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

FEB 05 1979

Docket Nos. 50-373
and 50-374

Mr. L. O. Del George
Nuclear Licensing Administrator
Boiling Water Reactors
Commonwealth Edison Company
Chicago, Illinois 60690

Dear Mr. Del George:

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

In our letters of October 16, 1978, December 21, 1978, and January 22, 1979, we advised you of those matters identified as outstanding issues in the SER as of those dates. As of the issuance of our January 22, 1979 letter we had not received SER inputs for Instrumentation and Control and Quality Assurance. We have now received draft copies of SER inputs for these two areas. The enclosure to this letter provides a listing of the outstanding matters identified in those two inputs. When the final SER inputs for Instrumentation and Control, Quality Assurance, and Reactor Systems are received there may be some adjustment in the final number of outstanding issues.

As indicated in our letter of January 22, 1979, we are referencing each issue to a question or questions which were provided to you, and items identified late in the review do not have referenced to previous requests for information.

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

Sincerely,

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

Enclosure:
As stated

cc w/enclosure:
See next page

520 258

7908080549

cc: Richard E. Powell, Esq.
Isham, Lincoln & Beale
One First National Plaza
2400
Chicago, Illinois 60670

Dean Hansell, Esq.
Assistant Attorney General
State of Illinois
188 West Randolph Street
Suite 2315
Chicago, Illinois 60601

520 259

ENCLOSURE

OPEN ITEMS

(1) The transient of generator load rejection without bypass results in the minimum critical power ratio of 1.02 which is below the safety limit of 1.06. You classified this event as an infrequent occurrence which would allow some fuel damage. We do not concur with this classification for this event, and we require that the operating limit be modified to satisfy the minimum critical power ratio limit of 1.06. (Reference Question 212.129)

(2) Your analysis for the failure of the feedwater controller indicates that the temperature drop is no greater than 100°F. At a domestic boiling water reactor, an actual feedwater temperature occurred which demonstrated a temperature difference of 150°F. Justify the decrease in temperature drop used for the loss of feedwater heater event or recalculate the transient by using a justified temperature decrease to assure conformance with applicable criteria. (Reference Question 212.142)

(3) Confirmation of pump coastdown characteristics must be verified during preoperational testing. Any deviations which would suggest a nonconservative analyses of transients or accidents must be explained.

(4) In analyzing anticipated operational transients, credit was taken for equipment which has not been shown to be reliable. You must identify this equipment in the technical specification with regard to availability, setpoints, and surveillance testing. Submit your plans for implementing this requirement along with any system modifications that may be required to fulfill the requirement. (Reference Question 212.144)

(5) Analyze the consequences of a turbine trip generator load rejection resulting from an SSE event without credit for nonseismically qualified equipment including reactor scram or recirculation pump trip or in the case of load rejection include the effect of turbine overspeed on recirculation flow. (Reference Question 212.129)

(6) Reanalyze the shaft seizure accident without allowance for the use of non-safety grade equipment. (Reference Question 212.135)

(7) You will be required to provide plant modifications in conformance with anticipated transients without scram criteria and in a shedular implementation procedure that may be provided in a rule or as adopted by the Commission.

(8) Insufficient information has been provided on your seismic qualification testing program for seismic Category I instrumentation and control equipment. (Reference Questions 031.52; 031.83 (1, 2, 3); 031.84; 031.148; 031.154 (1 & 2); 031.236; 031.244)

520 260

(9) You have not described adequately your environmental qualification program for your safety-related mechanical and electrical equipment. (Reference Questions 031.5; 031.40; 031.65; 031.83 (46); 031.102 (1, 2, &3); 031.127; 031.128; 031.250; 031.258)

(10) In your instrumentation design, Class IE instrumentation do not adhere to adequate separation criteria, have not been qualified, and do not adhere to separation of Class IE to non-class IE instrumentation. (Reference Questions 031.93; 031.108; 031.128, 031.136; 031.137; 031.146; 031.228; 031.240; 031.256; 031.223; 031.224; 031.245)

(11) In order to perform routine surveillance testing, it is necessary to pull fuses. We consider such a design does not satisfy the requirements of IEEE Std 279-1971 Paragraphs 4.11 and 4.20 (031.149 and 031.237)

(12) We are concerned that the worst-case combination of setpoint and accuracy may exceed sensor ranges. (Reference Questions 031.18; 031.159; 031.167; 031.185; 031.212; 031.234)

(13) We consider the rod block monitor as a protective system. (Reference Questions 031.67; 031.117; 031.150; 031.157;)

(14) We identified a single failure to the MSIV leakage control system which could lead to possible failure of the system during testing or operation. (Reference Question 031.152)

(15) For the standby liquid control system, Figure 807E161TD sheet 3 does not show that the two heater supplies are diverse. (Reference Questions 031.110; 031.214 (1); 031.261)

(16) The oneline drawings and schematics contradict the functional control drawings and system discription which are provided in the FSAR. Furthermore, contact utilization charts contradict the actual schematics. (Reference Questions 031.75; 031.164; 031.259, 031. 261; 031.262)

(17) The design of the safe shutdown indication does not satisfy the requipment of IEEE Std 279-1971, Paragraph 4.20. (Reference Questions 031.49; 031.51; 031.113; 031.134; 031.222; 031.251)

(18) The feedwater high level trip is a non-class IE control system which is required to provide adequate margin during certain transients. Sufficient information on this system has not been provided. (Reference Questions 031.085; 031.097; 031.124; 031.125;031.215; 031.255; 031.256; 031.257;)

(19) Your response to our positions for the quality assurance program for fire protection is incomplete. Your description does not indicate whether fire protection for quality assurance is under the management control of the quality assurance organization and what this control consists. In addition, sufficient detailed description has not been provided to specifically satisfy the quality assurance criteria

520 261

contained in ASB 9.5-1 in order for us to perform a detailed evaluation. (Reference Questions 421.1; 421.2)

(20) The revision number of the Commonwealth Edison Company topical report on quality assurance intended to apply to La Salle is not clearly provided, see Revision 6 of the topical report. (Reference Question 421.3)

(21) You have not provided specific commitments or acceptable alternatives to Regulatory Guides and ANSI Standard shown in the enclosed Table 1 for the operation phase. (Reference Question 421.3; 421.4)

(22) Adequate justification to include the dryers, steam separators and certain reactor components (feedwater sparger, jet pump instrumentation) from being under the control of the La Salle quality assurance program for maintenance and modification during the operation phase has not been described. The exclusion of these items is contrary to the guidance provided in Regulatory Guide 1.29 (Revision 3). (Reference Question 421.5)

(23) Any exceptions taken by the La Salle quality assurance program from the Commonwealth Edison Company Topical Report on quality assurance should be enumerated in Section 17 of the FSAR. (Reference Question 421.3)

(24) Insufficient information has been submitted for us to assess that the methods used to provide stress intensity values, are equivalent to those obtained from Appendix G of ASME Code. Clarification and justification of the methods used to construct the operating pressure temperature limits sh...d be provided. (Reference 121.1)

(25) The revised gradation for the riprap submitted in Amendment 41 to the FSAR is acceptable. However, the as-built spillway riprap does not conform to either the stated design in the FSAR or to good riprap erosion protection practices. Your are taking remedial action. When this action is completed, we will inspect the as-built riprap. (Reference Questions 371.22; 371.23)

(26) La Salle Station reactor vessels do not meet the specific requirements of Appendix G of 10 CFR Part 50. Identify and justify your exemptions. (Reference letter dated January 27, 1977)

(27) La Salle Station surveillance program does not comply with Appendix H of 10 CFR Part 50. Identify and justify your exemptions. (Reference letter dated January 29, 1979)

(28) Document your reevaluation of the reactor vessel, its internals, supports, and attached piping for combined loss-of-coolant accident and safe shutdown earthquake loads, including the annulus pressurization effects as presented at [illegible]ting of December 12, 1978. (Reference letter dated September [illegible])

520 262

TABLE 1

REGULATORY GUIDANCE FOR QUALITY ASSURANCE

1. Regulatory Guide 1.8-Rev. 1-R, "Personnel Selection and Training," (9/75).
2. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," (6/7/72).
3. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment," (8/11/72).
4. Regulatory Guide 1.33-Rev. 2, "Quality Assurance Program Requirements (Operation)," (2/78).
5. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," (3/16/73).
6. Regulatory Guide 1.38-Rev. 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," (5/77).
7. Regulatory Guide 1.39-Rev. 2, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants," (9/77).
8. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel," (8/73).
9. Regulatory Guide 1.64-Rev. 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants," (6/76).
10. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions," (2/74).
11. Regulatory Guide 1.88-Rev. 2, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," (10/76).
12. Regulatory Guide 1.94-Rev. 1, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," (4/76).
13. Regulatory Guide 1.116, Rev. 0-R, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems," (6/76).
14. Regulatory Guide 1.123, Rev. 1, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," (10/76).
15. ANSI Standard N45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," Draft 3, Rev. 4, (2/74).

(29) Document your reevaluation of the safety-related systems and components based upon the load combinations, response combination methodology, and acceptance criteria required by us as presented at our meeting of December 12, 1978. (Reference letter dated September 18, 1978)

(30) Clarify your consideration of the cyclic loadings due to the operating basis earthquake and safety/relief valve actuation in your fatigue analyses.

(31) Document your test program for all non-class 1, 2 and 3 high energy piping systems outside containment and all seismic Category I portions of moderate energy piping systems outside containment. (Reference Question 111.18)

520 264