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

JAN 22 1979

Docket Nos. 50-373
and 50-374

Mr. L. O. DelGeorge
Nuclear Licensing Administrator
Boiling Water Reactors
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

SUBJECT: STATUS OF STAFF REVIEW REGARDING LA SALLE COUNTY STATION, UNITS 1 & 2

In our letters of October 16, 1978 and December 21, 1978, we advised you of those matters identified as outstanding issues in the SER inputs as of those dates. The enclosure to this letter provides a listing of all outstanding matters identified since December 21, 1978. We have yet to receive SER inputs for Instrumentation and Control and Quality Assurance. The outstanding issues associated with these two areas will be made available to you in the near future. Additionally, the issues associated with the Reactor Systems area were based on a draft SER input which could change somewhat when the final version is received.

In our two previous letters we have referenced each outstanding issue to a question or questions which were provided to you at the first request or second request for additional information stage. We are continuing that practice in the enclosure in this letter. You will note, however, that several of the outstanding issues are not referenced to previous requests for information. These items have been identified late in the review and represent matters for which questions have not been issued.

Please contact us if you desire any discussion or clarification of these matters.

Sincerely,

Olan D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosure:
As stated

ccs w/enclosure:
See next page

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ENCLOSURE

OPEN ITEMS

1. Due to malfunction of equipment, we do not have a complete second year meteorological data as required by Regulator, Guide 1.70. The additional data were scheduled to be submitted in December 1978. (Reference Question 372.11)

2. Although we conclude that the GEXL correlation is acceptable for initial core load, we are concerned that GEXL correlation may not be conservative for reload operation. Therefore we require boiling transition data for 8x8 fuel bundles with two water rods be provided for our review in order to support the use of the GEXL correlation beyond the first cycle of operation. The operating license will be conditioned accordingly. (Reference Question 221.10)

3. Your hydrodynamic stability analysis performed was for the first cycle conditions only. We require that a new analysis be submitted and approved prior to second cycle operation indicating the results for appropriate exposure core conditions. The operating license will be conditioned accordingly.

4. As a result of eleminating the control rods drive systems return line we are reviewing generically with regard to the impact on the control rod drive system performance. Consequently we require you submit system performance data directly applicable to La Salle and will require you to conform to the conclusions of the generic study as applicable to La Salle. (Reference Question 212.145)

5. You stated that the operator has at least 20 minutes before scram capability would be impaired. In order to accept the 20 minute period available, we require a plan for periodic testing of the accumulators in the control drive hydraulic systems to assure that there is at least 20 minutes available before scram becomes marginal. (Reference Question 212.140)

6. Sufficient information has not been provided on the ability of the level monitoring and alarm systems to meet the single failure criterion with regard to protection from flooding in safety-related areas, or on the capability of the systems to handle flooding due to pipe breaks. (Reference Question 010.11)

7. Sufficient information has not been provided to complete our review in QA the initial test programs. The following areas are enumerated:

   (a) We can not conclude that testing will be performed in accordance to Regulatory Guides 1.20, 1.68, 1.68.2 and 1.80.

   (b) We have not completed our review of test plans for systems and conponents to mitigate ATWS.

(c) Responses to our positions were not satisfactorily and additional information is required.

(d) Additional information is required for 30 preoperation tests and 16 startup tests abstract.

8. The forces in the reactor pressure vessel affecting the design of the support skirt are acceptable. However, additional evaluations may be necessary following completion of our task action plan A-2.

9. Preservice and inservice inspection of Class 1, 2, and 3 components have not been submitted. (Reference Question 121.4)

10. Your approach to calculating the suppression pool bypass is not consistent with Branch Technical Position CSB 6-5. You must commit to perform a low power surveillance leakage test of the containment at each refueling outage. (Reference Question 021.43)

11. We have completed our review of the short-term program and developed acceptance criteria. We require that you commit to our acceptance criteria or justify any exceptions taken. (Reference NUREG-0487)

12. We require a leak test of the secondary containment to verify the inleakage assumption and the drawdown time for reestablishing the -0.25 inches of water gauge. (Reference Question 021.11)

13. Since the reactor building closed cooling water systems, which you have identified as containment isolation barrier, are designed to Quality Group D. requirements, we require inservice inspection of the systems in accordance with Section XI of the ASME Code these periodic inspection will assure the reliability of these system.

14. You stated that a 2-inch vent line exists in the purge system to bleed-off excess primary containment pressure during operations. We require you to evaluate this 2-inch bypass purge system in light of the criteria of Branch Technical Position CSB-6-4. (Reference Question 021.54)

15. Your proposed combustible gas control system is designed in accordance with requirements of 10 CFR Part 50.44. However because of certain system characteristics we require that you commit to the following:

(a) If the containment pressure is above 15.3 psig and the hydrogen concentration is 3.3 volume percent, the containment spray system must be actuated to reduce the containment pressure.

(b) Following a LOCA, the recombiner system becomes an extension of the containment boundary. We require that the leak tight integrity of the recombiner be demonstrated. (Reference Question 021.58)

16. In order to complete our review, we require the additional information requested relating to containment leakage testing to show compliance with Appendix J. (Reference Questions 021.9, 021.34, 021.35, and 021.59)

17. Verify that the suction lines in the suppression pool leading to the ECCS pumps are designed to preclude adverse vortex formation and air injections which could effect the pumps performance. (Reference Question 11.2.127)

18. Instrumentation is not sufficiently sensitive to detect voids at the top of ECCS pipe lines. Provide adequate instrumentation to assure filled ECCS line. (Reference Question 112.134)

19. You must show that the air supply for the ADS is sufficient for the extended operating time required and assure us by reliability data that the ADS valves will function as required. (Reference Question 112.132)

20. Assurance must be provided to indicate that the ECCS pumps can function for an extended time (maintenance free) under the most limiting post-LOCA conditions. (Reference Question 112.128)

21. Reevaluate your LOCA and simultaneous flow control valve failure considering more realistic valve closure dynamics. (Reference Question 112.143)

22. Show that adequate time is available for operator action to restore core cooling prior to excessive core headings as a result of a crack in a residual heat removal line. (Reference Questions 112.92 and 112.133)

23. Recent results obtained from the Two Loop Test Apparatus are being reviewed by us to ascertain the functional performance of the emergency core cooling system. When we have completed our assessment we will report any concerns resulting from our review.

24. Demonstrate that adequate core cooling is maintained when the low pressure coolant injection diversion is considered. (Reference Question 212.126)

25. For the staff to accept the control room habitability system against potential airborne radiation, you must commit the control room receives a high radiation alarm from the outside intakes, and (2) yearly test the filter train in conjunction with the testing once-through charcoal filter. (Reference Question 312.23)

26. Additional information is required both for the qualification tests and operating experience with your safety/relief valve. (Reference Question 212.131)

27. You have not submitted the analysis for the trip of the recirculation pumps at high pressure to mitigate ATWS. (Reference Question 212.136)

28. Additional information is required to determine whether the RCIC pump suction has to automatically switched from the condensate storage tank to the suppression pool in the event of a safe shutdown earthquake and concomitant failure of the condensate storage tank. (Reference Question 212.141)

29. Show that the reactor core isolation cooling system will not be shutdown unintentionally because of spurious temperature signals (Reference Question 212.130)

30. Show how you intend to detect leakage from the reactor coolant system into both the low pressure coolant injection (3 trains) and low pressure core systems. (Reference Question 212.138)

31. The valves which serve to isolate the residual heat removal system from the reactor coolant system should be classified category A/C in accordance with the provisions of Section XI of the ASME Code. (Reference Question 212.137)

32. Provide additional justification for exempting piping less than 3 inch in diameter from the augmented inservice inspection program for containment penetration piping. (Reference Question 111.81)

33. Additional information is required concerning the bases for the allowable vibration amplitude derived and clarification of the use of twice this allowable is acceptable. (Reference Questions 111.19 and 111.64)

34. Components which have been qualified by testing or analysis to other than current standards (IEEE Std 344-1975 and Regulatory Guides 1.92 and 1.100) will be required to be reevaluated and possibly requalified. This will be depended on the findings of the seismic qualification review team. (Reference Questions 111.40 and 111.74)

35. You have reference General Electric Topical Report NEDE 24057 as your basis of the reactor internal vibration test programs. We find that we need further information and clarification to complete our review. (Reference Questions 111.20, 111.22, 111.23 and 111.73)

36. We are sudying the problem of utilizing the square root of the sum of the squares for determining dynamic responses other than LOCA and SSE as you have used. By not utilizing the absolute sum method, the review may be entended if we do not agree that the square root of the sum of the squares methodology is applicable. (Reference Question 111.26)

37. You have not considered the combination of normal operating loads plus safe shutdown earthquake loads plus loss-of-coolant accident

loads plus loads due to safety/relief valve actuation. (Reference Question 111.75)

38. You have not completed your analysis and have not reported your evaluation in essential system piping against our criteria. (Reference Question 111.75)

39. We have not completed our review of GE Topical Report NEDE-4821 addressing reactor feedwater nozzle/sparger design modification for cracks nor have we completed GE's generic modification to the control rod drive return nozzle. This may require additional request for information. (Reference Question 111.35)

40. Additional information has been requested regarding your analytical and testing methods for your pump and valve operability assurance program. (Reference Questions 111.74, 111.76, 111.77, 111.78, 111.79, 111.80)

41. You have not provided the allowable limits for buckling for the reactor vessel support skirt subjected to faulted conditions. In addition, we requested information concerning the design of support bolts and bolted connections. (Reference Question 111.48 and 111.82)

42. You have not submitted your proposed program for the inservice testing of pumps and valves as required by 10 CFR 55.55a(g). (Reference Question 111.49)

43. We require the following additional LOCA analysis to complete your break spectrum:

    (a) The design basis accident with a discharge coffidient of 0.6, and

    (b) A small break analysis for a recirculation line break of 0.02 square feet. (Reference Question 212.125)

44. Confirm the operability of the post-accident leakage detection system following a seismic event or a loss-or-coolant accident.

45. Certain areas of your security plan need to be upgraded to comply with the requirements of Section 73.55 of 10 CFR Part 73.