Consumers Power Company

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December 15, 1981

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Director, Nuclear Reactor Regulation
Att Mr Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - SEP TOPIC III-6,
SEISMIC DESIGN

During the meeting between the NRC, Stevenson and Assoc, Lawrence Livermore Laboratories and Consumers Power Company held in Jackson on November 6, 1981, Consumers Power Company was requested to provide additional information concerning several open SEP seismic design issues for Palisades. This letter provides a partial reponse to that request.

Attachment 1 provides responses to several questions regarding NSSS component design. It is anticipated that responses to the two remaining questions regarding CRDMs and reactor internals will be ready for submission within one week.

Attachment 2 provides a type test report for the new battery racks installed at Palisades during the current outage. A copy of this report was given to the NRC during the meeting.

Attachment 3 provides the seismic evaluation of the plant service water pumps. A copy of this report was also provided to the NRC during the meeting.

The status of the remaining open issues is as follows:

1. DFO Day Tank - A copy of the seismic evaluation report was provided to the NRC at the meeting. The report is in final form, but supplementary information is being developed to discuss modifications which are being made to the tanks.

2. Safety Injection Tank - A copy of the draft seismic evaluation report was provided to the NRC at the meeting. This report has been revised and is currently being reviewed.

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3. Eccentric Small Pipe Loads - Information is being prepared to show how this subject has been addressed.

4. Reactor Coolant Pump Support Structure - Information is being prepared to address Dr Stevenson's question concerning buckling of the support columns. As we discussed in the meeting, however, we do not anticipate any problems in this area because the columns are very short.

5. Electrical Panels - Information is being prepared to discuss how this general question was addressed during the recent anchorage modifications. We do not anticipate that new calculations will necessarily be required.

It is anticipated that discussions of Issues 1 through 4 can be submitted within approximately one week. It must be noted, however, that if new calculations are necessary for Item 4, additional time will be required. Our target date for a submittal on Issue 5 is December 15, but it is still not certain that this date is realistic. As with Item 4, if we determine that other calculations are needed, additional time will be required.

Robert A Vincent

Robert A Vincent
Staff Licensing Engineer

CC Director, Region III, USNRC
NRC Resident Inspector - Palisades
John D Stevenson and Assoc

RESPONSES TO QUESTIONS CONCERNING SEISMIC DESIGN OF NSSS COMPONENTS

1. Reactor Coolant Pumps

Question a - In Table 1 of Appendix C of Attachment 2 of letter P-CE-5732, for the pump discharge nozzle, should P_M be used instead of P_L, and should the code allowable stress be 2.4 S_m instead of 1.5 S_m?

Answer - P_L is the primary local membrane stress intensity. P_M is the primary general membrane stress intensity. The evaluation of P_L and the use of 1.5 S_m is consistent with the original RCP analysis as shown on Page C-13 of Appendix C.

Question b - In Table IV of Appendix C, are the code allowable stresses of 1.5 S_m for the primary membrane stress and 2.25 S_m for the primary membrane plus bending stress correct?

Answer - For primary membrane stress, the allowable stress evaluation is per Appendix F, Par F-1323.1 of Section III, Division 1 of the code. The allowable stresses are"... the greater of 150% of the tabulated S_m value of 120% of the tabulated yield strength, but not to exceed 0.70 S_u, with both values taken at the appropriate temperature."

For the primary membrane plus bending stress, the allowable stress evaluation is per Subsection NB, Par NB-3221.3 of the code. This paragraph says that the faulted membrane plus bending stress allowed is 1.5 times the faulted membrane allowables or 1.5 x 1.5 S_m = 2.25 S_m.

Question C - Should the RCP be analyzed seismically for its impeller, shaft or flywheel?

Answer - It is not a requirement that the RCP remain functional during a seismic event. Items not related to the structural integrity of the pressure boundary and supports are not analyzed seismically.

2. Question - Regarding the steam generators, a statement was requested indicating the seismic adequacy of steam generator tubes.

Response - The final stress report (see Reference (2), Page C-294) considered seismic loads as specified in Reference (1), but elected to neglect their effect in favor of the more controlling gravity and flow loads (three times normal flow during a steam line break accident).

Lateral seismic loading on tubes was evaluated in Reference (3) (see Page 7) using twice the design loads specified in Reference (1). In addition, a dynamic analysis was performed for "LOCA shaking" loads on the tubes, which was similar in nature to lateral seismic loading and produced twice the stress on the tubes (see Page A.19).

Vertical seismic loading on tubes was considered in Reference (4) (see Page 6) using twice the design loads specified in Reference (1), but were conservatively neglected in combination with gravity loads.

References -
(1) Engineering Specification 70P-002, Revision 2, for the Palisades Steam Generator, February 1979.

(2) CENC-1120, Analytical Report for Consumers Power Steam Generator, May 1969.

(3) CENC-1264, Revision 2, Analysis To Determine Allowable Tube Wall Degradation for Palisades Steam Generators, March 1976.

(4) CENC-1288, Main Steam Line Break Analysis of Palisades Steam Generator Internals (Including Tube Sleeves) June 1977.

3. Question - Regarding CRDMs, an expanded explanation of Page 14 (Reference P-CE-5732) was requested; it was also requested that an explanation be provided which addresses the differences between this analysis and the original analysis, Reference #59, Page 133 of the Lawrence Livermore Report.

Response - The previous submittal is being amended to include the expanded explanation and comparison to the original study. This information will be submitted in approximately one week.

4. Question - Regarding reactor vessel internals, we were requested to review our files to see if we could provide an example of an original calculation, preferably of a component in compression; eg, a lower support column calculation, where the potential for buckling was considered.

Response - We have been unable to find such an example after conducting a limited review of our files. Instead, we will provide a new calculation of such an example using the "old" methodology. This information will be submitted in approximately one week.