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Ltr re their 6-29-73 ltr, trans the following:

PLANT NAME: H. B. Robinson Unit # 2

ENCLOSURES:
REPORT: "Postulated Pipe Failure Analysis Outside of Containment"

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Regulatory File

Carolina Power & Light Company

September 12, 1973

File: NG 3514 Serial: NG-73-387

50-261

Mr. Robert J. Schemel, Chief
Operating Reactors Branch No. 1
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Schemel:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
POSTULATED PIPE FAILURE OUTSIDE CONTAINMENT

In our letter of June 29, 1973, we submitted a document entitled "Postulated Pipe Failure Analysis Outside of Containment," which outlined the analyses which would be performed and the methods employed in determining the effects of pipe failures outside the Containment Building of the H. B. Robinson Unit No. 2 Plant. Included in this document as Appendix A was a number of specific areas for which analyses were required. The purpose of this letter is to provide the results of the analyses to date, and indicate the remaining areas under consideration and the analyses to be performed. The results are attached and can be summarized in the following manner:

1. A postulated steam line break at the turbine would whip and impact containment. The resultant penetration and effective impactive load are limited to about 20 percent of the allowable. It can, therefore, be concluded that containment integrity is maintained.

2. The steam header does not fail as the result of a postulated steam line break.

3. The safeguards switchgear is not compromised by a postulated feedwater line break.

4. The pressure, temperature and flooding levels in the auxiliary building pipe chase, the charging pump room and auxiliary motor driven feedwater pump rooms are not sufficient to cause structural problems nor failure of other essential systems.

The submittal of a final summary report to the AEC is expected to be made by October 31, 1973, upon completion of the remaining analyses.

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A discrepancy between our submittal of June 7, 1973, and June 29, 1973, has been pointed out by a member of your staff, and concerns the identification of specific areas for analysis and investigation. On June 7, we identified the sampling room as an area where pipe breaks should be considered. This was subsequently deleted in our June 29 submittal without explanation. The reason for deleting the sampling room resulted from further investigations in the intervening time period which showed that the small size of the sampling lines resulted in energies in the lines lower than those required to damage the structural integrity of the room. Thus, this item is considered resolved.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:mvp

Enclosure

cc: Messrs. C. D. Barham
N. B. Bessac
T. E. Bowman, III
B. J. Furr
D. V. Menscer
D. B. Waters

Item A.1

(a) An investigation of most probable break locations has been made with the conclusion reached that with a single exception, stress level criteria as defined in Section 2.2.1*does not govern break locations. Table 3.1 shows the magnitude and location of the three highest stressed points in steam lines MS-1 through MS-5, which comprise the three main steam lines from the containment boundary to the steam header and the two lines from the steam header to the turbine as shown in Figure 3.1 and 3.2. While they do not govern containment evaluation, feedwater line stresses are shown in Table 4.1 and routing in Figure 4.1 through 4.4.**

The limiting break location is at the terminal end inlet to the turbine. For such a postulated circumferencial break, the steam line is assumed to hinge at the main steam header. As shown in Figure 2, such a postulated plastic rupture would permit the rotating steam line to impact containment. Calculations for a resultant pipe whip velocity of 569 ft./sec. and an effect mass of 25.04 $lb\text{-}sec.^2/in.$ and contact area of 17.54 sq. ft. results in a containment penetration of about 8.0 inches. The equivalent static impact load is determined as 1175 kips.

No detailed structural analysis of the containment shell was performed for this missile effect as an impulsive load, but a comparison was made with the equivalent static load as determined by the forcing function developed by Riera for the analysis of the 3 Mile Island containment structure for aircraft impact. The effective load determined for the steam line break is about 20 percent of the load used in the successful evaluation of 3 Mile Island containment. It is, therefore, concluded that the containment will maintain its integrity in the event of a postulated steamline break.

(b) A structural analysis was performed to determine the structural integrity of the steam header in the event of a postulated steamline break. In

*"Postulated Pipe Failure Analysis Outside of Containment," submitted June 29, 1973.

**Note: the referenced figures are unavailable at this time, but will be incorporated in the final report.

this analysis, it was assumed that the broken pipe would exert on the steam header a moment equal to the upper bound plastic hinge capacity of the broken pipe and an axial load equal to the thrust load determined from the broken line. The header was assumed restrained by the four other intact steam lines attached to it plus the support structure of the turbine building.

The turbine building structure was modeled as two interacting planer frames having the composite stiffners of the several turbine building bents in the immediate vicinity of the header support. Using the computer program STRUDL, a 6x6 diagonal stiffness matrix was determined which represented the structural stiffness of that portion of the turbine building supporting the steam header. This building stiffness was used as an input boundary restraint to a mathematical model of the steam header and attached piping. This system was analyzed by the computer program WESDYNE for loads determined from the postulated pipe break to develop resultant loads in the attached pipe and forces in the restraint structure. The loads in the attached piping were evaluated and found in the limiting case to be about 15 percent of the lower bound plastic hinge moment of the attached piping. The loads in the support structure were reapportioned to the actual building frame bents in proportion to their relative stiffness and evaluated in accordance with AISC-69 Building Specification requirements with allowable stresses increased by the factor of 1.65 which is consistent with the Faulted Condition nature of the pipe break load.

Results of this analysis indicate the building structure will not fail as the results of postulated pipe break with loads in the turbine bent members averaging about 60 percent of their capacities with the maximum member load in the highly redundant reaching 102 percent of its capacity. Therefore, it can be concluded the steam header support system will not fail in the event of steamline break.

(c) Since the steam header is determined not to fail, the limiting pipe whip effect has been evaluated as given in (a).

Item A.2

(a) It has been determined, as indicated in Table 8.0-1,* that the steam pressure transmitters will be required to remain functional.

(b) In the event of a feedwater line crack as defined in Section 2.2.3, in the limiting case a transmitter could be subjected to a line load of about 0.745K/in. over 8 inches. Evaluating this load on a transmitter either as a horizontal line load along the top of the transmitter on a 45^{0} angle or as a vertical load along the length of the transmitter has not as yet been completed; the physical characteristics of the transmitter have not as yet been determined. However, in the limit if the transmitters are incapable of carrying the superimposed jet load, a simple jet shield may be installed requiring a relatively small 6 kip load capacity. Final evaluation of the existing transmitters should be available by October 1, 1973.

Item A.3

(a) It has been determined a plastic hinge mechanism would form resulting in a feedwater line pipe whip on the concrete structure surrounding the safeguards switchgear room.

(b) A detailed analysis of the equations of motion of the feedwater line in the event of postulated rupture indicate it would reach a normal impact velocity on the 18 inch thick switchgear room reinforced concrete wall of 190 ft./sec. With an impact area of 1.396 sq. ft., the depth of penetration is determined as $\sim$1.0 inch. The equivalent static impact load is 2.1 kips. The capacity of the section is about 10 kips. It can, therefore, be concluded the impact of the feedwater line would not effect the safeguards switchgear.

Item A.4

The volume of 40,700 cu.ft. and blowdown areas of about 108 sq.ft for the piping penetration gallery have been determined and preliminary calculations indicate the pressure buildup in the gallery would be less than 1.5 psi,

*"Postulated Pipe Failure Analysis Outside of Containment," submitted June 29, 1973.

the temperture less than 200^{0}F, and flooding would be limited to less than 2 feet. A definitive computer analysis is underway which is expected to be complete by October 15, which should confirm these preliminary calculations.

Item A.5 and A.6

Flooding of the charging pump room and auxiliary feedwater is expected to be limited by floor drain capacity and the fact that these rooms are located at the grade. The drain capacities are currently being evaluated and should be completed by September 30. Preliminary calculations indicate flooding will be less than 1.5 ft.; hence, it can be concluded unbroken charging and feedwater systems will continue to function.

TABLE 3-1

Stress Values — Main Steam #1, #2 & #3

Containment to Main Steam Header

Allowable Stress Values:
(0.8 $S_{allowable}$)
Operational Plus Seismic Stresses < 27,400 psi

Thermal Stresses < 16,440 psi

Node Number *	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
1	6,372	2,000**	13,745	3,063	25,180
11	6,372	2,000	14,953	417	23,742
12	6,372	2,000	15,148	1,986	25,506
18	6,372	2,000	4,667	217	13,256
20	6,372	2,000	12,710	3,989	25,071
31	6,372	2,000	14,371	533	23,276
32	6,372	2,000	14,490	2,264	25,126
39	6,372	2,000	779	235	7,386
40	6,372	2,000	12,365	4,856	25,593
52	6,372	2,000	15,782	668	24,822
53	6,372	2,000	15,049	2,584	26,005
61	6,372	2,000	6,052	438	14,852

* Node Numbers refer to figure 3-1

** Assumed ~~maximum~~ dead weight stress

TABLE 4-1

Stress Values -- Feedwater Piping

From Auxiliary FDW Pumps "A" & "B"

Allowable Stress Values:

Operational Plus Seismic Stresses < 30,000 psi

Thermal Stresses < 18,000 psi

Node Number *	Operational and Seismic Stresses: Pressure	Gravity	Thermal	Seismic (OBE)	Total
90	4,017	2,000**	9,934	3,394	19,345
92	4,017	2,000	9,296	5,991	21,304
93	4,017	2,000	8,034	4,720	18,771
127	4,017	2,000	5,716	1,774	13,507
128	4,017	2,000	5,864	1,274	13,155
133	4,017	2,000	5,187	573	11,777
140	4,327	2,000	7,459	592	14,378
141	4,327	2,000	9,698	1,889	17,914
142	4,017	2,000	3,077	2,898	11,992
143	4,017	2,000	2,895	2,188	11,100
145	4,017	2,000	3,285	5,756	15,058
146	4,017	2,000	1,859	1,202	9,078
148	4,017	2,000	2,062	2,660	10,739
149	4,017	2,000	4,312	998	11,327
154	4,017	2,000	5,546	1,252	12,815
155	4,017	2,000	5,469	1,420	12,906

* Node Numbers refer to figure 4-1

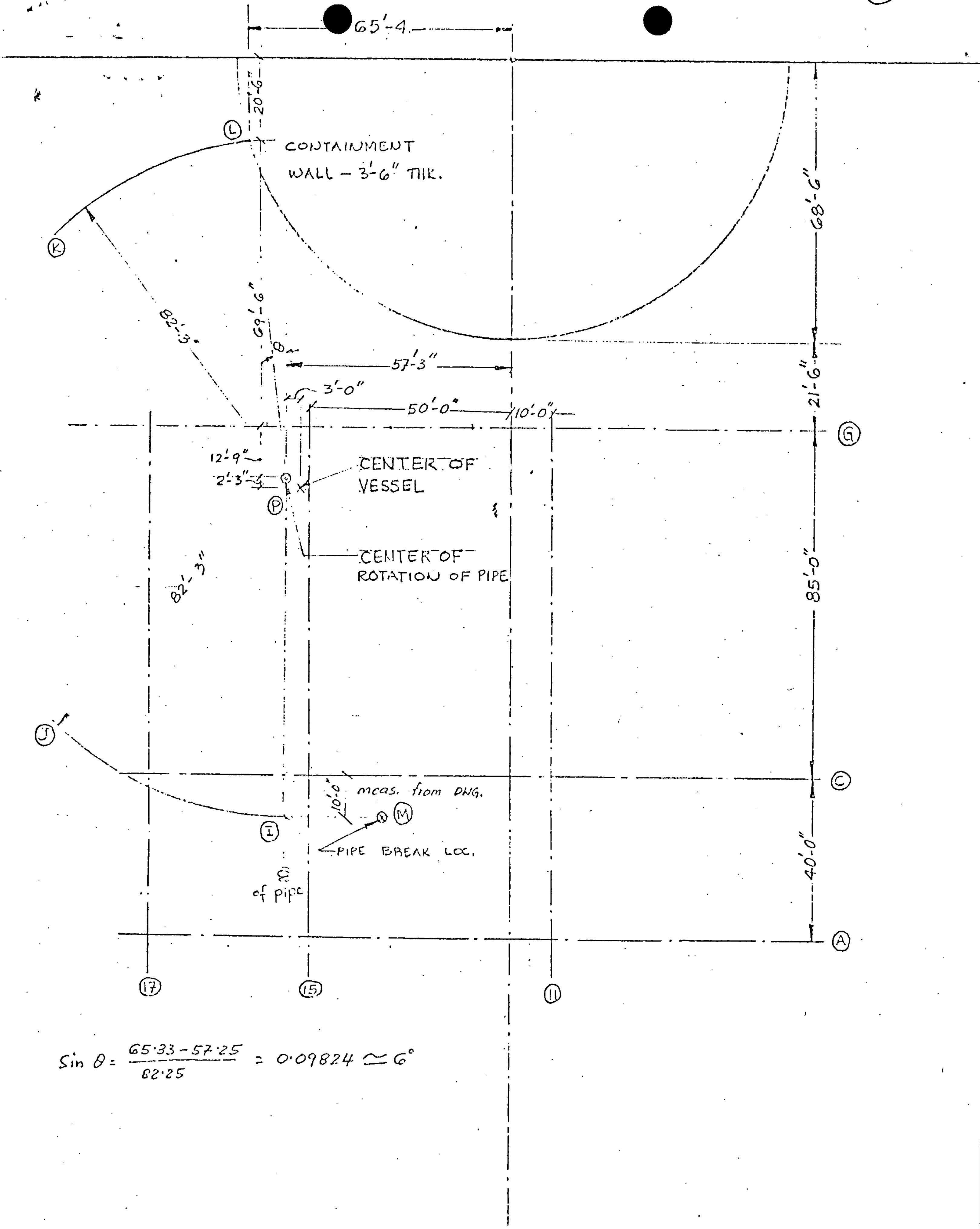
** Assumed maximum dead weight stress

TABLE 4-1 (Continued)

Stress Values — Feedwater Piping

From Auxiliary FDW Pumps "A" & "B"

Node Number	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
160	4,327	2,000	6,390	1,064	13,781
161	4,327	2,000	6,928	2,084	15,339
183	4,017	2,000	8,641	1,683	16,341
188	4,017	2,000	8,128	7,263	21,408
191	4,017	2,000	9,713	4,477	20,207
222	4,017	2,000	10,679	602	17,298
223	4,017	2,000	11,033	638	17,688
250	4,017	2,000	6,569	3,485	16,071



$$\sin\theta = \frac{65.33 - 57.25}{82.25} = 0.09824 \simeq 6^\circ$$