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Docket Nos: 50-438
and 50-439

Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

On November 14, 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 Bellefonte 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 December 22, 1975, states that you have reviewed the design bases for the reactor vessel support system for the Bellefonte Nuclear Plant and have determined that the transient loads described in the enclosure to our letter are being appropriately accounted for in your design.

The enclosure to our letter of November 14, 1975, was a preliminary listing of potential requests for additional information should we later determine that a reassessment of the vessel support design is required. We have now determined that a reassessment of the Bellefonte Nuclear Plant reactor vessel support design is required. Please inform us within 30 days, after receipt of this letter, of your schedule for providing your evaluation of the adequacy of the pressure vessel supports under the most severe design basis pipe rupture condition. Your evaluation should include responses to the request for additional information contained in Enclosure 1. Forty copies of the responses are required for the staff.

We have been discussing with the pressurized water reactor vendors and various architect-engineer firms the generic aspects of this problem. 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, our generic review may consider other areas in the nuclear steam supply system and further evaluation may be required.

OFFICE➤						
SURNAME➤						
DATE➤						

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
O. D. Parr

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

Enclosure:
As stated

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
629 New Sprankle Building
Knoxville, Tennessee 37902

Mr. E. G. Beasley
Tennessee Valley Authority
307 Union Building Annex
Knoxville, Tennessee 37902

Mr. T. Spink
Licensing Engineer
Tennessee Valley Authority
303 Power Building
Chattanooga, Tennessee 37401

OFFICE➤	LWR-3	LWR-3 ODP	~~OELD~~			
SURNAME➤	WPike:ld	OParr	W.D. Paton			
DATE➤	6/21/76	6/25/76	6/22/76			

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.

3.89 Provide engineering drawings of the reactor support system sufficient to show the geometry of all principle elements and materials of construction.

3.90 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 maximun stress in each principle element of the support system and the corresponding allowable stresses.

3.91 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.

3.92 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.

3.93 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:

3.94 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.

3.95 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.

3.96 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.

3.97 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.

3.98 Provide and justify the break type and area used in each analysis.

3.99 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.

3.100 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.

3.101 Provide a table of blowdown mass flow rate and energy release rate as a function of time for the reactor cavity design basis accident.

3.102 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.

3.103 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:

3.104 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.

3.105 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-02-8

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

3.106 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.

3.107 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).