosol loading (including non-radioactive materials) for the various accident sequences.
- (d) What is the basis of the statement that the three conditions listed on page D-8 "dictate" the degree of suppression pool decontamination? Indicate the relative importance of such variables as degree of subcooling, gas composition (non-condensable gas fraction), gas flow rate, and iodine concentration.
- (e) On page D-9 it is stated that the reason for increasing the saturated pool DF for CsI is the greater solubility of CsI. Since saturated pool DFs are limited by reduced surface interaction, as stated on the previous page, explain how a difference in solubility of highly soluble compounds can produce an order of magnitude change in DF.
- (f) Since any cesium iodide reaching the suppression pool is in particulate form, explain why CsI is treated differently than other particulates.

(g) Quantify the "additional credit" in decontamination factors discussed on p. D-9 and explain how this additional credit is achieved by pH, particularly in view of the discussion of CsI in the previous paragraph.

PRA D.04 The natural deposition analysis of WASH-1400 assumed iodine predominantly in the elemental form. In view of the assumption of CsI discussed in the previous section, explain how the WASH-1400 model is applicable.

PRA D.05 Provide the "data from the TMI accident" which indicate that CsI is a "much larger constituent than previously believed," and discuss this data with respect to the expected partitioning of elemental iodine.

PRA D.06 The first paragraph of p. D-12 states that RB overpressurization results in ground level releases, while the last paragraph states that pressurization of the RB would result in release via the SGTS exhaust stack. Please clarify.

PRA D.07 The discussion of radioactive material inventory and risk associated with the spent fuel pool is inaccurate in several respects:

- a) The spent fuel pool inventories quoted from WASH-1400 are not necessarily applicable to LGS. Past experience indicates that inventories of many discharged cores must be expected to be stored in the pool. As a result, the inventory of several radiologically significant long-lived isotopes (e.g. Sr-90) may be substantially larger than the core inventory. The text should be revised accordingly.
- b) NUREG-CR/0603 discusses risks from Classes 3 - 8 only, and therefore, provides no basis for the claim that risks (including Class 9 events) from spent fuel pool events are negligible. This section should be revised to provide a basis for the claims made concerning accidents involving the spent fuel pool or, if no such basis is provided, the conclusions should be revised accordingly.

## APPENDIX E

PRA E.01 What effect does the formation of CsI have on the postulated accident sequences? How much Tellurium is oxidized during the various sequences?

Why is there no Co-58 or Co-60 at the Limerick plant? Why are Cs-134, Cs-136 and Cs-137 inventories so much smaller than WASH-1400 (p. E-30)?

PRA E.02 In accident sequences in which the containment building fails due to steam overpressurization, there is the chance of a large amount of "fog" or vapor formation around the building due to the expanding steam. What effect would this have on the release fractions as calculated by CORRAL?

PRA E.03 It is not made clear in Section 3.7 or in the Appendix E as to what type of meteorological sampling scheme was actually used in the Limerick site-specific consequences analysis. Clarify whether it was the stratified meteorological sampling of WASH-1400 using 91 start times, which is the normal sampling method for the CRAC analysis, for which 8760 consecutive hourly met-data must be input; or was it a non-standard sampling scheme of invariant meteorology (for the Start Code 9 of the CRAC Manual) generated by joint frequency distribution of meteorological data over any non-specific period (continuous, or with gaps) of time? Use of the latter sampling scheme of CRAC is known to result in acute fatality CCDF about an order of magnitude lower compared to the acute fatality CCDF generated by the former (the standard) sampling. Mention of features of both sampling schemes in Item II in Table E.1 has led to this lack of clarity.

PRA E.04 Provide justification for use of the numerical values of 25 miles, 1.2 mph, and 0.0 days respectively (See Table E.6) for the evacuation distance, effective speed of evacuation and time lag before evacuation in Limerick site-specific consequence analysis. Values of these parameters should have been obtained from evaluation of Limerick site's plans for emergency response within the plume exposure pathway Emergency Planning Zone which is approximately a circular region, centered at the reactor, of about 10-mi radius.

What is the basis for assuming that 95% of the people participate in the emergency evacuation? How sensitive are the consequences to this assumption (p. E-17, 18, 19)?

PRA E.05 Provide bases for (a) the value of 0.29 used for the ground shielding factor without sheltering in Table E.6 in contrast to 0.33 assumed in WASH-1400 for situations of normal activities of people, and (b) the breathing rate of  $1.1 \times 10^{-4} \text{ m}^3/\text{sec}$  rather than the standard value of  $2.3 \times 10^{-4} \text{ m}^3/\text{sec}$ .

PRA E.06 Clarify as to which is the spatial interval over which the shielding factors for sheltering in Tables E.2a, E.2b and E.3 were used in CRAC analyses and as to the impact on the Limerick site-specific consequences.

PRA E.07 In situations where the emergency response would be the sheltering mode rather than the evacuation or the no-response modes, there would still be a time lag before people would actually be in the sheltering mode (due to delay in notification advising people to shelter). During this time-lag the shielding factors only for the situation of normal activities of people as assumed in WASH-1400 would be appropriate. Further, for deriving any benefit from the improved shielding factors for inhalation (given the sheltering mode) it is also necessary to advise the people to open the windows and enhance ventilation to expel the contaminated air trapped inside the buildings for exchange with the outside fresh-air after the radioactive plume has left the area. Unless this latter action were taken, the dose from prolonged inhalation of the contaminated air trapped in the buildings would result in higher doses from plume inhalation exposure pathway (see WASH-1400, Appendix VI page 11-8 and Figure VI II-5). Therefore, provide a discussion of the emergency response scenario used and matching analysis of how the shielding protection factors in Tables E.2a, E.2b and E.3 have been factored-in in the Limerick site-specific consequence analysis.

PRA E.08 Provide the following additional information for use in staff's confirmatory Limerick site-specific consequence calculations:

- a. Population input for CRAC standard spatial grid, i.e. for each area element generated by the 16 direction sectors of the compass and the 34 rings of outer radii as specified in the description of the Subgroup SPATIAL in page 49 of the CRAC Manual, for the year 1970 and the year 2000.
- b. State code and habitable land fraction for each area element of the CRAC spatial grid, and

- c. Estimates of evacuation times including the notification times, and travel (response) times for clearing a 10-mile plume exposure pathway Emergency Planning Zone under normal and adverse conditions, consistent with the expected traffic loading on the existing road networks and for various segments of the population (in schools, factories, hospitals, etc.).

PRA E.09 Provide a basis for the "conservative estimates of saturated pool DF", as well as a reference for the "other data evaluations." Provide the data for these evaluations and discuss their applicability to the accident conditions at LGS.

APPENDIX G

PRA G.01 It is stated that the use of mean values in point estimates of a fault tree will result in the mean value of the top event, provided that the basic events are independent. This is not the case, however, if the basic events of the fault tree (or any other technique) include identical components that fail independently, but are characterized by the same failure rates. Was this effect taken into consideration in estimating the mean values of the accident sequences?

PRA G.02 In order to compare results of the Limerick PRA to those of WASH-1400, median values were estimated for the Limerick results based on mean values. What distributions were assumed in this process? Are the Limerick median results shown in the report?

## APPENDIX H

- PRA H.01 The Core Dispersal Model described in Appendix H involves a molten jet exiting the vessel and attacking the concrete. How does the erosion of concrete influence the strength of the diaphragm floor and potentially the pedestal wall?
- PRA H.02 The Core Dispersal Model, as described in Appendix H, involves the rapid cooling of 50% of the core materials. This appears to be inconsistent with Appendix C, which considers the steam spike associated with vessel failure to be uncertain. Hence, containment failure is based on a gradual pressure rise and is predicted to occur several hours after vessel failure. Explain this apparent inconsistency.
- PRA H.03 There appears to be drains directly below the vessel through the diaphragm floor covered only by thin steel plates. These steel plates would offer little resistance to the attack of a molten jet of core materials at vessel failure. Failure of these plates would open up a direct path between the wetwell and the drywell. The core materials could then mix with water increasing the potential for steam explosions and/or rapid steam generation. Would this increase the potential for containment failure at vessel failure?
- PRA H.04 The core dispersal model involves large dispersal forces. What is the effect on the integrity of the containment of such large dispersal forces?
- PRA H.05 The above questions imply (for those accident sequences with the containment intact at vessel failure) that there would be the potential for containment failure at vessel failure rather than due to gradual over-pressurization failure. What is the impact of this on the appropriateness of the release categories and its influence on risk?



## APPENDIX I

PRA I.01

Provide a description of the process used to discover Limerick plant-specific intersystems dependencies and common cause failures. For example, those compromises in redundancy due to maintenance and testing procedures, HVAC dependencies, AC power dependence upon DC control (DC control of EDG start), EDG support systems dependencies, the assumption that equipment can perform in the hostile environment resulting from the accident initiator, and location dependent failures. Include a summary of the contrasts between the process used to discover intersystems dependencies at Limerick versus the process used in WASH-1400. Provide a description of the mathematical and graphical (ET/FT) methods used to accommodate the increased probabilities of failure due to the discovered dependencies.