NRCPDR

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

NOV 13 1978

Docket Nos. 50-373
and 50-374

Mr. Byron Lee, Jr.
Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Lee:

SUBJECT: SECOND ROUND REQUESTS FOR ADDITIONAL INFORMATION CONCERNING LA SALLE COUNTY STATION, UNITS 1 & 2

As a result of our continuing review of the La Salle Final Safety Analysis Report, we find that we need additional information to continue our evaluation. The specific information required is listed in the Enclosure.

Please inform us after receipt of this letter of the date you can supply the requested information so that we may factor that date into our review schedule.

Please contact us if you desire any discussions or clarification of the information requested.

Sincerely,

Olan D. Parr

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

Enclosure:
As Stated

cc w/enclosure:
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ENCLOSURE

210.0 REACTOR SYSTEMS BRANCH

212.125 We do not concur that the Zimmer LOCA analysis is an appropriate break spectrum analysis for La Salle because of (1) higher power level in La Salle, (2) different fuel assembly design in La Salle, and (3) higher peak clad temperatures predicted for La Salle.

We require that the applicant provide the following LOCA analyses to complete the break spectrum:

(1) One additional recirculation line break with a C_D coefficient 0.6 times the DBA, using the large break model analysis.

(2) One additional recirculation line break (0.02 ft^2) using the small break model analysis.

212.126 Demonstrate that your proposed system for LPCI flow diversion to wetwell spray cooling in the event of a LOCA will not adversely effect core cooling. Discuss the effects on core cooling and provide the necessary information to show that the requirements of GDC 35 of Appendix A to 10 CFR 50 and para. 50.46 of 10 CFR 50 are not violated for the following:

(1) When presenting the LOCA analysis for a limiting small break with LPCI diversion, you must provide assurance that this analysis bounds all possible LPCI diversion scenarios considering wetwell spray, dry well spray and suppression cool cooling, and assuming a single failure.

(2) Assure us that the system provided for diversion of LPCI flow meets single failure criteria so that diversion before 10 minutes need not be considered.

(3) Provide the results which includes the worst break, worst single failure, and diversionary time which provides the most conservative results.

Justify your conclusions regarding the worst break and worst single failure. This discussion should include the affects upon important parameters in the LOCA calculations, e.g. reflood, core heat transfer. Show that at times, other than that your selected for conservatively calculating LPCI flow diversion, the effect on core cooling is reduced.

(4) Provide a sensitivity study showing peak clad temperature as a function of break size for small break LOCA's assuming diversion will be initiated at 10 minutes. Perform this study for HPCS and recirculation line breaks. For the most limiting break, provide the following figures:

(a) Water level inside the shroud as a function of time during the LOCA

(b) Reactor vessel pressure vs. time

(c) Convective heat transfer coefficient vs. time

(d) Peak clad temperature vs. time

(e) ECCS flow rate vs. time

Should any time other than 10 minutes show worse results in item (3), above, provide the sensitivity study assuming diversion at this time instead of at 10 minutes.

(5) Provide a discussion which balances the need for LPCI diversion for the break size producing the highest peak clad temperature with the need for abundant core cooling (GDC 35). For example, this discussion could relate to Figure 6.2-14 with regard to the likelihood of LPCI diversion for this size break.

212.127 Show how the ECCS suction lines in the suppression pool are designed to prevent the formation of vortices and air ingestion when the ECCS is in operation.

212.128 Significant dimensional non-conformities were reported for La Salle County Station RHR, HPCS and LPCS pumps which were dismantled to ascertain damage as a result of flooding. In general, we are concerned with proper operation of all ECCS pumps, and these dimensional deficiencies add to our concern.

Provide assurance that ECCS pumps will operate as required including the RHR pumps in the long-term post-LOCA cooling mode. For your justification, we require operating histories of identical pumps in other facilities in both testing and/or operating modes. Note any differences in conditions under which pumps in other plants operated from the conditions expected for La Salle.

212.129 (5.4.6) (15.0) Your response to Question 212.115 is incorrect regarding generator load rejection and turbine trip transients resulting from an SSE since the events reported in Sections 15.2.2 and 15.2.3 did utilize the relief function of the S/R valves. Therefore, we require reanalyses of generator load rejection and turbine trip events resulting from an SSE. Components and equipment not qualified for seismic Category I and/or qualified components in structures not qualified for the SSE should not be used in the analysis, e.g., bypass valves, S/R valves in the relief mode, condensate storage tank (CST), reactor pump trip signals within the turbine building. The analysis should include the most limiting single failure in ESF equipment and justification for this worst single failure. Provide the initial MCPR, ΔCPR, lowest MCPR attained, and percent of rods in boiling transition for the most limiting event. All rods where MCPR is less than the safety limit should be considered to be in boiling transition.

212.130 (5.4.6) Show how the design of the RCIC protection system prevents unintentional shutdown of the system, when the system is required, because of spurious ambient temperature signals from areas in and around the system (especially in the RCIC pump room).

212.131 (6.3) Recent event reports from operating BWRs have shown that multiple relief valve failures may occur from a common mode failure. Provide assurance that your relief valve design is qualified (including testing after being subjected to an environment representative of an extended time period at normal operating conditions) to support your assumption that 6 of the 7 ADS valves will operate. A history of S/R valve operation, including similar valves in other plants, should be included in this evaluation. Both satisfactory and unsatisfactory operation should be included (the number of times the valve opened or failed to open, and the number of times the valve closed or failed to close).

Unless your response clearly demonstrates the unlikely probability of several relief valves failing from a common mode failure, we will require a sensitivity analysis which demonstrates the effect of failure of several relief valves on the results of the LOCA analyses.

212.132 (G.3) Show that the air supply to operate the ADS valves is sufficient in the event of:

(1) Small LOCA's,

(2) ADS valve system used as an alternate cooling method, because of valve failure in the RHR suction line, and

(3) Seismic event with no credit for non-seismic components and structures.

The evaluation should cover the total period during which ADS valve system operates or may be required to operate. Identify the core cooling mode which is used following completion of ADS operation.

212.133 (5.4.7) Your response to Question 212.92 is incomplete. You show the times at which cooling is reestablished but no mention is made of the time within which operator action is necessary in order to maintain adequate cooling. Provide the time the operator has to initiate and complete necessary actions in each case.

212.134 (6.3) We are concerned that pressure switches installed to ensure full HPCS, LPCS and LPCI lines (as noted in your response to Question 212.93) may be insufficient to detect voids at the top of piping in these systems. Initiation of ECCS with a partially voided line could severely damage the system. Provide a design modification to assure completely filled lines at all times. Identify the vertical locations of the presently installed pressure switches and highest point in each system.

212.135 (G.6.3) Standard Review Plan, Section 15.3.3, covers the event, recirculation pump shaft seizure. In paragraph III b, it is stated that "... safety functions should be accomplished assuming the worst single failure...." Your analysis does not indicate which single failure was taken. Table Q212.105-1 also shows that nonsafety-grade functions were assumed to actuate in the event (Level 8 turbine trip, turbine bypass, relief valves). Only safety-grade functions are allowed to mitigate the event. Reanalyze this event, taking credit only for safety-grade functions and assuming a single failure in ESF equipment. Justify the single failure you select as the most conservative.

212.136 (5.2.2, 5A) We require a reanalyses of the transient used to demonstrate suitable overpressure protection, MSIV Closure, utilizing the high pressure reactor pump trip.

212.137 (5.2.5) We consider that the check valves at the RCS/LPCS and RCS/LPCI boundaries perform an isolation function, which aide in the protection of the low-pressure systems. We require that these valves be classified category A/C in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and that they be tested periodically. Provide a discussion of the leak tests and the acceptance criteria for these tests.

212.138 (5.2.5) We require a system to detect intersystem leakage into the LPCS and LPCI systems from the RCS. Show how you detect inleakage into both the LPCS and LPCI systems (all three trains) including sensitivities necessary to meet Technical Specification requirements.

212.139 (3.5.1.2) Your response to Questions 212.75 is incomplete. Provide a list of potential missiles inside containment. Include a list of possible targets consisting of safety-related structures, systems and components inside containment, including, the containment structure itself, and show how each target is protected.

212.140 (4.6) We require that you provide a plan for periodic testing of the scram accumulators in the CRD hydraulic system to assure that there is a least a 20 minute period available, following the loss of both drive water pumps in the control rod drive hydraulic system, before scram becomes marginal, as discussed in your response to Question 212.6.

212.141 (5.4.6) (15.2.2) 15.2.3) Since an SSE requires the use of the RCIC (as noted in FSAR Sections 15.2.2 and 15.2.3) in the event of single failure of the HPCS, show either that the RCIC can operate for at least 10 minutes without any water source in the event of seismic failure of the condensate storage tank (CST) or show that the SSE will not cause failure of the CST or provide a seismically qualified automatic switchover of the RCIC suction from the CST to the suppression pool. If neither an alternate source of water nor an automatic switchover is available, provide procedures for the operator to switch the RCIC suction to the suppression pool no earlier than 10 minutes in the event of an SSE, with acceptable consequences.

212.142 (15.1.1) The calculated consequences for the loss of feedwater heating (LFWH) event are currently based on the maximum feedwater (FW) temperature reduction which could be associated with the loss of a single feedwater heater string. For La Salle, the loss of a single FW heater string results in a temperature drop no

greater than 100°F, which is used in the licensing basis LFWH analysis. As indicated in the plant's FSAR, the 100°F assumption stems from the maximum loss of feedwater heating capability which could result from the worst malfunction of a single valve within the FW heating system, i.e., spurious opening or spurious closure of one valve. Thus, the severity of the LFWH event is based on a single mechanical component malfunction.

However, an actual FW temperature transient which occurred at a domestic BWR demonstrated that a single electrical component failure, e.g., circuit breaker-trip (of a motor control center) could precipitate an even more severe FW temperature transient. The subject electrical equipment malfunction, which temporarily caused a complete loss of all FW heating due to total loss of extraction steam, resulted in a FW temperature drop of about 150°F. Accordingly, the staff requests that La Salle either (1) submit a detailed failure modes and effects analysis to demonstrate the adequacy of a 100°F feedwater temperature reduction relative to single electrical malfunctions or (2) submit calculations using a limiting FW temperature drop which clearly bounds current operating experience.

212.143 (6.3) (G.3.1.2) You have analyzed the effect on LOCA for instantaneous closure of the flow control valve (FCV) in the unbroken loops, in FSAR Appendix G, Section 3 1.2. This overly conservative result indicated an increase in peak clad temperature (PCT) of 300°F which, if added to the LOCA PCT, would be in excess of the maximum PCT criterion of 10 CFR 50.46. Provide an analysis showing the effect of a realistic maximum FCV closing rate upon the DBA, indicate which single failure of ESF was taken, and discuss the details of the analysis and results.

212.144 In your response to Question 212.105, you have provided a list of non-safety grade systems or components for which credit was taken in the analysis of transients. We require that before credit can be taken for this equipment, justification of its reliability must be provided in conformance with GDC 29. Alternatively, analyses may be provided which demonstrate that no safety limits would be violated if this equipment is not considered in calculating transient performance.

All transients tabulated in response to Question 212.105 must be considered in this evaluation, including "Turbine Trip Without Bypass" and "Generator Load Rejection Without Bypass."

Report the results of your evaluation and reanalysis including operating MCPR, ΔCPR, minimum MCPR and justification for the event producing the largest ΔCPR.

212.145 In the response to Question 121.8, you state that the control rod drive (CRD) hydraulic return line has been eliminated and that the CRD drive water has been rerouted to a low oxygen source.

Provide a P&ID of the CRD hydraulic system containing these modifications. Demonstrate that these modifications will not impair the operability of the CRD's. Provide a comparison of the reactor vessel makeup capability for both single and multiple CRD pump operation before and after the proposed modification.