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June 15, 1981

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Crutchfield:

Subject: SEP Topics III-6, Seismic Considerations
and III-11, Component Integrity
Oyster Creek Nuclear Generating Station
Docket No. 50-219-1

The attached is in response to your letter of March 20, 1981 which transmitted nineteen open items concerning the subject matters and requested additional information.

If you should have any questions concerning this information, please contact Mr. J. Knubel (201-299-2264) of my staff.

Very truly yours,

Philip R. Clark
Deputy - Chief
Operating Executive

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RESPONSE TO OPEN ISSUES
OYSTER CREEK SEISMIC REVIEW

1. NRC Open Issue

Emergency Service Water Pump - Functional integrity was not evaluated due to lack of design detail. A determination of the material used for the pump housing should be made. If the material is cast iron, a justification should be provided for its adequacy.

Response

An evaluation of the structural and functional adequacy of the emergency service water pumps was made by Burns and Roe as a result of the Senior Seismic Review Team (SSRT) site visit in 1979. This evaluation indicated that the stresses in the pump housing are acceptable and the calculated deflection of the pump volute is such that the pump manufacturer, Byron Jackson, does not consider that there will be any effect on pump function.

Our investigtion indicates that the material of the pump housing is wrought iron. Based on the allowable stress (2/3 yield strength) of this material and maximum loading acting at the worst location of the pump housing, indicates that the housing is adequate to sustain seismic loadings.

2. NRC Open Issue

Emergency Isolation Condenser - The audit analysis indicated that the anchor bolts at the center saddle were found to be overstressed in seismic shear from the postulated landing. Provide detailed analysis to demonstrated design adequacy of these anchor bolts.

Response

An evaluation of the Emergency Isolation Condenser anchor bolts for their design adequacy will be performed using more recent reactor building floor response spectra being developed. The results of the analysis will be forwarded to you by October, 1981.

3. NRC Open Issue

Containment Spray Heat Exchanger - The anchor bolts were found to be overstressed from the postulated seismic loads. Provide detailed analysis to demonstrate design adequacy of these anchor bolts.

Response

An analysis will be performed to demonstrate design adequacy of these anchor bolts. The results of the analysis will be transmitted to you in July, 1981.

4. NRC Open Issue

Recirculation Pump Support - No information was provided for evaluation.

Response

Recirculation piping seismic and flexibility analyses were transmitted as References 9 and 10 to JCP&L letter, dated July 9, 1979 and recirculation pump drawings were provided as Enclosure 10 to JCP&L letter, dated August 29, 1979.

In addition, analyses of the recirculation piping, including pump supports (with and without a pump motor removed), were made by our consultant in support of the plant modification in which a recirculation pump was removed and a cover plate installed. The analysis include the support load information requested by the NRC. The information will be summarized and forwarded to you by July, 1981.

5. NRC Open Issue

Motor Operated Valves - The seismic accelerations used in your evaluation (refs. 77 and 78 of Enclosure 1) are unrealistically low. Detailed reevaluation with proper seismic accelerations should be provided to demonstrate its design adequacy. No information was provided to evaluate functional adequacy.

Response

We are currently developing reactor building floor response spectra for Oyster Creek Nuclear Generating Station using more current criteria. Results of this study will be forwarded to you by September, 1981.

6. NRC Open Issue

CRD Hydraulic Control Units - Support system of the free standing (cantilever) type was found to be overstressed from the postulated seismic loads. Provide detailed reanalysis to demonstrate design adequacy.

Response

This item was the subject of a conference call among Mr. Schmidt (MPR), Messrs. Cheng, Hermann and Paulson (NRC), Dr. Stevenson (NRC consultant) and Nagai (JCP&L) on March 10, 1981. During this conversation we reported that (1) the SSRT draft report utilized an input seismic load which is approximately a factor of 2 too high based on a more recent site specific spectra recommended for Oyster Creek by the NRC, (2) the more realistic NRC seismic input would result in an acceleration of about 0.16g at the CRD hydraulic control units; (3) the as-built unit anchorages were verified during the 1980 Refueling Outage and calculations show the unit is adequately supported for acceleratios of up to at least .7g loads; and (4) seismic qualification report provided by GE for similar CRD hydraulic control units indicate that they are qualifited for at least 0.64g. On this basis, it was concluded that the CRD hydraulic control units at Oyster Creek will not be overloaded. Dr. Stevenson concurred in this conclusion. The basis for this evaluation was transmitted to you by our letter dated May 7, 1981.

7. NRC Open Issue

Reactor Vessel, Support and Internal - No detailed information was available to evaluate design adequacy.

Response

Available design drawings and calculations on the reactor vessel supports were transmitted to the NRC by JCP&L letters, dated August 24, 1979 and December 21, 1979. The SSRT draft report states that the vessel acceleration is calculated to be 0.63g for 7% damping at the fundamental vessel frequency of 7.75 Hz. However, it should be noted that the SSRT draft report utilized an imput seismic load which is approximately a factor of 2 too high based on a more recent site specific spectra recommended for Oyster Creek by the NRC and the more realistic NRC seismic input would result in an acceleration of about 0.40 g.

Further analyses on reactor vessel and support will be performed utilizing the reactor building floor response spectra being developed. The results will be forwarded to you by October, 1981.

The information concerning the reactor internals will be transmitted to you by October, 1981.

Items 8, 9, 10, 12 and 13 - NRC Open Issues

8. Motor Control Centers - No information was available to evaluate either structural integrity or functionability of these components.

9. Transformers - No information was available to evaluate design adequacy of these components.

10. Switchgear Panels - Provide information to show that the panels are positively anchored to resist seismic overturning moments and sliding forces.

12. Control Room Panels - No information was available for evaluation.

13. Battery Room Distribution Panels - No information was available for evaluation.

Response

The structural adequacy of the anchorage of these components was evaluated in 1979 and during the 1980 refueling outage. As indicated in JCP&L letters, dated February 22, 1980 and May 7, 1981 to the Commission, these components were evaluated; any which were determined to be deficient were modified during the 1980 refueling outage; and all such modifications are complete. With regard to the functionality of this electrical equipment, this question is the subject of a generic SEP Owners Group electrical equipment qualification program which is being conducted by the SEP Owners and Westinghouse. The status of this generic qualification program was reported to the NRC in a meeting on March 5, 1981.

11. NRC Open Issue

Emergency Diesel Generator - No information was available to evaluate design adequacy of either the anchorage system or the functionality of the diesel generator.

Response

Information on the adequacy of the emergency diesel generator anchorage was provided to the NRC by JCP&L letter, dated December 21, 1979. The information indicates that sufficient friction exists between the base of the generator skid and the foundation to preclude significant sliding or tipping of the diesel generator during a seismic event.

It should be noted that unlike the open issue given in Enclosure 2 to the NRC letter, the draft SSRT report indicates that based on their analysis the emergency diesel generator will remain stable under SSE seismic loading. However, the report recommends that the licensee provide positive anchorage for the diesel generator to resist starting torque and vibration affects. The results of our communication with the manufacturer of the Oyster Creek diesel generator indicates that the starting torque and vibration effects are absorbed internally by the base support structure. Therefore, anchoring of the diesel generator to the floor is not necessary.

14. NRC Open Issue

Isolation Phase Ductwork Supports - The duct support was found to be overstressed from the postulated seismic forces.

Response

The above conclusion was reached by JCP&L as a result of Burns and Roe calculations performed in support of the SSRT site visit. As a result, action was taken to add duct supports, and this modification was completed during the 1980 outage. This was reported to the NRC in JCP&L letter, dated February 22, 1980.

15. NRC Open Issue

Condensate Storage Tank - The anchor bolts were found to be overstressed from the postulated seismic loading conditions. A similar conclusion was drawn from the results of analyses reported in your FDSAR (ref. 40 of Enclosure 1), but no corrective action was taken. Provide justification to show the design adequacy of the tank and the outlet piping.

Response

Anslyses were performed by J. A. Blume during plant construction which indicated that the hold down bolts for the condensate storage tank would be overstressed during a seismic event. However, Burns and Roe's evaluation of the Blume recommendation indicated that, if friction between the tank and its foundation is considered, the hold down bolts are adequate, and no corrective action is required. Further evaluation will be performed and results of the evaluation will be forwarded to you by November, 1981.

16. NRC Open Issue

Torus - Insufficient information was provided to evaluate the design

adequacy of the supporting columns and its connections to the lateral bracings.

Response

The adequacy of the torus for operational dynamic and seismic loads is being evaluated as part of the Mark I Containment generic program. The results will be forwarded to you upon completion of the evaluation.

17. NRC Open Issue

Reactor Building - Provide detailed analyses to demonstrate that the cables are slack enough to accommodate differential displacements between the buildings.

Response

Site inspections were made in 1979 by JCP&L to determine the gap available between the reactor and turbine buildings to accommodate displacements of the building during a seismic event. The results of these inspections confirmed that a few inches of space are available to accommodate predicated differential building movement during an earthquake. A similar site inspection will be performed to confirm by engineering judgement that adequate slack and cable tray flexibility exists within the vital cable runs to accommodate the few inches of relataive motion predicted between the reactor and turbine buildings. The results of the investigation will be transmitted to you by July, 1981.

18. NRC Open Issue

Piping - Provide a detailed reanalysis to demonstrate design adequacy of the following piping systems:

A. Main Steam Line - Several snubbers were found to be overstressed by the potential seismic loads.

B. The results of the audit analysis showed loading conditions at several locations to exceed the ASME Code allowable limits.

Response

Results of main steam and CRD return line seismic analyses conducted by EG&G, Idaho (an NRC consultant) were discussed in a telephone conversation on March 11 among EG&G (Mr. M. Nitzel), JCP&L (Mr. R. Ashby), the NRC (Mr. R. Hermann and Mr. T. Cheng), and MPR (Mr. W. Schmidt). A report of this telephone conversation was previously transmitted to JCP&L and is attached. Based on the information provided by EG&G, the following was concluded:

A. The snubbers in the main steam line are loaded to a maximum of about 18,000 lbs. during a safe shutdown earthquake. This value is in excess of the rated capacity of the snubbers of approximately 11,000 lbs. However, this rated capacity has a factor of safety of at least 4 on actual failure capacity. Since the loads due to an SSE are not required to be within normal rated capacities, and the predicted loads are a factor of at least 2 below the ultimate capacity of the snubbers, this loading condition is considered acceptable.

B. It is believed that the analysis referred to in A above, refer to the CRD return line analyses. In the telephone conversation, referenced above, it was determined that higher than allowable stresses were calculated for the CRD pump discharge. This area will require further evaluation, and the results of our evaluation will be transmitted to you in November, 1981.

19. NRC Open Issue

Electrical Cable Raceways - No information was available for evaluation.

Response

This open issue is the subject of a generic electrical cable and conduit raceway qualification program being conducted by URS/Blume for the SEP Owners and will be addressed as a part of that program. A status report on this program was presented to the NRC by URS/Blume and the SEP Owners on March 5, 1981.