September 12, 1995

Mr. Terry L. Patterson
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 399
Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska 68023-0399

Dear Mr. Patterson:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) TO ASSIST CLOSURE OF NRR STAFF REVIEW OF FORT CALHOUN NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NO. M74412)

By letter dated December 1, 1993, Omaha Public Power District submitted its Individual Plant Examination (IPE) for Fort Calhoun Station in accordance with Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and NUREG-1335, "Individual Plant Examination Guidance." The NRC is reviewing your submittal and has determined that additional information is needed.

The enclosed questions were discussed by the NRC staff and OPPD staff on August 23, 1995. As discussed during the telecon, we anticipate your responses to our request for additional information to be submitted by December 1, 1995.

This requirement affects fewer than nine respondents and, therefore, is not subject to Office of Management and Budget review under Public Law 96-511. If you have any questions about this matter, please do not hesitate to call me at 301-415-1313.

Sincerely,
ORIGINAL SIGNED BY:
Steven Bloom, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure: Request for Additional Information

cc w/encl: See next page

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J. Dyer, RIV	J. Roe	R. Hernan

DOCUMENT NAME: FC74412.RAI

OFC	LA:PD4-2	PM:PD4-2
NAME	EPeyton	SBloom:pk
DATE	9/11/95	9/12/95

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

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Request for Additional Information About the IPE Program

Fort Calhoun Station

1. The submittal is not clear regarding the date to which plant operation and procedures are represented by the IPE analysis (freeze date). In addition, the submittal does not indicate whether exceptions to the freeze date configuration were included in the analysis.

   (a) Please identify the freeze date of the analysis.

   (b) Please identify any exceptions to the freeze date configuration.

   (c) Please identify the effect of any of any freeze date exceptions on the estimate of the core damage frequency (CDF), both individually and collectively.

2. Frequencies for the following initiating events are approximately an order of magnitude lower than corresponding data typically used in other IPE probabilistic risk assessment (PRA) studies: turbine trip, loss of main feedwater, medium loss-of-coolant accident (LOCA) and large LOCA. Also, although the submittal describes the treatment of automatic scrams, it is not clear how manual scrams from full power were considered in the IPE.

   (a) Please discuss the quantification of the initiating events for turbine trip, loss of main feedwater, medium LOCA, and large LOCA, specifically addressing sources of data, methods used for quantification, and applicability to the Fort Calhoun plant.

   (b) Please describe how manual scrams from full power were accounted for in the IPE.

3. The submittal does not give a definition of core damage. Please define the term "core damage" as it is used in the IPE.

4. The IPE assumes that low-pressure safety injection pumps are not required for the mitigation of a large LOCA. Although flow from the high-pressure safety injection (HPSI) system can match decay heat removal requirements in the long term, it is not clear how core damage can be prevented in the early phase of the accident with flow from one of three HPSI pumps and three of four accumulators as assumed in the IPE. Many other PWR IPE/PRA studies have assumed that LPSI pumps are required for mitigating large LOCA. Please give the basis for this portion of the IPE success criteria; specifically addressing the expected peak cladding temperature (if available) and the extent of any radionuclide release.

5. The IPE does not include total loss of dc power as an initiating event. Please give the reason for excluding this initiating event from the IPE.

6. The submittal states that the interfacing-systems (ISLOCA) events represent piping or valve failures. No mention is made of any consideration given in the IPE to failures of pump and valve seals and gaskets. Other IPE and PRA studies have shown that seal and gasket failures are important contributors to ISLOCA frequencies. Also, the analysis of the reactor coolant pump (RCP) seal cooler ISLOCA does not address the potential for losing high-pressure injection as a result of adverse environmental conditions created by the ISLOCA.

   (a) Please discuss the consideration given to failure of seals and gaskets in the development of the IPE ISLOCA models. If seals and gaskets have not been accounted for in the ISLOCA analysis, please provide justification.

   (b) Please explain how it was determined that high-pressure injection could be made available for the RCP seal cooler ISLOCA given possible adverse environmental effects of coolant discharged outside the containment.

7. As indicated on page 3.1-57 of the submittal, the consideration of RCP seal LOCAs was limited to primary system leak rates in excess of the charging makeup capacity (120 gpm). An RCP seal LOCA having a leak rate greater than 120 gpm is postulated to occur either as a result of failure of all four seals in one RCP, or common cause failure of three or four seals in all RCPs. The model does not consider smaller leak rates, for example, a 35-gpm leak associated with the failure of three or four seals in a single RCP. The exclusion of smaller RCP seal LOCAs from the IPE model may underestimate the total CDF. For example, during a station blackout all of the electric-driven pumps will be disabled, as a result, charging flow and component cooling water will be disabled. The loss of component cooling water may in turn cause an RCP seal LOCA, possibly a LOCA less than 120 gpm. If an extended station blackout condition were to exist, the lack of makeup flow to the primary system would eventually cause core uncovery. Please explain how it was ensured that a vulnerability or important CDF contribution was not missed because of the exclusion of RCP seal LOCAs less than 120 gpm.

8. The submittal does not give a complete breakdown of CDF by initiating event. In addition, it does not give the CDF contribution from station blackout.

   (a) Please give the CDF contribution from each initiating event listed in Table 3.8 of the submittal.

   (b) Please give the station blackout contribution to CDF.

9. The transmittal letter for the submittal states that a number of improvements, as well as minor modifications, were identified and implemented as a result of the analysis. It further states that future areas for plant improvement are also under consideration. Although Section 6.0 of the submittal describes improvements related to the IPE

analysis, it is not clear that this description reflects the current plant status with regard to modifications and improvements. Please clarify the information in the submittal 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, that is, whether the improvement has already 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 in 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); and

(e) the basis for each improvement, that is, whether it addressed a vulnerability, was otherwise identified from the IPE review, or was developed as part of other NRC rulemaking, such as the station blackout rule.

10. It is not clear in the submittal if plant changes due to the station blackout rule were credited in the analysis.

(a) Please state 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 identify the specific plant changes that were credited.

(b) Please identify the total effect, if any, of these plant changes on the total plant CDF and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF).

(c) Please identify the effect, if any, of each individual plant change on the total plant CDF and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF).

(d) Please identify any other changes to the plant implemented or planned to be implemented and separate from those in response to the station blackout rule that reduce the station blackout CDF.

(e) Please state whether the changes in item d are implemented or planned.

(f) Please state whether credit was taken for the changes in item d in the IPE.

(g) Please identify the effec., if any, of the changes in item d on the station blackout CDF.

11. The submittal indicates that a cut set frequency threshold of 1E-9/yr or less was applied to quantified flooding sequences. However, it does not indicate the accident sequence cut set threshold applied to the other portions of the front-end analysis. Please give the truncation value applied to accident sequence cut sets in the remaining portions of the front-end analysis.

12. The IPE modeled four separate categories of events representing loss of offsite power (LOSP): loss of 345 kV with 161 kV unavailable (plant-centered); loss of 161 kV with failure to fast transfer (plant-centered); grid-related LOSP; and weather-induced LOSP. According to the submittal, non-recovery probabilities for these LOSP initiating events are based on data from an Electric Power Research Institute (EPRI) document (EPRI 6780). However, the submittal does not provide a complete set of the LOSP non-recovery data used in the analysis. Please provide the IPE non-recovery data as a function of time for each of these four LOSP initiating events.

13. The submittal states that plant-specific component hardware data were gathered for 16 systems modeled in the IPE analysis. However, only plant-specific data pertinent to the auxiliary feedwater system are presented in the submittal.

(a) Please give the plant-specific failure data gathered for the following components and failure modes: diesel generator - start and run; HPSI pump - start and run; LPSI pump - start and run; raw water pump - start and run; component cooling water pump - start and run; emergency core cooling system motor-operated valve fail to open, fail to close; battery - failure frequency; battery charger - failure frequency; and circuit breaker - fail to open, fail to close.

(b) For each of the components and failure modes identified above, please identify the source of data used to support the IPE analysis (plant-specific and/or generic).

(c) If any of the above component failure modes are based solely on generic data, please provide justification.

14. Please give the failure data used for the diesel-driven auxiliary feedwater pump and discuss how recent experience with pump vibration supports the IPE failure data used for this pump.

15. Table 6-1 of the submittal describes a plant enhancement involving the installation of a door to mitigate ISLOCA effects of an RCP seal cooler rupture.

    (a) Please clarify this plant enhancement by identifying the location of this door and how closure of the door will isolate a rupture of the CCW boundary.

    (b) Also, please identify the major equipment items that would be protected by closing this door.

16. The submittal does not clearly discuss the process used to identify and select pre-initiator human events including those involving failure to properly restore instrumentation to service after test and maintenance, and miscalibration of instrumentation. The process used to identify and select these types of human events may include the review of procedures and discussions with appropriate plant personnel on interpretation and implementation of the plant's test, maintenance, and calibration procedures.

    (a) Please describe the process used to identify human events involving failure to properly restore to service after test and maintenance, and miscalibration of instrumentation.

    (b) Please give examples illustrating this process.

17. It is not clear from the submittal what was the justification for the screening of pre-initiator human events to ensure that the screening process did not eliminate potentially important human events and accident sequences. The rationale presented for a screening probability value of 0.003 (for example, the rationale provided on page 3.3-26 of the submittal that it is higher than "typical" THERP analyses of 1E-4 to 1E-6) is not understood. For example, analyses of licensee event reports (LERs), such as those performed for failures of valves (e.g., NUREG/CR-1363, Rev. 1), indicate that human-caused failures are about 1E-3.

    The concern is that, by using a value of 0.003 for screening purposes, the screening process could inadvertently cause important pre-initiator human events to be eliminated from further analysis. For example, in the development of initial systems fault trees, it is common for different systems analysts to identify the same or similar human actions under different event labels. If the screening takes place without these events being explicitly recognized as the same, it is possible to inadvertently eliminate important accident sequences, particularly when a screening value such as 0.003 is used.

    Please provide an additional discussion concerning the screening process used to ensure that important sequences and human events were not inappropriately screened out. Specifically, please discuss how the important human events were not erroneously eliminated by the use of such a screening value.

18. It is not clear from the submittal what plant-specific performance shaping factors were used for modifying nominal pre-initiator human error probabilities.

    (a) Please provide a list of the plant-specific performance shaping factors and their associated values that were used to modify the nominal pre-initiator human error probabilities.

    (b) Please include a description of the process used in the assessment of the performance shaping factors. For example, this description could include examination and walkthroughs of procedures, interviews with plant personnel, examination of administrative controls, and evaluations of displays and controls.

19. In the analysis of pre-initiators, event KJUMPER (failure to remove reactor protection system interposing relay jumpers before going to full power operations) is assigned a probability of 1.3E-6. Actions with such low estimated values typically have associated characteristics such as redundant indications, independent operator checks, compelling signals or alarms.

    Please explain, with example calculations, how such a low failure probability is to be achieved in practice at Fort Calhoun.

20. The submittal gives no description of the plant-specific experience of pre-initiator human events.

    Please compare any operating experience associate with pre-initiator human events (e.g., data from LERs or other plant records) to the failure probabilities calculated in the HRA modeling. In other words, please describe to what degree the results of the modeling of pre-initiator human events represent experience at Fort Calhoun.

21. In Section 3.3.3.3, "HRA Quantification Methods," the submittal describes dependency guidelines for the time-independent HRA method. In particular, guidelines are given for assessing dependencies among plant personnel, such as the shift technical advisor (STA) and the shift supervisor.

    (a) Please confirm that this set of dependencies was only used for post-initiator human events

    (b) Please describe the guidelines used in assessing of interpersonal dependencies used for pre-initiator human events.

    In addition, in the discussion of dependencies among plant personnel, such as the STA and the shift supervisor, the submittal states that the guidelines may be varied in the case of specific analyses.

(a) Please identify during which (if any) events were the guidelines varied.

(b) Please describe how the dependencies were modeled in those event analyses.

(c) Please give the rationale for the changes from the guidelines.

22. Three different correlations of reliability of human response with time are described in the submittal. These are described as the "basic model," the "rule-based model" and the "verification model," in Section 3.3.3.3 of the submittal. In the basic model, a median response time of 4 minutes is used; in the rule-based model, a 2-minute median response time; and in the verification model, a 1-minute median response time. The submittal discusses the use of the rule-based model as being applied to "any strong symptom-oriented rules within the EOPs [emergency operating procedures]." This description is not clear.

(a) Please provide an additional discussion of how these different models were used in the analysis.

(b) Please give the criteria for selecting the model used in an analysis of a particular human action event.

23. It is recognized, as indicated in the submittal, that the data used in time-reliability-based HRA methods are essentially judgmental. The technique used in the Fort Calhoun analysis is not unique in that regard. However, data from simulation studies such as the EPRI operator reliability experiments (OREs) and NRC's RMIEP (risk methodology integration and evaluation program) recovery actions described in NUREG/CR-4834 have indicated a potential for significantly longer median response times than those used in the submittal, particularly for the so-called "rule-based actions" that assume a 2-minute median response. If longer median response times are assumed or different time-reliability correlations are used, the probabilities of failure can increase significantly.

(a) Please explain how the value for the time available for operator actions was selected to represent the range of detailed accident conditions implicit in the sequences for which the value of "time available" was used to calculate human error probabilities. For example, the value may represent a bounding condition or may represent a more typical or mean value.

(b) To the extent possible, please indicate the major sources of uncertainty in the estimates of the available time, particularly those that might significantly reduce estimates of the available time.

(c) Please illustrate your response by indicating the times available and the bases for these times for the following events taken from Table 3.3.3.1:

- OPER-10, "Human Failure To Achieve Shutdown Cooling"
- OPER-4, "Human Failure To Initiate Feed & Bleed (Transients Except T4)"
- OPER-9, "Operator Fails To Terminate Faulted Steam Generator Leakage"

24. The description of the analysis of particular post-initiator human events is very limited. In one instance where sufficient data are supplied to verify the probability of failure for a particular event, there appears to be an error. In particular, the probability of operator action OPER-8, "Human Failure To Initiate Feed-and-Bleed during SLOCA," is quantified as 9.1E-6 in the submittal (see Table 3.3.3.1). On the basis of the information in Figure 3.10, the event consists of an unburdened action with a time available of 25 minutes. Preliminary check calculations indicate, in fact, that action OPER-8 may have been quantified using the "verification" time/reliability model, whereas Table 3.3.3.1 indicates that this action is a "type 4" action; a rule-based action taken in the control room. Failure of an unburdened rule-based action within 25 minutes is estimated to be approximately 1E-4. Such a value would be more consistent with other time/reliability based methods.

    To be assured that the appropriate time/reliability based models were used in the analysis of human actions, please describe the analyses of those operator actions listed as significant in the sensitivity analyses in Section 3.4.5.

25. In the illustration of the "SAIC TRC" in comparison with publicly available simulator data (Figure 3.3.3.5), the shading appears to indicate "public data" representing failure probabilities as low as 10^{-6} for times as short as 0.5 minute.

    (a) Please identify the sources of such data.

    (b) Please describe how they are relevant to the analysis of post-initiator human actions whose reliability models they are being claimed to support.

26. The analysis of dependencies between time-dependent events is not clearly described in Section 3.3.3. In particular, the discussion of time factors on Page 3.3-40, item 4, is not understood.

    (a) Please give an expanded discussion of this issue, and illustrate your response with examples if more than one post-initiator human event has been modeled in an accident sequence.

(b) Please include a summary of those sequences where multiple post-initiator human events are incorporated and the overall probabilities of the human events in combination.

27. In the description of LOCA sequences in the submittal, several human actions are identified that are not discussed (or even listed) in the analysis of human actions (Section 3.3.3). A review of the fault trees (Figures 3.36 - 3.42) associated with the ISLOCA indicates that the following human events were omitted from Table 3.3.3.1:

    OPER-60, -61, -65, -70, -71, and -101

    Please review your analysis of ISLOCA events and list all human events included in that analysis, together with their assigned probabilities and the bases for those assigned probabilities.

28. In the analysis of internal flooding events (Section 3.3.6), several human actions are identified that are not discussed in the analysis of human actions (Section 3.3.3). For example, in the analysis of internal flooding events, item 26 on page 3.3-70 states: "Human error and non-recovery events were examined and the internal events probabilities for these human events were adjusted upwards as required based on the perceived scenario effects on human performance."

    In addition, the description of important flood-related accident sequences in Section 3.3.6.3, several human events are identified as contributors to several sequences. It is not clear from the submittal (because of the brevity of the descriptions of human events) whether these human events are listed in Table 3.3.3.1.

    (a) For the sake of clarity, please describe all flood-related human events, including the basis for the assignment of probabilities of these events.

    (b) Please include a list of those internal human events that were adjusted, the adjusted probabilities, and the basis for the adjustments.

29. The submittal is not clear as to what kinds of recovery-type actions were considered. These actions can include those performed to recover a specific failure or fault and for which procedures may not have been established. For example, suppose the EOP directive instructs the operator to maintain level using system x, but the system fails to function and the operator 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 submit a list of the recovery actions considered in the analysis, and indicate whether they are actions for which procedures have been established and what evaluations were performed to ensure that the necessary actions can be accomplished within the available time.

(b) Please illustrate your response with two examples indicating how such events were quantified.

30. For the containment event trees, the submittal describes the analysis of the recovery of containment spray systems, event SPRAYRECOV (Section 4.6.9.1.3). In the analysis of event SPRAYRECOV for events where containment spray is operable but not operating (containment spray status "CSO" in the plant damage state definitions in Table 4.3.1.3), the probability of the containment spray system being started is assigned a value of 1.0 (i.e., the probability of failing to start the containment spray system is 0). It is not clear from the description whether the containment spray system is started automatically or manually in this event.

(a) If the initiation is manual, please describe the procedures and operator training associated with this action, and explain why failure to perform this action is negligible.

(b) Alternatively, please describe what would be the consequence in terms of risk of assigning a possibly more realistic probability of failure to this event (e.g., 0.1).

31. Table 5.3.2, of the submittal does not indicate any comments associated with HRA.

Please confirm that the external peer-review process included the HRA methods and results.

32. Section 3.4-11 of the submittal describes how the IPE team screened the front-end results using the guidelines in NUMARC 91-04 to identify possible core damage and containment bypass vulnerabilities applicable to Fort Calhoun. Section 7 of the submittal notes that the IPE team found no vulnerabilities at Fort Calhoun.

Because it was not clear from the submittal, please describe whether and how the IPE team screened the back-end results to identify possible containment vulnerabilities applicable to Fort Calhoun.

33. Figure 4.7.2.4, page 4.7-51 of the submittal, shows that isolation failures, including steam generator tube ruptures (SGTRs), contribute to 5.64 percent of the total CDF at Fort Calhoun. Many PRAs categorize SGTRs as separate from containment isolation failures and as containment bypass events.

(a) Please describe the process used to determine the frequency of isolation failures other than SGTRs.

(b) Please give the containment failure size and the corresponding release rate assumed for isolation failures.

34. In applying the one-dimensional heat transfer calculation results for the multi-dimensional, external cooling of the reactor vessel bottom head, the IPE submittal states the following (page 4.2-5): "while the net impact of multidimensional effects is detrimental to the cooling process, the resulting stresses will still be approximately represented by the bounding 50% [of decay heat] downward heat flux calculations shown in Figure 4.2.1.3."

    Please describe the mechanisms removing the remaining 50 percent of decay heat that is not transferred downward.

35. Table 4.3.2.1 of the submittal shows the mapping of plant accident sequences into three PDS bins: bin 1, bin 2, and bin 3. However, these bins are not defined.

    Please define these bins and indicate what bearing they have on accident progressions at Fort Calhoun.

36. Section 4.1.2.7, page 4.1-29 of the submittal, states:

    An engineering analysis (EA-FC-9226) was done to determine the ability of instrument and power cable to withstand extreme temperature. The cables are rated from the manufacturer to be able to survive 100 hours at 266 °F. Testing was also done to determine that cables could withstand 700 °F for a short period of time such as would occur with hydrogen burn.

    Please discuss the survival of other pieces of equipment necessary to mitigate core damage and radionuclide releases under the harsh environments of severe accidents that are possible at Fort Calhoun.

37. Section 4.2.2.1.2.3, page 4.2-20 of the submittal, notes that operator activation of the power-operated relief valve (PORV) was assessed not only to be possible but highly likely at Fort Calhoun. However, in contrast, Section 4.6.4.1, page 4.6-5, states the following: "Since no procedures currently exist to ensure the operator depressurizes the Reactor Coolant System (RCS) to minimize RCS pressure prior to vessel break, the operator use of the PORV was neglected."

    (a) Please explain this apparent discrepancy.

    (b) What is the sensitivity of the radiological release results, assuming that the operator successfully depressurizes the RCS.

    (c) Please identify any other operator, recovery, or mitigation actions that are important for the back-end analysis and describe how they were evaluated.

38. Table 4.8.2.4, pages 4.8-19 through 4.8-22, lists, with extensive descriptions, the Fort Calhoun dominant release classes. Because many release classes are listed in this table, it is difficult to relate the overall effect of different plant damage states to containment performance.

    In order to understand the effect of RCS pressure at the onset of core damage, please provide the frequency of releases in terms of early, late, and containment intact for the accident sequences with RCS pressures in the ranges "at SRV (safety relief valve) pressure," "high," "intermediate," and "low."

39. As a result of the containment performance improvement program, recommendations were made for licensees to consider it as part of the IPE process. These recommendations were identified in Generic Letter 88-20, Supplement 3. The recommendation applicable to Fort Calhoun is as follows: "Licensees with dry containments are expected to evaluate containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures) as part of the IPE."

    (a) Please describe the way in which you responded to the above recommendation, the plant improvements identified, if any, and your plan to implement the improvements.

    (b) Please describe the criteria used to determine if implementation of CPI program recommendations was warranted.

    (c) Please give the technical bases for the plant improvements or the technical bases for determining that no plant improvements were needed.

    (d) Please include a listing of all potential equipment vulnerabilities to localized H_2 combustion.

40. The following requests for information are applicable to the containment strength evaluation results reported in the submittal:

    (a) Section 4.1.2.1, page 4.1-8, does not give a definition of high confidence of low probability of failure (HCLPF). Please give your definition of HCLPF.

    (b) Section 4.1.2.1 notes that the Fort Calhoun containment has a median failure pressure of 190 psig; Table 4.1.2, page 4.1-13, lists it as 215 psig. Please explain this apparent discrepancy.

    (c) Table 4.1.2 lists the median failure pressures of the Fort Calhoun, Surry, and Zion containments as 215, 120, and 134 psig, respectively, which indicates that the Fort Calhoun containment outperforms those of Surry and Zion by significant margins. Please describe the particular structural characteristics of the Fort

Calhoun containment that causes it to outperform those of Surry and Zion.

(d) Figure 4.4.3-10, page 4.4-23, shows three containment fragility curves with 95 percent, 50 percent and 5 percent confidence levels for the bending failure of the containment structure (all modes included). Please state which fragility curve was used to determine the failure probability of the containment.

(e) Section 4.4.3, page 4.4-5, notes that the containment capacity under overpressure load was evaluated by performing finite element analysis and that "local models" were developed to evaluate the areas of the penetrations.

- Please explain these local models noting whether they were finite element models, and the methodologies and failure criteria used.

- Please give the criterion used to determine the failure of the containment using the global finite element model.