July 15, 1992

Docket No 50-412
Serial No. BV-92-029

Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: GENERIC LETTER 88-20 INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES - REQUEST FOR ADDITIONAL INFORMATION (TAC NO. M74379)

On March 17, 1992, Duquesne Light Company submitted a report titled "Beaver Valley Power Station Unit 2 Probabilistic Risk Assessment, Individual Plant Examination, Summary Report." This submittal is in response to Generic Letter 88-20. The staff review of this report is underway, but based on the review to date, certain additional information identified in the Enclosure to this letter is required. The requested information relates to the internal event analysis in the individual plant examination and to the containment performance improvement program. DLC is requested to provide the requested information no later than August 31, 1992.

The requirements of this letter affect fewer than 10 respondents, and therefore, are not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Ronald W. Hernan for

Albert W. De Agazio, Sr. Project Manager
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Enclosure:
As stated

cc w/enclosure:
See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

BEAVER VALLEY UNIT 2 INDIVIDUAL PLANT EXAMINATION

1. a) Describe briefly the peer review performed on the Individual Plant Examination (IPE) to help assure the analytic techniques used in the back-end analysis were correctly applied. Identify specific areas reviewed, expertise of the reviewers, and characterize the peer review findings and any significant comments.

   b) As an example of the internal review performed, provide a copy or summary of peer review comments and resolutions (as appropriate) for aspects of the probablistic risk assessment (PRA) involving the "Emergency Switch Gear Ventilation" from system analysis through event tree quantification, plant improvements and conclusions.

2. Describe how containment loading was assessed for each of the containment event tree (CET) end-states. Discuss the development of plant-specific probability distribution functions of failure likelihood for the range of failure pressures.

3. Describe how phenomenological uncertainties were accounted for during the quantification of containment event trees.

4. Section 4.1.4 "Equipment Survivability" (page 4.1-6) of the IPE states that "survivability of equipment for BV-2 is such that equipment failures under severe accident conditions would not create instances of unusually poor containment performance (UPCP) given a severe accident."

   a) State the definition of UPCP and discuss the basis for this definition.

   b) Was the conditional and absolute probability of UPCP for internal events only estimated? If so, please provide the estimates.

5. a) Provide a concise discussion of how the IPE process treated equipment survivability during a severe accident scenario.

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

   c) Section 4.1.4.1 of the BV-2 IPE (Page 4.1-6) states that the containment response reported in Reference 4-7 for the Zion plant can be taken as representative of that for BV-2. Discuss the applicability of the Zion analysis to BV-2.

   d) Explain how the information in Table 4.1-3 was used in the BV-2 IPE process.

6. Describe briefly the plant-specific insights obtained from the BV-2 back-end analysis, and discuss how the BV-2 back-end insights were or will be used to enhance plant safety.

7. Discuss the considerations given to in-vessel steam explosion as a contributor to early containment failure probability.

8. a) Provide a discussion of the ignition sources and limits used in the hydrogen combustion analyses. Were sensitivity studies performed to evaluate the impact on the IPE results due to the uncertainties of the ignition limits used?

   b) Provide the information requested in NUREG-1335 (Section 2.2.2.1), i.e., accurate but simple representations of the containment showing the instrument tunnel, reactor cavity compartment, loop compartment(s), annular compartment(s) and upper compartment with specific identification of potential reactor release points and vent paths indicated. Estimates of compartment free volumes and vent path flow areas should also be provided. Please address specifically how this information is used in the assessment of hydrogen pocketing and detonation.

   c) Discuss the plant-specific effects on containment integrity and equipment survivability due to local detonations. The discussion should cover likelihoods of local detonation and potentials for missile generation as a result of local detonations.

   d) In page 4.6-19 on Top Event 20 - Late Burn of Combustible Gases, the IPE states that "If the containment is not inerted ..., hydrogen burns are assumed to be assured in this time period; however, these burns are not expected to challenge the containment." Please discuss briefly the reasons for not expecting the hydrogen burns to challenge the containment.

9. NUREG-1335 recognizes the importance of considering uncertainties in the accident progression and CET quantification. EPRI recommends that sensitivity studies be performed by MAAP users, which could provide qualitative insight into understanding uncertainties. Please specify what specific revision(s) of the MAAP-3.0B code were used for the BV-2 PRA. Address the Gabor Kenton & Associates report prepared for EPRI ("Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP-3.0B). In particular with respect to Appendix A of the report, indicate for each of the 78 indicated parameters:

   a) If the recommended value(s) were used,

   b) If value(s) other than the recommended value(s) were used and the basis for the choice; or

   c) If the sensitivity study indicated was not performed, provide the reasons for omitting the recommended analyses.

10. Discuss briefly the quantification results for each containment isolation failure mode (including common-mode failure).

11. The table on pages 2.4-1 and 2 identifying walk-throughs does not explicitly identify any specific system walkdowns by analysts to account for the impact of plant modifications prior to walk-throughs or modifications conducted during the time frame of the IPE. In addition in the list of information sources (Table 2.4-1), there is no mention of engineering documents used to control plant modifications.

    What is the "FREEZE" date used for the plant configuration analyzed in the IPE?

    Since there is usually a lag time between documents that request plant modifications and revision to documents that were used to base the models on, were any modifications incorporated in the plant that were being done just before the freeze date that were not incorporated in the model?

12. Duquesne Light Company (DLC) has stated that the PRA for BV-2 was originally performed by Pickard, Lowe and Garrick, Inc. (PLG) and Stone & Webster Engineering Corporation (S&W), and that DLC personnel incorporated plant-specific data and requantified the model. However, Table 5.3-1 shows minimal involvement of the DLC organization in reviewing the quantification.

    Since expertise in the methods is important to ensure that the techniques are correctly applied, please discuss DLC personnel participation in the update of the BV-2 Model and the completion of the Beaver Valley Unit 1 (BV-1) PRA.

13. Section 5.4 resolution of comments indicates that the review comment/resolutions were documented in accordance with the PLG-0223, "Quality Assurance Program." Does conformance with this program comply with the DLC in-house requirements for documentation?

    Will comment/resolution for BV-1 use PLG's program or DLC's?

14. Table 3.1.1-2 identifies Instrument Air as being captured under Initiating Event "TLMFW." However, there is no discussion in Section 3.1.1 (Initiating Events) which indicates that the frequency of this event was added to the "TLMFW." Please identify the frequency of Loss of Instrument Air (LOIA), and the source, i.e., whether the frequency was obtained from generic or plant-specific data.

15. Discuss the impact of LOIA on front line and support systems designed to mitigate the effects of failures sustained during or after a trip, and the rationale used in combining the event with TLMFW as opposed to treating it as a unique initiating event.

16. Discuss the technical basis or provide a reference for "assuming" that very small LOCAs" (Less than ½-in equivalent diameter) are within the

makeup capacity of the normal charging system, and therefore, these events could be "conservatively" included with "small LOCA" Initiating Events (Page 3.1-7 in Section 3.1.1).

17. Discuss the impact of LOCAs or Steam Line Breaks on mitigating systems as initiating events.

18. Unlike the information provided for component data, there is no discussion or identification of plant-specific data used in the "updating process" for initiating events.

    a) Provide a listing of the frequency of initiating events (e.g. Turbine trip, Reactor Trip, Loss of Offsite Power/Main F.W/Inst. Air) that were obtained from plant operating experience as opposed to those arrived at through system analysis.

    b) Include a discussion of the updating process for the initiating events and a discussion of the frequency of those events whose total frequency is made up of multiple events (e.g., TLMFW).

    Section 1.1 states that in 1991 DLC developed a plant-specific database and used it to requantify the Unit 2 PRA model. However, Section 3.3.2.1 indicates that the plant-specific data presented and discussed in Section 3.3.2 was collected between 11/87 and 12/88.

    c) Has the data presented been captured through 1988 or 1991?

    d) Is the PRA model quantified using plant-specific data different from what is presented in the IPE?

    e) If the PRA model has been quantified using plant-specific data through 1988, please provide a discussion of any plans to update the database and the PRA model and any component failures or initiating events occurring since 1988 which would impact the IPE results.

19. Generic Letter 88-20 and NUREG-1335 request that the IPE submittal provide a list of all generic plant data for equipment and initiating events, including origin and method of analysis.

    Since Section 3.3.1 indicates that for a majority of components the generic component failure rates were taken from "Database for PRA of Light Water Nuclear Power Plants" PLG-0500, 1989, and since this document is not in the public domain, please provide a listing of the generic component failure rates used for the BV-2 IPE (or the PLG database used in the analysis). This list should include those generic values used as a basis for updated values.

20. In verifying that the submittal contained a listing of initiating event frequencies it was noted:

    That the system initiating event frequencies in Table 3.1.1-3 were different from the values provided in Table 3.3.5-2.

    A constant value is displayed for all parameters of the distribution for Initiating Events: WAX, WBX, and WXB.

    Explain these apparent discrepancies and provide a discussion regarding any possible impacts on the results presented in the IPE due to these discrepancies.

21. The Internal Flooding Analysis indicated that mitigating features such as redundancy and separation were considered. However, actual operating experience has demonstrated that separate rooms do not necessarily provide protection because of drain systems that are plugged or allow backflow, unsealed doors, or maintenance actions or situations. Discuss how consideration was given to these conditions in the flooding analysis, and how they impacted the choice or quantification of initiating events.

22. Sections 1.4 (Summary of Major Findings), 3.3.8 (Interior Flooding Analysis) and 4.8 (Back-end Results) do not characterize the impact of internal flooding events either as important or not significant. However, Fig. 4.8-1 shows that Control Building Flood (CBFL) events contribute approximately 6.6% of the "small early containment failures or bypasses" which is the 3rd largest contributing initiator.

    Provide a discussion of the flooding analysis addressing whether the process yields non-conservative, realistic or conservative estimates and DLC's assessment of the IPE conclusions in light of this, especially with regard to CBFL.

23. It is noted that in the discussion of top events DO, DP, IE, IB, IW, and IY the time that power is specified to be available is dependent on "How Long The Batteries Last" and is identified as either 3.5 or 8 hours. However, the system's description for DC Electric Power (Section 3.2.1.2.9) states the assumption that following a loss of AC power DC power is evaluated for a mission time of just 2 hours. The BV-2 FSAR Chapter 8 also indicates that the life of the batteries under design loads is 2 hours.

    Discuss the technical basis or provide a reference for the assumption of battery life longer than 2 hours as relates to the top events above.

24. In Section 3.1.3.1 (General Transient/Small LOCA Tree) under the description provided for top event CI (Containment Isolation) a discussion is provided which relates to the Seal LOCA Model. However, the discussion and Section 3.3.3 (Human Failure Data) which is referred to therein as containing the Seal LOCA Model, do not explicitly describe the Model used for the IPE submittal.

Provide a discussion of the Seal LOCA Model as used in the BV-2 submittal including the various leak rates, timing of seal failure and the probability of their occurrence with and without the seal return line isolated.

In addition, discuss the impact on core damage frequency (CDF) if the assumption is incorrect that the low pressure seal leakoff pipe will withstand high pressure on failure of the number one seal.

25. Section 3.4.3 of the submittal provides information on the importance of the five systems that perform decay heat removal (DHR) functions and indicates that no particular vulnerabilities have been found. However, the values provided in Table 3.4.3.-1 as the "percentage of CDF in which event is failed" show a non-negligible contribution for some top events due to loss of support (e.g., MFF 9.7% and AFF 20.2%). A value for HHF (High Head Safety Injection Pumps, Support Unavailable) is not provided; however, Table 3.4.2-1 shows the percentage of CDF with this split fraction as 62%.

    Generic Letter 88-20 and Appendix 5 therein, indicate that support systems are important to the DHR Function and suggests that they be considered in the search for DHR related vulnerabilities. Therefore, please discuss the impact of support systems on these five systems differentiating between the contribution from Loss of Power (LOSP & BVX) and other supports such as Service Water, Primary Component Cooling Water and Instrument/Containment Instrument.

26. Table 3.4.3-1 shows the percentage of CDF in which the event AFF is failed as 20.2% (3.84E-5) identifying it as due to Large Flood in safeguards Area. However, Figure 3.3.8.2 (comparative contributions to core damage from floods) shows that only 16.6% of the CDF from all floods (7.32E-6 x 0.166 = 1.22E-6) is due to safeguards floods. Provide a discussion of this apparent discrepancy and other values in the table which may likewise impact the results of the IPE.

27. As indicated in the paragraph on feed and bleed cooling, the BV-2 design "minimizes the frequency of sequences involving failure of AFW and Bleed and Feed Cooling relative to other PWRs previously studied" because of credit taken for realigning the electric motor-driven MFW pumps. It would appear that this capability is of significant benefit to BV-2.

    Discuss the benefit derived from this capability in terms of CDF with and without this capability. In concert with this, please provide the benefit derived from the capability to feed and bleed upon loss of all secondary cooling (i.e., MF & AF) in terms of CDF with and without this capability.

28. Provide a list of the types of initiating events identified as "other" in Figure 3.4.0-2 and the breakdown of their contributions to CDF.

29. The submittal identified core damage as having occurred when loss of core heat removal progressed beyond the point of core uncovery <u>and</u> core exit temperatures exceed 1200°F.

    How many sequences were screened out because of this double criteria? Discuss the impact on the resultant CDF obtained using this criteria.

    Please address the following:

    - The basis for the temperature chosen (1200°F).

    - Do all sequences with the core uncovered go to core damage, or was there recovery prior to reaching 1200°F?

    - Would the CDF be significantly different without the 1200°F core exit temperature criterion?

30. The BV-2 submittal has identified loss of emergency switchgear room HVAC as a significant contributor to CDF, due to the relatively rapid rise in room temperatures that will exceed the qualification temperature of equipment in the room. However, experiences of other plants have indicated that temperature rise determined by test on loss of HVAC is not as rapid as determined by calculation.

    The possible prediction by calculation of temperature rises significantly more rapidly than might be experienced could cause a distortion in the identification of contributors to CDF and subsequent misapplication of resources. Is DLC giving consideration to verification of the rate of temperature rise determined for the emergency switchgear room on loss of HVAC, to establish if the contribution from this event appropriate?

31. Section 6.1 indicates that the two risk factors of merit that have been considered are CDF and early release frequency. In addition, Section 6.3.1 states that in order to determine vulnerabilities the major accident "CATEGORIES" were evaluated along with top ranking sequences.

    a) Provide the definition of vulnerability and describe the process used in conjunction with the above to identify the vulnerabilities as requested by NUREG-1335.

    b) Discuss the findings related to identifying potential vulnerabilities with respect to containment failure or by-pass and assessing any associated plant modifications.

    c) Discuss the anticipated benefit (decrease in CDF or impact on release category), the rationale by which the listed option was chosen from the potential options, and the respective timing of implementation for those "under review."

    d) Discuss the consideration given to independent failure of the Service Water Headers (WA and WB involved in 13.7% CDF and in top ranking sequences involving small LOCAs which contribute 21% to CDF) and the common check valve in the suction of the HHSI pumps (VL-1, involved in approximately 15% CDF, and also in top ranked sequences involving loss of vital bus and small LOCA) as vulnerabilities.

32. Discuss briefly the IPE results (including the contributions to CDF) of any analysis related to a small break LOCA due to a stuck-open safety valve event if the PORVs are blocked off to stop any leakage. The discussion should address the percentage of time the PORVs are blocked off due to leakage and failures of operator actions to open the PORV block valve during accident conditions.