Carl L. Newman
Vice President

Regulatory Docket File

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September 4, 1975

Re Indian Point Unit No. 3
Docket No. 50-286

Mr. D. B. Vassallo, Chief
Light Water Reactors
Project Branch 1-1
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Vassallo

Your letter dated July 22, 1975 requested additional information in order to assess the adequacy of the reactor vessel support system for a double-ended break in the reactor coolant system cold leg pipe at the reactor vessel nozzle. By letter dated July 30, 1975, we indicated that the information requested in Items 1 and 2 would be provided by August 15, 1975. By letter dated August 15, 1975, we forwarded the information requested in Item 1 for your consideration and also indicated that the response to Item 2 would be provided by August 29, 1975.

Item 2 of your letter requested the detail loads used in the design analyses of the reactor supports giving magnitude, direction of application and the basis for each load, as well as the calculated maximum stress in each principal element of the support system and the corresponding allowable stresses.

The basis for the design of the Indian Point 3 reactor vessel support system included consideration of breaks outside the primary shield wall. These break locations are shown in attached FSAR Figure 04.30a-1. Break locations 1 and 2 on this figure, which are postulated as double-ended breaks at the juncture of the hot leg and elbow entering the steam generator and at the reactor coolant pump discharge respectively, are the most significant, as they induce the highest loading on the reactor vessel supports. Break locations 8,9 and 10, although nearer to the vessel, are less severe than break locations 1 or 2 because they are postulated ruptures in the attached branch lines with opening areas much less than a double-ended rupture of the primary loop.

The reactor vessel support system was sized using maximum combined load set cases of (1) 1,109 kips tangential, 322 kips radial, and 1329 kips vertical; (2) 613 kips tangential, 322 kips radial and 1459 kips vertical; and (3) 1004 kips tangential, 322 kips radial and 934 kips vertical. Each load set was applied to an individual support in the design phase. The loads used in the analysis of the

support system were derived from the criteria in the FSAR for Indian Point 3.

The reactor vessel support system was analyzed for the loading combination of the normal loads, seismic loads, and the loads due to postulated pipe ruptures. These loads are considered to be statically applied to the vessel. The loads from the seismic event and the postulated pipe rupture are combined by the square-root-sum-of-the-squares method. The pipe rupture loading effect on the support considers the jet thrust acting normal to the plane of the postulated pipe rupture with a magnitude equal to the operating pressure multiplied by the pipe flow area.

Table 1 shows the maximum loads and combined loads derived from the application of this criteria to the reactor vessel supports. These loads are applied in the analysis to each individual support. Since the applied lateral loads on the reactor vessel are resisted by two support shoes, the total lateral load applied to the reactor vessel is twice that applied to each shoe. The highest lateral load the reactor vessel will see given the above described design basis is 2444 kips. The steam generator supports and the reactor coolant pump supports attached to the unbroken loops also resist the applied lateral loads. This effect which was conservatively not included, is significant as it will reduce the loads transferred to the vessel support, since the total loads would be redistributed to all the supports.

The concrete wall was designed to the requirements of the 1963 American Concrete Institute Code (ACI). For the applied loads in Table 1, the maximum bearing stress in the concrete is calculated to be 530 psi. This occurred in the concrete under the ring girder. The ACI allowable bearing stress for combined seismic plus pipe loads is 2700 psi.

The ring girder was designed to the requirements of the 1963 American Institute of Steel Construction Code (AISC). For the applied loads in Table 1, the maximum stress is calculated to be 31.8 ksi in the top beam of the ring girder on the edge away from the reactor vessel. The AISC allowable stress for non-normal loading conditions usually associated with plant emergency condtions is 32 ksi ($.9f_y$). The extreme loadings of the pipe rupture are, therefore, below this stress limit used for the upset condition. The yield stress of the ring girder material is approximately 36 ksi.

The reactor vessel shoe was designed to the requirement of the AISC Code, also. For the applied loads, the maximum stress was a shear stress in the pins between the shoe and the cooling plate/ring girder assembly. The shear stress in these pins was 54 ksi which can be compared to a shear yield stress ($f_y/\sqrt{3}$) of 75 ksi. This is the

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stress at which the pins will yield in shear.

The ring girder and the shoe are both held to stresses below the yield stress of the material. This is very conservative, as additional capacity could be developed by taking the plastic response of the structures into account. The criteria for supports in the Indian Point 3 FSAR (Table A.1-2) states that the permanent deflections must be limited to maintain the supported equipment within its faulted condition stress limits. By keeping the stresses below the yield stress, there is assurance that the deflections will be very small and that there will be no permanent deflections. Therefore, the FSAR criteria is conservatively satisfied.

In our letter of July 30, 1975, we indicated that a schedule for the information requested in Items 3 and 4 would be provided by August 29, 1975. Our evaluation and scoping of this information is continuing. We anticipate this schedule will be supplied by September 30, 1975.

Very truly yours

Carl L. Newman

mrb

Carl L. Newman
Vice President

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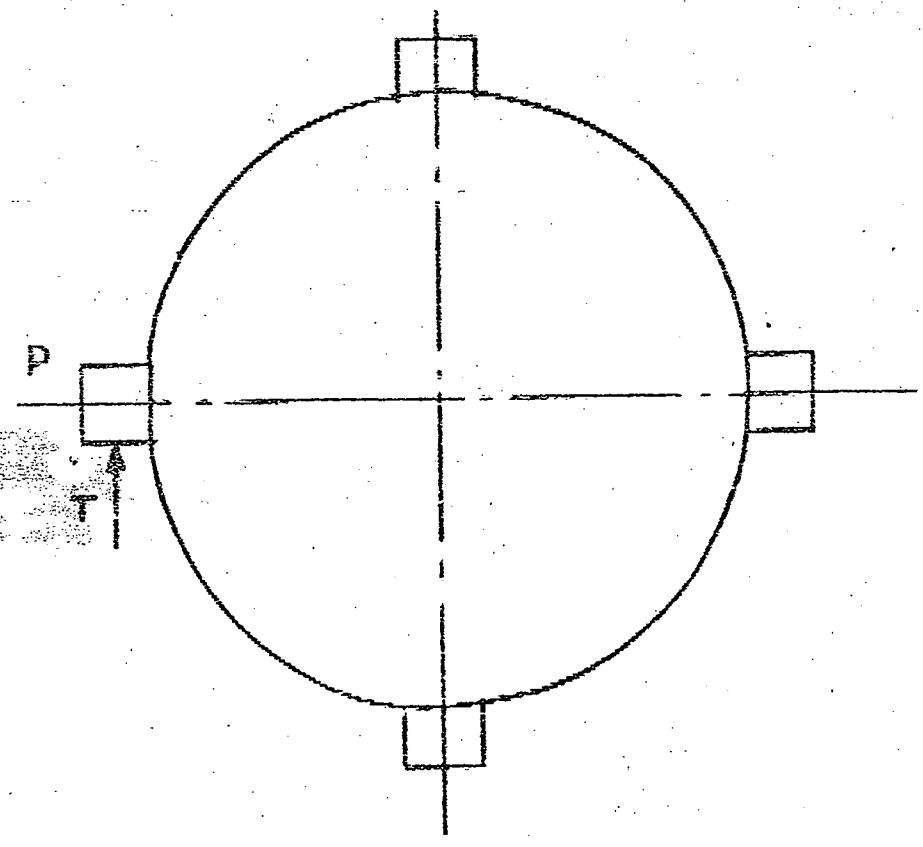
TABLE 1

FORCES ACTING ON EACH REACTOR VESSEL SUPPORT

	A	B	C	D	Combination	
	Reactor Vessel Weight & Piping Reaction	Cold Leg Pipe Rupture	Hot Leg Pipe Rupture	Seismic	A,B,D	A,C,D
P (kips)	934	0	0	395	1329	1329
T (kips)	0	669	744	969	1178	1222



P - Vertical

T - Tangential

RV = REACTOR VESSEL

S.G. = STEAM GENERATOR

RCP = REACTOR COOLANT PUMP

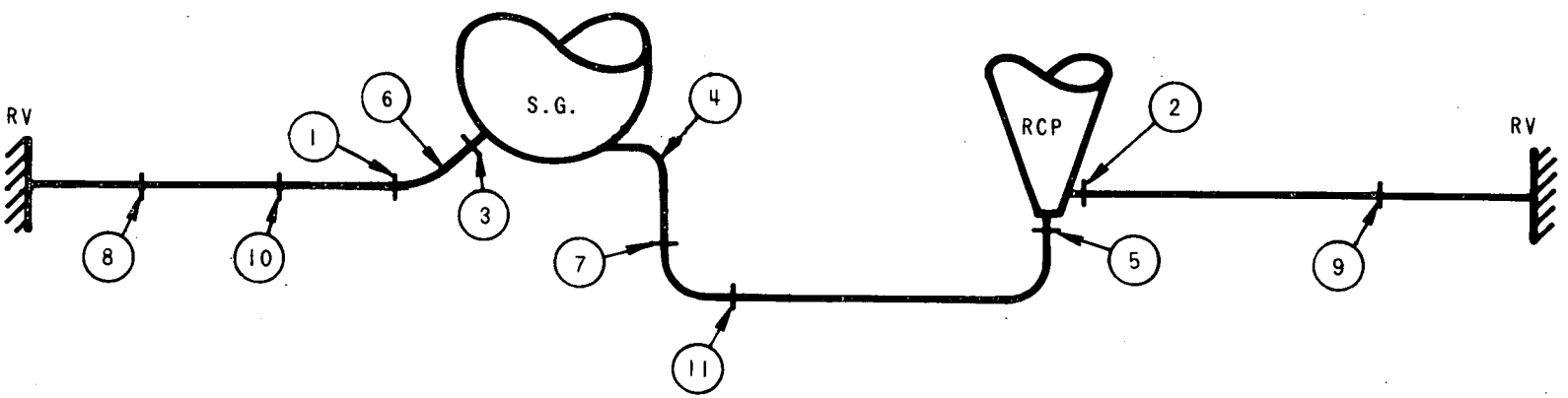


Figure Q4.30a-1 Reactor Coolant Loop Postulated Break Locations