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Docket No. 50-289

JUN 9 1976

Metropolitan Edison Company
ATTN: Mr. R. C. Arnold
Vice President - Generation
P. O. Box 542
Reading, Pennsylvania 19603

Gentlemen:

RE: THREE MILE ISLAND UNIT NO. 1

On October 15, 1975, we informed you of a potential safety question which has been raised regarding the design of reactor pressure vessel support systems. We requested that you review the design bases for the reactor vessel support system for your facility to determine whether the transient loads described in the enclosure to our letter were appropriately taken into account in the design.

Your reply of November 21, 1975, indicated that the transient differential pressures in the annular region between the reactor vessel and the cavity shield wall and across the core barrel were not considered in the support design.

In our letter of October 15, 1975, we indicated that on the basis of your initial review, a reassessment of the vessel support design might be required. We have now determined that such a reassessment is required.

As you are probably aware, we have been discussing with the PWR vendors and various architect/engineer firms the generic aspects of this problem. Should you contemplate utilizing organizations other than your PWR vendor for calculation of the sub-cooled internal loads, we suggest you contact us for the benefit of a brief review of our generic discussions to date. We will continue these generic discussions with the vendors and architect/engineers, but such discussions are not intended to pace your evaluation of this concern nor to eliminate the possibility that we may have additional questions regarding your evaluation after submittal. While the emphasis given in this letter deals with the reactor vessel cavity, for your information and guidance our generic review may consider other areas in the nuclear steam supply system and further evaluation may be required.

Please inform us within 30 days after receipt of this letter of your schedule for providing us your evaluation of the adequacy of the pressure vessel supports when the sub-cooled loads are calculated and taken into account in a manner which you determine best represents these phenomena. Your evaluation should include the answers to the attached request for additional information.

This request for generic information was approved by GAO blanket clearance number B-180225 (R0072). This clearance expires July 31, 1977.

Sincerely,

Original Signed by

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

Enclosure:
Request for Additional
Information

cc w/encl:
G. F. Trowbridge, Esq.
Shaw, Pittman, Potts, & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

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GPU Service Corporation
Richard W. Heward, Project Manager
Thomas M. Crimmins, Jr., Safety
and Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

Pennsylvania Electric Company
Mr. R. W. Conrad
Vice President, Generation
1001 Broad Street
Johnstown, Pennsylvania 15907

cc w/encl.:
Mr. Weldon B. Arehart, Chairman
Board of Supervisors of Londonberry
Township
2148 Foxiana Road
Middletown, Pennsylvania 17057

Miss Mary V. Southard, Chairman
Citizens for a Safe Environment
P. O. Box 405
Harrisburg, Pennsylvania 17108

Government Publications Section
State Library of Pennsylvania
Box 1601 (Education Building)
Harrisburg, Pennsylvania 17126

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OFFICE ▸	ORB#4:DOR	C-ORB#4:DOR				
SURNAME ▸	CNelson:rm	RReid / RR				
DATE ▸	6/8/76	6/9/76				

REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that reactor pressure vessel supports may be subjected to previously underestimated lateral loads under the conditions that result from the postulation of design basis ruptures of the reactor coolant piping at the reactor vessel nozzles. It is therefore necessary to reassess the capability of the reactor coolant system supports to assure that the calculated motion of the reactor vessel under the most severe design basis pipe rupture condition will be within the bounds necessary to assure a high probability that the reactor can be brought safely to a cold shutdown condition.

The following information should be included in your reassessment of the reactor vessel supports and reactor cavity structure.

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principle elements and materials of construction.
2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximum stress in each principle element of the support system and the corresponding allowable stresses.
3. Provide the information requested in 2 above considering a postulated break at the design basis location that results in the most severe loading condition for the reactor pressure vessel supports. Include

a summary of the analytical methods employed and specifically state the effects of asymmetric pressure differentials across the core barrel in combination with all external loadings including asymmetric cavity pressurization calculated to result from the required postulate. This analysis should consider:

(a) limited displacement break areas where applicable

(b) consideration of fluid structure interaction

(c) use of actual time dependent forcing function

(d) reactor support stiffness.

4. If the results of the analyses required by 3 above indicates loads leading to inelastic action in the reactor supports or displacements exceeding previous design limits provide an evaluation of the following:

   (a) Inelastic behavior (including strain hardening) of the material used in the reactor support design and the effect on the load transmitted to the reactor coolant system and the backup structures to which the reactor coolant system supports are attached.

5. Address the adequacy of the reactor coolant system piping, control rod drives, steam generator and pump supports, structures surrounding the reactor coolant system, [core support structures, fuel assemblies, other reactor internals ....] and ECCS piping for both the elastic and/or inelastic analyses to assure that the reactor can be safely brought to cold shutdown. For each item include the method of

analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

The compartment multi-node pressure response analysis should include the following information:

6. The results of analyses of the differential pressures resulting from hot leg and cold leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity and pipe penetrations.
7. Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure within the reactor cavity. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variations circumferentially, axially and radially within the reactor cavity.
8. Provide a schematic drawing showing the nodalization of the reactor cavity. Provide a tabulation of the nodal net free volumes and interconnecting flow path areas.
9. Provide sufficiently detailed plan and section drawings for several views showing the arrangement of the reactor cavity structure, reactor vessel, piping, and other major obstructions, and vent areas, to permit verification of the reactor cavity nodalization and vent locations.

10. Provide and justify the break type and area used in each analysis.
11. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.
12. Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical justification for the removal of such items to obtain vent area. Provide justification that vent areas will not be partially or completely plugged by displaced objects.
13. Provide a table of blowdown mass flow rate and energy release rate as a function of time for the reactor cavity design basis accident.
14. Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressures.
15. Provide the peak calculated differential pressure and time of peak pressure for each node, and the design differential pressure(s) for the reactor cavity. Discuss whether the design differential pressure is uniformly applied to the reactor cavity or whether it is spatially varied.

In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during the subcooled portion of the blowdown analysis, the following information is requested:

16. A complete description of the hydraulic code(s) used including the

development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations.

17. In support of the hydraulic code(s) used provide comparisons with the code(s) to applicable experimental tests, including the following:

    (a). CSE tests B-63 and B-75

    (b). LOFT test L1-2

    (c). Semiscale tests S-02-6 and S-02-3

    The models developed should be based on the assumptions proposed for the analysis of a PWR.

18. Provide a detailed description of the model proposed for your plant and include a listing of the input data used and a time zero edit. Identify the assumptions used in developing the model, specifically the treatment of area, length and volume.

19. Typically the current generation of hydraulic subcooled blowdown analysis codes solve the one-dimensional conservation equations. However, they are used to model the multi-dimensional aspects of the reactor system (i.e. the downcomer annulus region). Provide justification for the use of the code(s) to model multi-dimensional regions, including the equivalent representation of the region as modelled by the code(s).