NRC PDR

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

July 9, 1979

Docket No.: 50-70

Mr. R. W. Darmitzel, Manager
Irradiation Processing Product
Section
General Electric Company
Vallecitos Nuclear Center
P. O. Box 460
Pleasanton, California 94566

Dear Mr. Darmitzel:

Based on our review of the seismic design information submitted in response to the Order to Show Cause dated October 24, 1977, we have determined that the additional information identified in the enclosure is necessary to complete our review. Please provide your response by July 31, 1979.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
Request for Additional
Information

cc w/enclosure: See next page

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General Electric Company

cc w/enclosure(s):
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U. S. Nuclear Regulatory Commission
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ENCLOSURE 2

GENERAL ELECTRIC TEST REACTOR (GETR)
DOCKET NO 50-70
REVIEW OF REPORTS SUBMITTED IN RESPONSE
TO NRC ORDER TO SHOW CAUSE DATED 10/24/77

REQUEST FOR ADDITIONAL INFORMATION

ENGINEERING BRANCH
DIVISION OF OPERATING REACTORS

1. Discuss how extensive the cracking of the floor slabs is due to the overturning moments. Verify that there is no impact on safety related components due to spalling or cracking.

2. Verify, with a detailed discussion addressing generated missiles, pipe and support deformation capabilities, structural stiffness and strength degradation, and reactor building leaktightness integrity; that the extensive cracking and failure resulting from surface rupture will not impact safety related components or systems.

3. Verify that the extensive failure resulting from surface rupture will not compromise the integrity of the interior radial wall, the circumferential wall connection or the ability of the containment to support the required loadings without impacting the integrity of any safety related components, systems or equipment. Discuss the extent of the predicted containment damage in detail to substantiate your statement that its deformations are acceptable. Specifically, address the possibility of a punching mode of failure.

4. Justify the material properties used for the soil spring model, including damping values and poisson's ratio. To what level in the actual subgrade do these values correspond? Discuss the impact on the factors of safety provided for the forces and floor accelerations if the "most realistic case" of subgrade parameters (as described in your Phase 2 Seismic Analysis Report) is not present. That is, what safety considerations are provided in the event that the actual response is greater than predicted by Case 1 parameters?

5. Verify that the calculations of the sliding and overturning resistance have accounted for the reduction of the weight of the building due to vertical uplift. If uplift due to vertical excitation is not considered, justify the appropriateness of the unconservative analysis.

6. Verify that the maximum sliding displacement of 1.3 inches results in no failure of safety related piping, components, or equipment.

7. In your Seismic Analysis of Reactor Building - Phase 2 report, you state that "there is no structural continuity between the foundation mat and the rest of the reactor building." Describe how this is represented in the mathematical model. Provide the properties of the member between the foundation mat and the basement slab, and describe how they were determined. (Provide the terms of the local stiffness matrix). Also, verify that the results in Table 2-9 for sliding at the interior concrete - foundation slab interface reflect these properties and the bounding case considering response variations due to all potential variations considered in your analyses. If relative motion is predicted, discuss the impact on the results of your analyses.

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8. Describe the procedures utilized in the determination of the soil spring boundary conditions in your Model B nonlinear analysis. Also demonstrate that this type of representation of the subgrade is appropriate considering soil depth, layering, etc. Discuss the acceptability of your modeling as opposed to using the current finite element techniques.

9. Provide a description of the core structure displacements associated with the yielding and settlement of the foundation mat. Verify that these displacements were considered in the design of the core structure and safety related components, systems and equipment, and that the integrity of these safety related items is not compromised.

10. In the post-offset analyses, provide the acceptance criteria for the seismic displacements and forces. Also, provide the factors of safety against sliding and overturning for this condition and summarize how they were determined.

11. Describe in detail the methods by which the allowable shear and tensile stresses were determined from the referenced test data. Justify the correspondence between the GETR walls and these test samples since the PCA tests were flange reinforced specimens. Verify that the stresses calculated via your finite element representation of the GETR are directly comparable to your stated allowables. Provide the bases for your statements. Include a discussion of how construction joints were considered in your evaluation and the possibility of degradation of these joints due to water seepage weakening the shear transfer across the joint.

12. Verify that the effects of the primary piping which is anchored to the concrete structure, have been considered in the seismic analysis and design of the concrete structure.

13. Discuss the procedures used to determine the location of impact of the cask drop on the canal slab which produces the maximum moment on the slab. Provide this moment, and verify that the slab is capable of withstanding this load. Also, verify that spalling of concrete due to the cask drop on the canal slab does not impact any safety related items.

14. If a cask drop results in damage to the liner and cracking of the concrete, verify that adequate canal water is maintained.

15. Provide justification that a non-mechanistic lower head nozzle rupture occurs with sufficiently low probability to assure the acceptability of the consequences of this event. Provide similar justification for rupture of the reactor pool. Include a discussion, in terms of radiation levels and stress levels, verifying that no embrittlement occurs, such as to preclude postulating the above failures.

16. Specify the maximum inner and outer fuel storage tank displacements, and verify that these maximum displacements are obtained using the more realistic configuration where sliding is permitted. Verify that these displacements do not adversely impact the safety related functions of the tanks. Discuss the consequences of a 1.4 inch inner tank maximum rocking deflection. Also, verify that sliding of the tanks does not result in impact to the canal liner.

17. In the inner fuel storage tank rocking analyses, provide justification for the use of a factor of 67% to reflect the energy losses and fluid inertia effects.

18. In the inner fuel storage tank rocking analysis, describe how the 123.25 lb/in. live load on the outer tank was resolved into the concentrated loads applied at nodes 22, 23, 24, and 25.

19. Explain how the response spectra for three percent damping used in the seismic analysis of the primary cooling system and PRV envelop the response spectra obtained for one percent damping, by a factor of 1.2. (See EDAC-117-217.05, page 2-2)

20. Describe the piping displacements resulting from the analysis of Run 1 & Run 2. Provide the design and acceptance criteria for pipe displacements, and verify that the maximum displacements are within design allowables. Also verify that seismic excitation does not result in impact between piping systems and any safety related equipment or components.

21. Discuss how the effects of a surface rupture offset have been considered, and verify that they will not compromise the integrity, of the primary cooling system and reactor pressure vessel.

22. List the types of restraint anchorages used for the GETR piping and equipment, and describe the procedures used in the design of these anchorages. Verify that cyclic loads have been considered, and describe and justify the anchor bolt and rock bolt cyclic load design requirements. Describe any inservice inspections which are planned for the bolts and justify the extent of the program.

23. Verify that the piping restraints and anchors are in the correct locations, as designed.

24. Verify that thermal loads and fluid transients were considered in the analysis and testing of the valves.

25. In your Structural Analysis of Third Floor Missile Impact System, as discussed on page 17, discuss how and why the normal impact loads are applied to the bent in the Z-direction, instead of the y-direction which appears to be consistent with a resultant force Px. Also, discuss why the lateral loading was applied in the x-direction. Verify that a lateral load applied perpendicular to the x-direction is not a more critical case. Provide bent allowable stresses, including buckling stresses, if appropriate, to verify that design stresses are small. Explain the inconsistencies of Figures 5 and 6 notations.

26. Verify that maximum tensile force in the base plate bolts due to lateral bent loading with upward seismic motion (and no normal impact loading) have been considered and are within allowables.

27. Discuss the in-service surveillance programs which will be conducted on all safety related components.

28. Justify the acceptability of bolted base plates where the jam nut is placed inside of the main nut. Specifically, verify that the system will not fail at the jam nut when loaded due to vibratory motion, thus unlocking the main nut and allowing it to back off.