REACTOR COOLANT SYSTEM
ASYMMETRIC LOCA LOAD EVALUATION

ST. LUCIE UNIT 1 - DOCKET NO. 50-335

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SUMMARY

In May 1975 the NRC Staff was informed by a pressurized water reactor licensee that loads resulting from a hypothetical rupture of the reactor coolant cold leg pipe in the immediate vicinity of the reactor pressure vessel (RPV) may have been underestimated.

In November 1975 the Staff agreed that these loads should be considered and evaluated on a generic basis.

Florida Power & Light's response to the Staff's letter of November 26, 1975 indicated that the support system design incorporated the reaction forces associated with the large arbitrary reactor coolant pipe ruptures, and that further, it had been shown to acceptably accommodate the additional loads associated with differential pressures within the reactor cavity as shown in Appendix 3H of the Final Safety Analysis Report.

The Staff requested that further internal asymmetric load (IAL) evaluations be conducted. FP&Ls letter of February 9, 1976 documents the Company's commitments to evaluate the reactor vessel support capability for the limiting break, a commitment which is restated in Supplement 2 to the Unit 1 Safety Evaluation Report (SER), dated March 1, 1976.

In September 1977 FP&L transmitted to the NRC a report assessing the margin in design of the vessel supports when the internal asymmetric loads are added to all previous loads. The report concluded that the supports would adequately withstand all the loadings. However, since the analysis did not account for gaps between the vessel and the core barrel, and also the vessel and the support structures, an analysis was initiated at the same time to account for these effects.

The Staff's letter of February 16, 1978 requested that the evaulations conducted to date be expanded in scope to include an assessment of the reactor pressure vessel, fuel assemblies and internals, control element assemblies, primary coolant piping and attached ECCS piping, all primary system supports, and the biological and secondary shield walls for a spectrum of breaks in the primary system. FP&Ls March 1978 response stated that its August 1977 report was fully responsive to the Staff's SER requirement, that the St. Lucie 1 design was acceptable and that the large instantaneous pipe breaks being postulated were overly conservative. The response went on to say that FP&L would pursue additional analyses once the Staff approved the analytical methods used in the August 1977 report. This reply notwithstanding, FP&L being sympathetic with the Staff's desire to assess any potential risk to public health and safety from postulated events, expanded the analysis referred to above, to also assess the additional items identified by the Staff.

This report discusses the results of this expanded analysis. The combination of thrust, external, and internal asymmetric loads resulting from the inlet pipe circumferential break present the largest load to the vessel supports among those that would ensue from any of the design basis breaks listed in Appendix 3H of the FSAR.

The results confirm that the vessel supports will adequately withstand all the loads resulting from the postulated circumferential break in the vessel inlet pipe. The cold leg guillotine break in the cavity is the break which results in the largest loading of the vessel supports. Therefore the vessel supports are clearly adequate for all other break locations. This reaffirms the conclusions of the August 1977 report.

Results also show that all supports for the primary system are adequate for all break locations, that the stresses in the intact primary piping and attached lines are sufficiently low to ensure performance of intended functions, and that the biological shield wall performs its intended function. The secondary shield wall is designed for postulated primary system ruptures within the steam generator subcompartments.

The analyses of the adequacy of fuel assemblies internals, and control element assemblies is in progress. Results are expected in July of 1980.

The Staff, at a meeting in January 1980, further requested that seismic loads be separately identified. All results presented herein, as well as the August 1977 report, include the SRSS combination of LOCA and seismic loads, consistent with the requirement of NUREG-0484. For those combinations design seismic loads have been used and are hereby attached for use by the Staff. In all cases, design seismic loads are considerably higher than calculated peak seismic loads.

Qualitatively, the small displacements observed for the vessel and core barrel for the worst break analyzed, strongly suggests that the analyses now in progress of the fuel/internals and CEDMs will indicate acceptable results.

It must also be noted that since the submittal of the August 1977 report to the Staff, additional work has been reported to support FP&Ls contention that the types of instantaneous pipe breaks being postulated by the Staff are excessively conservative.

1.0 INTRODUCTION

During a postulated loss of coolant accident in the form of a circumferential pipe rupture at the inlet nozzle of the reactor pressure vessel, a decompression of the reactor pressure vessel occurs over a short period of time. Decompression waves originated at the postulated break travel around the inlet plenum and propagate downward along the downcomer annulus. The finite time required by the decompression disturbances to travel about the vessel causes a transient pressure differential field to be created across the core support barrel (CSB) and the vessel inner surface. This field imposes a transient asymmetric loading on the core-support-barrel as well as the vessel itself. Since the postulated pipe break is located within the biological shield wall, the blowdown fluid flashing into the reactor cavity also causes a transient pressurization acting on the vessel. This external pressurization is also asymmetric. The internal asymmetric loading (IAL) and the external asymmetric loading act in the same direction for breaks occurring in the cold leg piping. For breaks in the hot legs, the internal asymmetric load is virtually absent in the horizontal direction, hence the two loads are additive in the vertical direction only. These loadings are transmitted to the reactor vessel support system. The resultant reaction forces at the support interfaces must be considered in the evaluation of the adequacy of the support system together with the thrust load resulting from the break, other operating loads, and postulated seismic loads. The seismic loads and normal operating loads, as well as the EAL have been previously analyzed in Appendix 3H of the Final Safety Analysis Report.

Breaks outside cavity can result in IAL imposed on the reactor pressure vessel and internals, and in EAL on the reactor coolant pump and steam generators.

For the breaks outside the cavity, the adequacy of the primary system supports is assessed for full breaks at appropriate primary system locations. The cavity breaks are determining breaks for the assessment of the adequacy of piping attached to the primary system piping.

The circumferential pipe rupture at the inlet nozzle of the reactor pressure vessel is determined to be the design basis break for the evaluation of the vessel support adequacy. A break at the outlet nozzle would not produce a horizontal asymmetric pressure loading to the vessel. Consistent with the Final Safety Analysis Report, a 4.0 sq. ft. cold leg guillotine break at the inlet nozzle is chosen for the analyses of the vessel support adequacy.

2.0 METHOD OF ANALYSIS

2.1 Reactor Vessel Supports

The adequacy of the reactor vessel supports is evaluated by determining the loads acting on the primary system which result from a postulated break at the inlet cold leg nozzle; the response of the primary system to the application of these loads; and the reaction forces generated by this response at the reactor vessel supports. The loads acting on the primary system consist of normal plus seismic loads, the thrust load, external asymmetric loads, and internal asymmetric loads. The latter three are combined in true time history fashion, added to the normal loads reactions, then the resultant reaction loads at the supports are combined with design reaction loads resulting from the postulated seismic (SSE) events by SRSS techniques, to obtain the overall reaction load at each of the supports. Design seismic loads are provided for each primary system support in the three orthogonal directions in Table 1. It should be emphasized that computed peak seismic loads are in general substantially less than the design seismic loads; thus providing an element of conservatism in this analysis. Table 2 gives a sample comparison of calculated and design seismic loads at representative locations.

The following subsections describe the methodology employed to evaluate each of the thrust, external asymmetric and internal asymmetric loads. Inherent in the evaulation of these loads is the determination of the time required to open up the break to the area being analyzed.

2.1.1 Break Opening Time and Thrust Loads

The St. Lucie plant primary coolant piping in the vicinity of the vessel is restrained from unlimited motion following complete severance in the portion within the cavity by restraints in the primary shield wall penetrations and wire ropes around the reactor coolant pumps. This restraining system has been previously described in the FSAR. Following an arbitrarily assumed instantaneous severance of the pipe at the nozzle, the two ends of the broken pipe separate under the action of the thrust imposed by the instantaneous tension release followed by the blowdown of the escaping fluid, and form a combined break area which varies with time as given in the following equation:

$$A_b(\tau) = \frac{\pi R_i \beta x(\tau)}{45} + 2R_i^2 \left[\pi - \left(\frac{\beta\pi}{90} - \sin 2\beta\right) \right] - \frac{\pi\beta(R_i+R_o)t}{180} \quad (1)$$

where R_i and R_o are the inner and outer pipe radii, t is the pipe thickness, x is the axial separation of the two ends which varies with time τ, and

$$\beta = \cos^{-1} \frac{y(\tau)}{2R_i} \quad (2)$$

wherein y is the radial separation of the two broken ends which also varies with time.

This equation is solved in iterative fashion together with the equation for the combined tension release and blowdown force, given below

$$F(\tau) = P_{dl}(\tau)A_p + \rho_{dl}(\tau)A_b(\tau) \frac{V_e^2(\tau)}{g} \quad (3)$$

to yield the correct forcing function and break area as a function of time. In equation (3), P_{dl} and ρ_{dl} are the pressure and fluid density in the discharge leg, respectively, A_p is the cross-sectional area of the pipe, V_e the blowdown velocity, and $A_b(\tau)$ is defined in (1) above.

The motion of the piping system under the application of the force given by (3), is computed by modelling the discharge leg, the pump, and the cross over leg with an elasto-plastic finite element computer program, PLAST[17] considering the steam generator and the vessel to remain motionless.

Results of the analyses indicate that at least 18 msec. are necessary for the pipe ends to separate the overall area of 4.0 sq. ft. referred to in the FSAR. This analysis also indicates that as a result of plastic rotation at the pump, it is possible for the pipe ends to separate further, to a maximum area of 7.78 sq. ft. The time required for this area to be achieved, however, would be in excess of 25 msec. The longer time required for opening the larger break insures that the IAL resulting from the two breaks are virtually identical. The larger break does however result in a larger external

horizontal asymmetric load (external vertical asymmetric loads are virtually identical for the two breaks). Since the 4.0 sq. ft. break had been one of the design basis breaks in the FSAR, all analyses used that break area. However, consideration is given to whether the system is capable of accommodating the larger break. As discussed in the subsequent section, the system is in fact adequate for the largest of the breaks.

2.1.2 External Asymmetric Pressure Loads (Reactor Cavity)

The reactor subcompartment analysis for St. Lucie Unit #1 had been performed for stipulated LOCA conditions including a 4.0 sq. ft. cold leg guillotine break, and the results had been submitted to the NRC in the FSAR and approved by the NRC during the course of the operating license review. The results for the 4.0 sq. ft. cold leg guillotine break, as reported in Reference 2, have been directly used in the present study. This results in conservatism of the analysis since the cavity response had been predicated on a break opening time of 10 msec, whereas 18 msec. is needed to achieve this size break. The peak external asymmetric forces across the reactor vessel, that would result from the larger 7.78 sq. ft. break, would be approximately 40 percent larger. This is predicated on a ratio of 1.39 between peak and average energy flow to the cavity resulting from a 7.78 sq. ft. and a 4.0 sq. ft. cold leg break respectively.

In the original analysis, however, two elements of conservatism had been introduced. First, the mass and energy releases had been increased by 10 percent and second, all insulation had been assumed to reamin in place in the reactor cavity and vent areas for the purposes of volume and vent area calculations in the mathematical model. The insulation in the upper cavity reaches would be crushed against the vessel upon cavity pressurization, resulting in an increased volume of approximately 15-20 percent.

Hence, realistic modeling of the insulation behavior, coupled with removal of the 10 percent conservatism in the mass and energy release would result in a predicted external asymmetric pressure load and cavity pressure load from a 7.78 sq. ft. break which is only 15 to 20 percent higher than those conservatively predicted.

2.1.3 Internal Asymmetric Pressure Loads

The model used to determine the pressure field at every point in the primary system following the postulated primary system breaks, from which the internal asymmetric forces on the vessel and core support barrel are deduced, is shown in Figure 1.

The RELAP-4[3] thermal hydraulic code is used to compute the thermodynamic properties in the model volumes and junctions. Results of the RELAP-4 model have been compared to results achieved by modelling the system with WHAM-6[4] for the period of time during which the latter can be applied with confidence, which is also the period of time of interest. Figure 2 shows the model employed for WHAM-6. A similar WHAM model and assumptions in its use, had been previously submitted to the Staff in the August 1977 report. The results of the two models are in good agreement, with RELAP-4 predicting a larger pressure differential across the core support barrel.

Results of the internal asymmetric loads analysis indicate that the peak forces across the core support barrel and the vessel are virtually insensitive to the break area, but extremely sensitive to break opening times. For instance, a change in area from 1 sq. ft. requiring 8 msec. to open to approximately 9.81 sq. ft. (complete double-ended area break) with an opening time of 36 msec., only results in a 2 to 3 percent increase in peak internal asymmetric loads, whereas a decrease in opening time from 36 msec. to 1 msec. for the full break brings about a threefold increase in internal asymmetric load.

2.1.4 Vessel and Primary System Structural Model

A non-linear elastic time history dynamic analysis of three-dimensional mathematical model of the reactor coolant system including details of the reactor internals, pressure vessel, supports, and piping was performed for the postulated pipe break to provide reactor vessel support reaction forces.

The structural model employed is shown in Figures 3(a) and 3(b). This model is three-dimensional and has 981 total static degrees of freedom and 77 mass degrees of freedom. The reactor vessel and all internal components are modelled at internal and support interfaces.

The STRUDL[5] computer code generates the condensed stiffness matrix used in the dynamic analysis from the physical definition of the structure. Hydrodynamic effects, including both virtual mass and annular effects are accounted for in the coupling between the RPV and the CSB, and between the CSB and the core shroud. The hydrodynamic (added) mass matrix is evaluated using the ADMASS[6] code.

The dyanmic analysis to determine the system response was performed using the computer code DAGS[7] and DFORCE[8].

The reactor pressure vessel support system is described in the FSAR. The modelling of the steel portion of the support is identical to that described in the FSAR in Appendix 3H. The basic model of the biological shield wall is also identical. However, a more refined analysis is employed for the latter, utilizing a NASTRAN[9] nonlinear solution procedure employing quadrilateral and triangular plane stress concrete cracking finite elements, instead of the STARDYNE method of solution described in Appendix 3H of the FSAR.

2.2 Reactor Coolant Piping, Connected Pipng and Other RCS Supports

2.2.1 Steam Generator Supports

Outside the reactor cavity, breaks have been assumed at appropriate locations.

The RCS supports most affected are the lower steam generator supports.

The primary system model is analyzed on an elastic basis for both hot leg and cold leg breaks, the hot leg break at the steam generator inlet being the determining event for the Steam Generator support.

This analysis is a static analysis which employs the computer code MEC-21 (Mare Island piping flexibility code)[12].

Both LOCA and design seismic loads are included in the analysis.

2.2.2 ECCS and Other Connected Piping

The analysis of the stresses generated in the ECCS lines and other lines attached to the primary loop involved a two step process. First, the time histories of the displacements are generated at each nozzle attaching said piping to the primary loop. The "worst" time history, irrespective of the location at which is occurs is

applied to the line which by configuration and other loading (normal and seismic) would result in the highest stresses. The stresses induced by LOCA motions for this particular configuration are added to previously computed normal and seismic (SRSS) stresses. The determining break for ECCS line evaluation is the cold leg nozzle break in the cavity.

2.2.3 Reactor Coolant Piping

The structural model for the primary system is also utilized to determine the stress conditions in the intact portion of the reactor coolant loop.

3.0 RESULTS OF THE ANALYSES

3.1 Vessel Supports

The loads calculated for each reactor vessel support by the method outlined in Section 2.1.4 are reported in Table 3 for the break chosen for the analysis; i.e., the 4.0 sq. ft. cold leg break at the inlet nozzle; for a range of reactor vessel support stiffnesses. This range covers the possible values of the overall stiffness of the individual reactor vessel supports, the real value being somewhere between the two extremes. It is not possible to quantify the stiffness value more precisely since the modelling of the boundary condition representing embedded steel in the biological shield is subject to variation.

In the support analyses however, the higher loads resulting from the use of the highest stiffness, have been utilized. This insures again that the absolute maximum load per support is computed. In reality, lower values are expected.

The capability of the reactor vessel supports is given in Figure 4 and Table 4 respectively for the RPV support pad capability and the weakest link in the steel support/biological shield structure.

Since the capability of the supports exceed the maximum loads computed for the given break, it is concluded that the existing support system is adequate for that break.

As stated in Section 2.1, it is possible that, as a consequence of the broken discharge line rotation about the pump, a larger break area could form within the cavity, up to a maximum of 7.78 sq. ft. This larger break area, requiring a proportionately longer time to open, has virtually no effect on thrust and internal asymmetric loads, but would increase the horizontal external asymmetric load by approximately 15-20 percent over that

used in the analysis, as explained in Section 2.1.2. The EAL represents approximately 40 percent of the overall load. Hence, a 20 percent increase in this load would result in less than a 10 percent increase in the overall loading. From Table 4 and Figure 4, it can be seen that this increase would be accommodated by the margins existing in the support system.

It is therefore concluded that the reactor vessel supports can withstand the largest break in the cold leg piping within the cavity.

Since cold leg breaks outside the cavity do not produce EAL loads and since the IAL is virtually unaffected by the area of the break as explained in Section 2.1.3, it is also concluded that the reactor vessel supports are capable of withstanding any load resulting from postulated ruptures outside the cavity.

A detailed analysis of the reactor loads resulting from hot leg breaks within the cavity has not been performed. The reasons are as follows: the stiffness of the hot leg pipe combined with the steam generator restraining action, results in a break area within the cavity which is smaller than the cold leg break area, hence resulting EAL would be lower than calculated for the cold leg break; although the thrust force initially would be larger, the IAL would not be colinear with thrust and EAL, but would in fact be approximately orthogonal to them. The resultant horizontal loads on the vessel supports therefore, would clearly be smaller.

For instance, the reactions at reactor vessel supports, due to a hot leg break have been compared to the reactions due to a cold leg break for thrust and subcompartment pressure only.

	Horizontal Hot Leg Break (Kips)	Horizontal Cold Leg Break (Kips)
Cold Leg Spt	4270	3270
Hot Leg Spt	0	3275

Although the load on the cold leg support is more severe for a hot leg break than for a cold leg break, when the effects of internal asymmetric loads are added, the cold leg break will govern.

Vertical loads would be of the same order of those experienced as a result of cold leg breaks, and the capacity of the support system to accommodate vertical loads is significantly higher than its horizontal capability. Hence clearly the reactor vessel support system is also capable of withstanding the effects of postulated hot leg breaks inside and outside the reactor cavity.

A similar conclusion had been reached in our August 1977 report. Differences in maximum loads reported herein from those reported in the August 1977 report are two fold. The August 1977 report did not consider internal gaps or gaps between the support pads and the support structure. The August 1977 report considered therefore that all loaded supports would be loaded simultaneously and share the load equally. The agreement of the overall loading between the present and the August 1977 results, confirm that the approach taken in 1977 to assess the loads was not unreasonable.

3.2 Other RCS Supports

The only supports on the primary system, other than the vessel supports, are the steam generator supports. Results of the analyses of the loads imposed on these supports from both hot and cold leg breaks in the system in combination with seismic loads, indicated that none of the design loads have been exceeded, with exception of the loads on the four holdown bolts at the vessel end of the steam generator sliding base and the sliding base itself. The computed and design loads are shown in Table 5. Individual examination of the sliding base, the bolts, and bolt anchorages however indicates that all can acceptably withstand the applied loads. It is therefore concluded that the existing supports design is adequate.

3.3 Reactor Coolant Piping

Table 6 reports the elastically calculated pipe rupture and seismic loads on intact reactor coolant piping associated with the broken loop for the worst break, which is the cold leg guillotine break at the vessel safe end. Examination of this table reveals that all loads fall within the allowable loads with the exception of the load at the RCP discharge nozzle, which exceed the allowable by about 13 percent, on an elastic basis.

Since this analysis is predicated on a 4.0 sq. ft. cold leg break, by the arguments presented in Section 3.1, consideration of the largest break that could occur at the vessel safe end; i.e., 7.78 sq. ft., requires that an increase in load of less than 10 percent be examined to assess the adequacy of the coolant piping. Such an increase can be readily accommodated at the RCP suction and RV outlet nozzles. The RCP discharge would be more overstressed (on an elastic basis) and the RV inlet would be very slightly overstressed.

Since only the fluid retaining integrity of this coolant piping needs to be maintained during the postulated LOCA, an analysis conducted on an elasto-plastic basis would conclude that this

integrity would be maintained at those nozzles. Since the amount of overstressing calculated on an elastic basis is relatively small, a plastic analysis was not considered necessary.

During the performance of this particular analysis it was calculated that the snubbers on the reactor coolant pumps are overstressed. These snubbers are not needed for these events. However their failure could affect the results. Hence, the analysis was repeated by taking no account of the snubbers. Results are also reported in Table 6. As can be clearly seen, the effect of the presence or absence of the snubbers is negligible.

3.4 ECCS and Connected Piping

The stresses computed from the analysis described in Section 2.2.2 are within 10 percent of the allowable, and hence it is concluded that the ECCS piping and other piping connected to the primary loop, is not adversely affected by the postulated event.

Table 7 compares the peak computed stresses, which include normal and seismic loads to the allowable stresses.

The margin existing between peak stresses calculated on an elastic basis and stresses that would be allowed within an elasto-plastic analysis further indicates that this attached piping would be able to withstand the imposed loads from the 7.78 sq. ft. larger cold leg guillotine break.

3.5 Seismic Loads

Pursuant to the Staff's request at the January 16, 1980 meeting, Table 1 provides the design seismic loads at the various support points in the Reactor Coolant System.

3.6 Control Element Assemblies

Although the analysis of the Control Element Drives response to postulated LOCA events is in progress, but will not be completed until July 1980, it is germane to point out that the CEAs are not needed for breaks in the RCS which exceed 0.5 sq. ft.

The assumption of a complete guillotine will result in breaks larger than 0.5 sq. ft.

4.0 CONCLUSION

Even though some analyses have not yet been completed, results obtained to date demonstrate that the existing design has significant capability to accommodate the postulated events. Additional information[10,11] which has become available since the August 1977 report, and which reinforces our contention, stated in that report, demonstrates that such events are of an acceptably low probability and cannot happen in the manner postulated for this analysis. The foregoing reaffirms our conclusion that the design of St. Lucie Unit 1 is acceptable.

5.0 REFERENCES

1/ "PLAST - An Elasto-Plastic Computer Program for Stress Analysis of 3-D Piping Systems and Components Subject to Dynamic Forces", submitted to the NRC as ETR-1001 - Ebasco Topical Report.

2/ St. Lucie Unit No. 1 FSAR, Docket No. 50-335, Amendment 44.

3/ "RELAP 4 - A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and Related Systems", User's Manual, ANCR-NUREG-1335.

4/ Fabic, S., "Computer Program WHAM for Calculating Pressure, Velocity and Force Transients in Liquid Filled Piping Network", Kaiser Engineering Report No. 67-49-R, November 1967.

5/ "ICES STRUDL II - The Structural Design Language Engineers User's Manual", MIT Press, Cambridge, Massachusetts, 1968.

6/ "ADMASS - A Computer Code for Fluid Structure Interaction Using the Finite Element Technique", Ebasco Services, Incorporated, 1979.

7/ "DAGS - CENPD 168, Revision 1 - Design Basis Pipe Breaks", September 1976.

8/ "DFORCE - Design Basis Pipe Breaks", September 1976.

9/ "MSC/NASTRAN - User's Manual", McNeal Schwandler Corporation, Los Angeles, California.

10/ "WCAP 9570 Class 3 - Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Thru Wall Crack", by Palusamy, S. S., and A. J. Hartmann, October 1979.

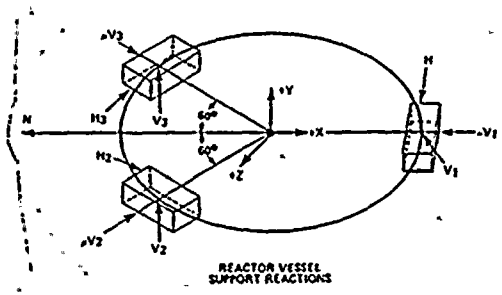
11/ Ayers, D. J. and T. J. Griesbach, "Opening and Extension of Circumferential Cracks in a Pipe Subjected to Dynamic Loads", Fifth International Conference of Structural Mechanics in Reactor Technology, Berlin, Germany, 1979.

12/ Griffin, J. H., "MEC-21 - A Piping Flexibility Analysis Program", TID-4500 (31st edition), LA-2924, UC-38, July 14, 1964.

TABLE 1
ST. LUCIE 1

NORMAL AND SEISMIC SUPPORT LOADS ($X10^6$ LB.)

CONDITION LOAD	NORMAL OPERATING DEAD WEIGHT	NORMAL OPERATING THERMAL + DEAD WEIGHT	DBE SEISMIC ±X	DBE SEISMIC ±Y	DBE SEISMIC ±Z	DBE SEISMIC ±X	DBE SEISMIC ±Y	DBE SEISMIC ±Z
H1	0	.028	.005	.002	−.644	.011	.005	−1.288
V1	.666	1.155	.032	.335	−.046	.064	.670	−.092
μV1	±.195	±.350						
H2	0	−.091	1.226	.001	+.355	2.452	.003	+.710
					−.264			−.528
V2	.664	.726	.017	.253	.380	.035	.507	.761
μV2	±.195	±.215						
H3	0	.079	1.139	−.019	.270	2.278	−.038	.540
				−.060	−.349		−.120	.698
V3	.634	.741	.367	.623	.743	−.006	.506	.746
μV3	±.195	±.215						
Z11	0	0	0	0	±.195	0	0	±.390
Z12	0	0	0	0	±.197	0	0	±.394
Y1	.300	0	.016	−.019	0	.033	−.039	0
Y2	.300	.315	−.057	.155	.060	−.114	.311	.121
Y3	.300	1.009	.086	.417	−.071	.173	.835	.143
Y4	.300	.320	−.054	.152	.058	−.108	.305	.116
X	0	0	0	0	0	0	0	0
μY	±.375	±.300						



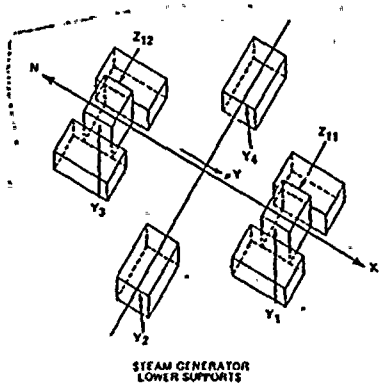


Table 2

Comparison of Peak Calculated
and Design Seismic (DBE) Loads
at Representative Locations

	Design (Kips)		Calculated (Kips)	
	Horizontal	Vertical	Horizontal	Vertical
Cold Leg Spt	2,455	1,268	522.6	354.6
Hot Leg Spt	1,293	762	515.0	429.4

Table 3

St. Lucie Unit #1

RV SUPPORT MAX ABS REACTIONS (KIPS) - LOCA + SEISMIC (SRSS)*

4 FT^2 CLG BREAK AT NOZZLE 1A OR 2A+

LOCATIONS	RV SPPT STIFFNESS VALUES	
	$K_H = 64.62 \times 10^6$ lb/in	$K_H = 77.54 \times 10^6$ lb/in
	$K_V = 59.71 \times 10^6$ lb/in	$K_V = 75.83 \times 10^6$ lb/in
#1A SPPT		
Vertical	1397	2317
Horizontal	1502	1587
#1B SPPT		
Vertical	2800	2251
Horizontal	5331	5473
Hot Leg SPPT		
Vertical	3458	3048
Horizontal	7493	7777

* For LOCA + Nop reactions, add these values to the vertical results:

#1A SPPT	710. K
#1B SPPT	726. K
Hot Leg SPPT	1157 K

+ For break at nozzle 1B or 2B, the loads on the cold leg supports would be reversed

Table 4

St. Lucie 1 Reactor Pressure

Vessel Support Capacity

Steel support structure	-	horizontal	8400 kips* (concrete is limiting)
Steel support structure	-	vertical downward	12000 kips
Reactor Cavity Wall	-	horizontal	~13000 kips**
Reactor Cavity Wall	-	vertical	not limiting
Reactor Support Pads	-	horizontal	See Figure 4
Reactor Support Pads	-	vertical	See Figure 4

* Load on individual girder

** Allowable resultant asymmetric mechanical load transmitted along girders to concrete, based on rebar mean axial stress being within yield.

TABLE 5

STEAM GENERATOR LOWER SUPPORT CALCULATED AND DESIGN LOADS

(REFER TO TABLE 1 FOR SYMBOLS) SUPPORT	CL GUILL NO. 1 + DBE (RSS)	HL GUILL NO. 2 + DBE (RSS)	DESIGN LOAD LOCA + DBE
Z11	727	42	3,600
Z12	806	40	–1,868*
FRONT Y1	–406.9	–2487.7	–1,770
SIDE Y2	–756.5	–1176.4	–1,737
BACK Y3	–605.0	+1249.0	– ,691
SIDE Y4	–300.1	–1175.9	–1,734
X-STOP		–5194.9	5,648
S	78	278	– ,301
Z1	194	40	–1,574
Z2	301	40	1,800

(K AND FT – KIPS)

@ ℄ STEAM GENERATOR/ SLIDING BASE SUPPORT SKIRT INTERFACE	(H L GUILL) LOCA	RSS LOCA & DBE	SLIDING BASE DESIGN LOADS
Fx	5205	5205	5653
Fy	–3582	– 3582	–2,471.0
Fz	–	6.8	11.0
Mx	8	418.3	32.0
My	65	33.6	24.0
Mz	–4609	4614	– 1003

*A NEGATIVE SIGN MEANS TENSION

Table 6

St. Lucie Unit #1 Reactor Coolant System Reactor Pressure Vessel and Reactor Coolant Pump Nozzle Loads Due to a 4 ft^2 Reactor Vessel 1A Inlet Nozzle Guillotine Break

PIPE RUPTURE RSS MOMENT (In-Kips)

Nozzle	RCP Snubber Acting	RCP Snubber Not Acting	Seismic Moment (In-Kips)	Allowable Moment (In-Kips)
RCP Discharge	109,300	109,600	5,910	96,810
RCP Suction	50,500	54,550	7,256	78,965
RV Inlet	71,750	71,910	5,272	78,965
RV Outlet	50,150	50,170	2,535	279,340

Table 7

St. Lucie Unit No. 1

Connected Piping Stresses Calculated vs. Allowable
4.0 sq. ft. CLG Inlet Break

Design Point (Refer to Figure 5)	Calculated Stress (Equ. 10 ASME)	Allowable Stress
1	39,070	48,600
2	75,152*	48,600
3	75,030*	48,600
5*	41,835	48,600
6*	43,475	48,600
7	47,690	48,600
8	33,430	48,600
19	20,171	48,600

* Functionability and integrity are assured if Level B (upset conditions) limits of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 are not exceeded. Functionability is important at points 5 and 6 where the valve is. At points 2 and 3, these limits are exceeded. However, Level D (faulted limits) are not exceeded at these two points. Level D limits are used to demonstrate that integrity is maintained. Equation (9) at those two points would yield 45,043 psi and 44,479 psi respectively with an allowable of 48,600 psi.

FIGURE 1
RELAP4 MODEL FOR ST. LUCIE PRIMARY COOLANT SYSTEM

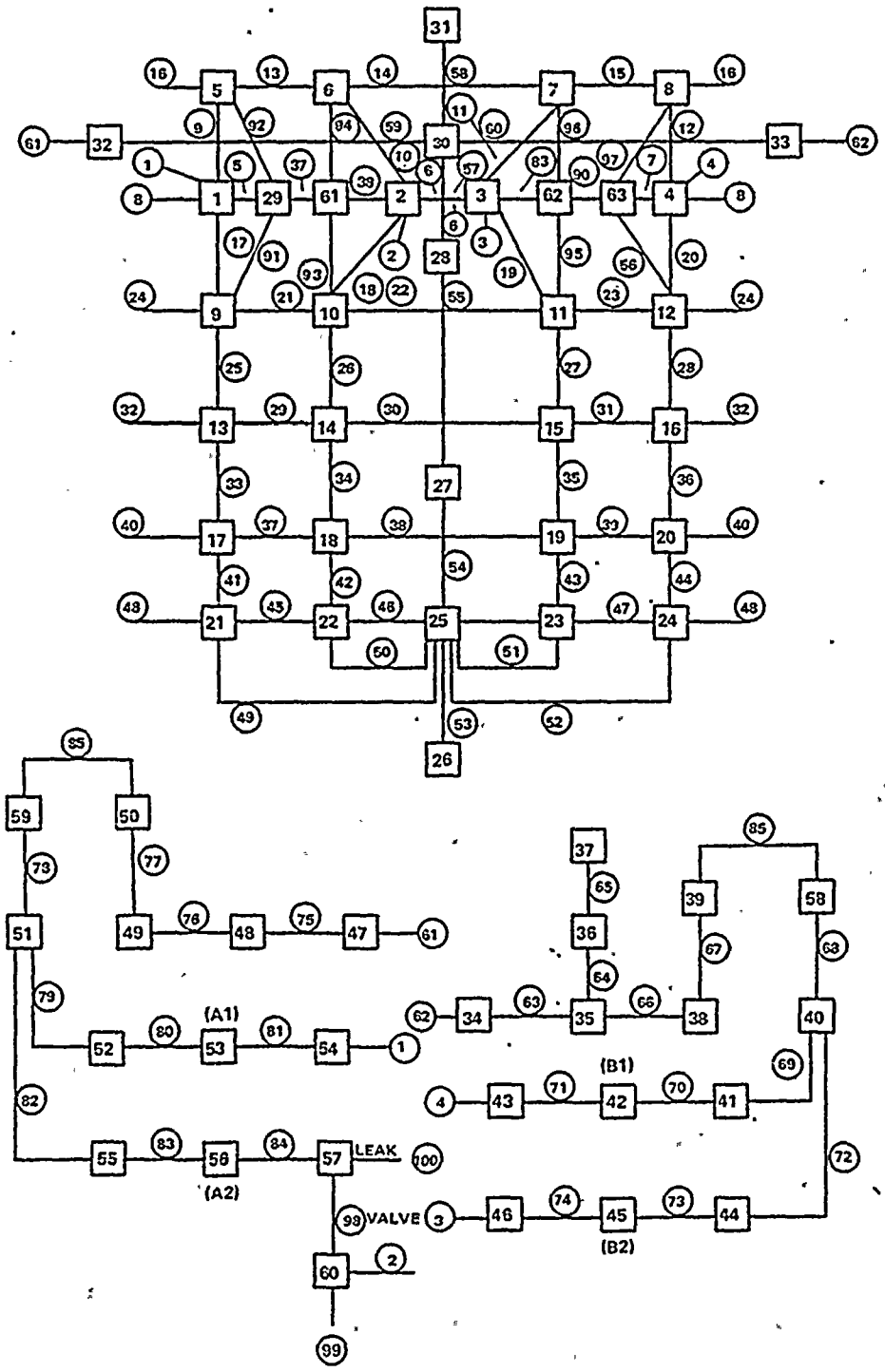


FIGURE 2
ST. LUCIE 1
RCS- WHAM/6 MODEL FOR IAL

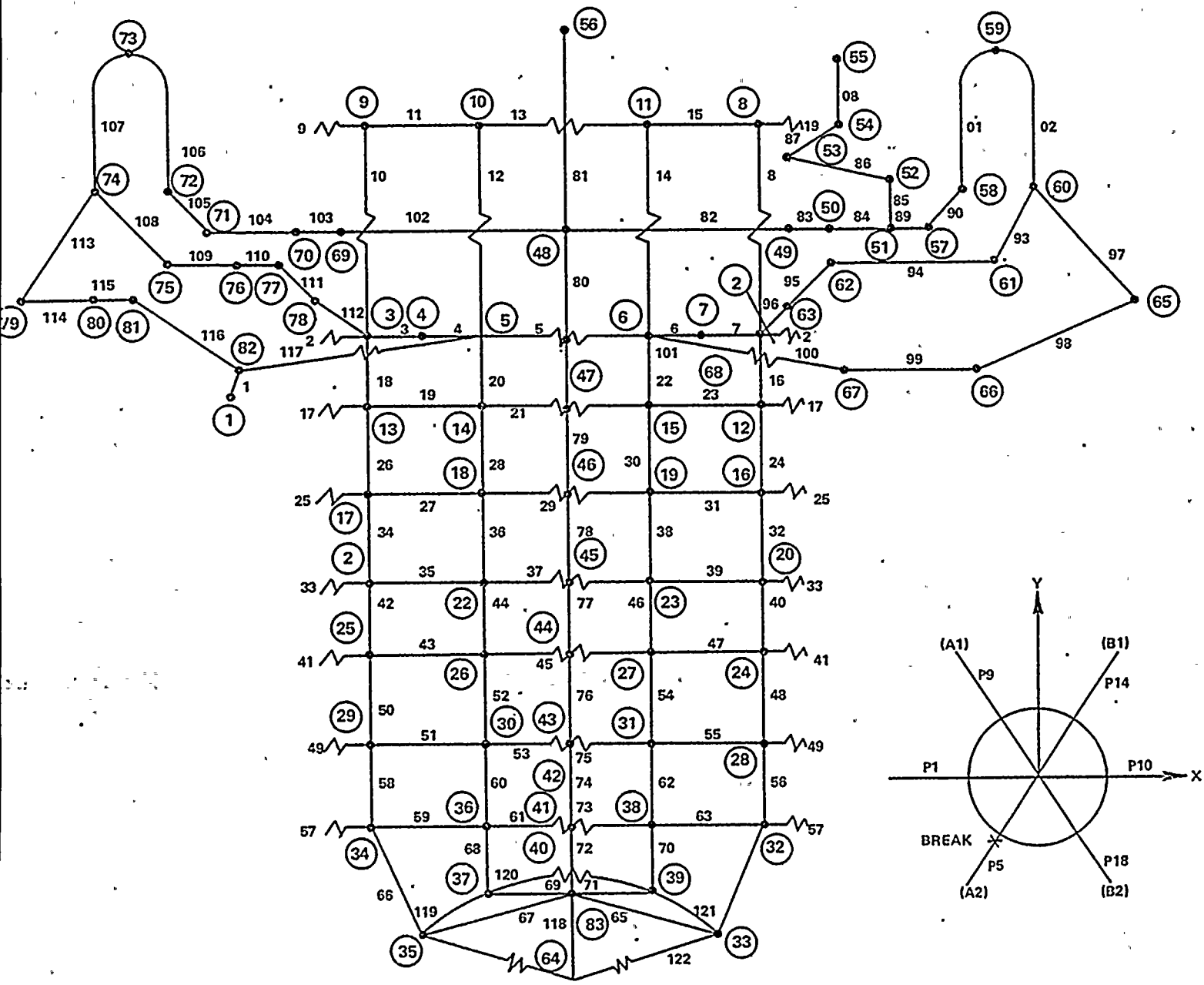
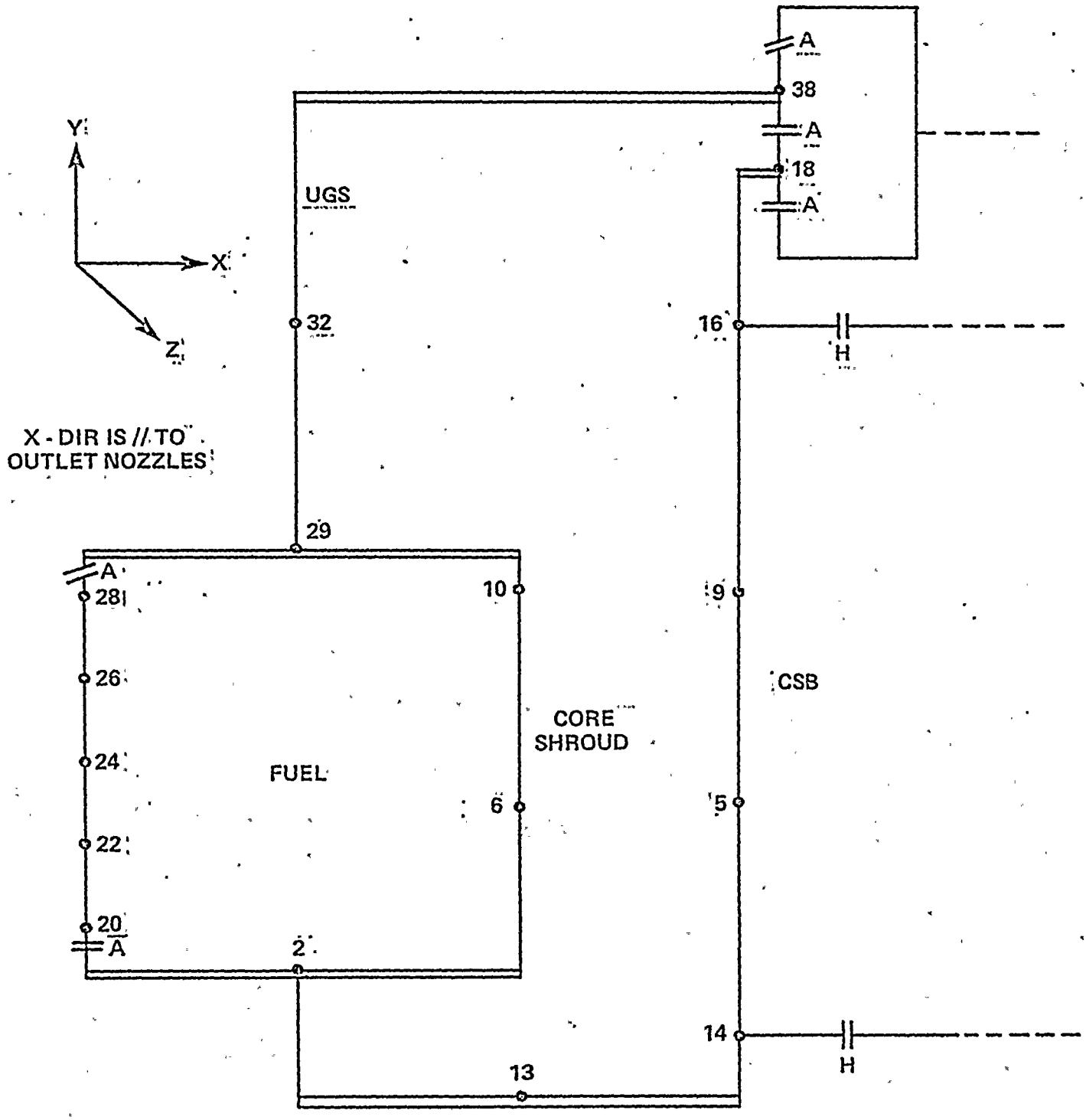


FIGURE 3A ST. LUCIE 1
REDUCED MODEL OF REACTOR INTERNALS



LEGEND

A = AXIAL GAP
H = HORIZONTAL GAP
// = PRELOADED COUPLING
= GAP COUPLING
= COLINEAR CONNECTOR

*SEE FIGURE 3b FOR DETAILS
OF REACTOR VESSELS AND PIPING

FIGURE 3B ST. LUCIE 1
REDUCED MODEL OF REACTOR COOLANT SYSTEM

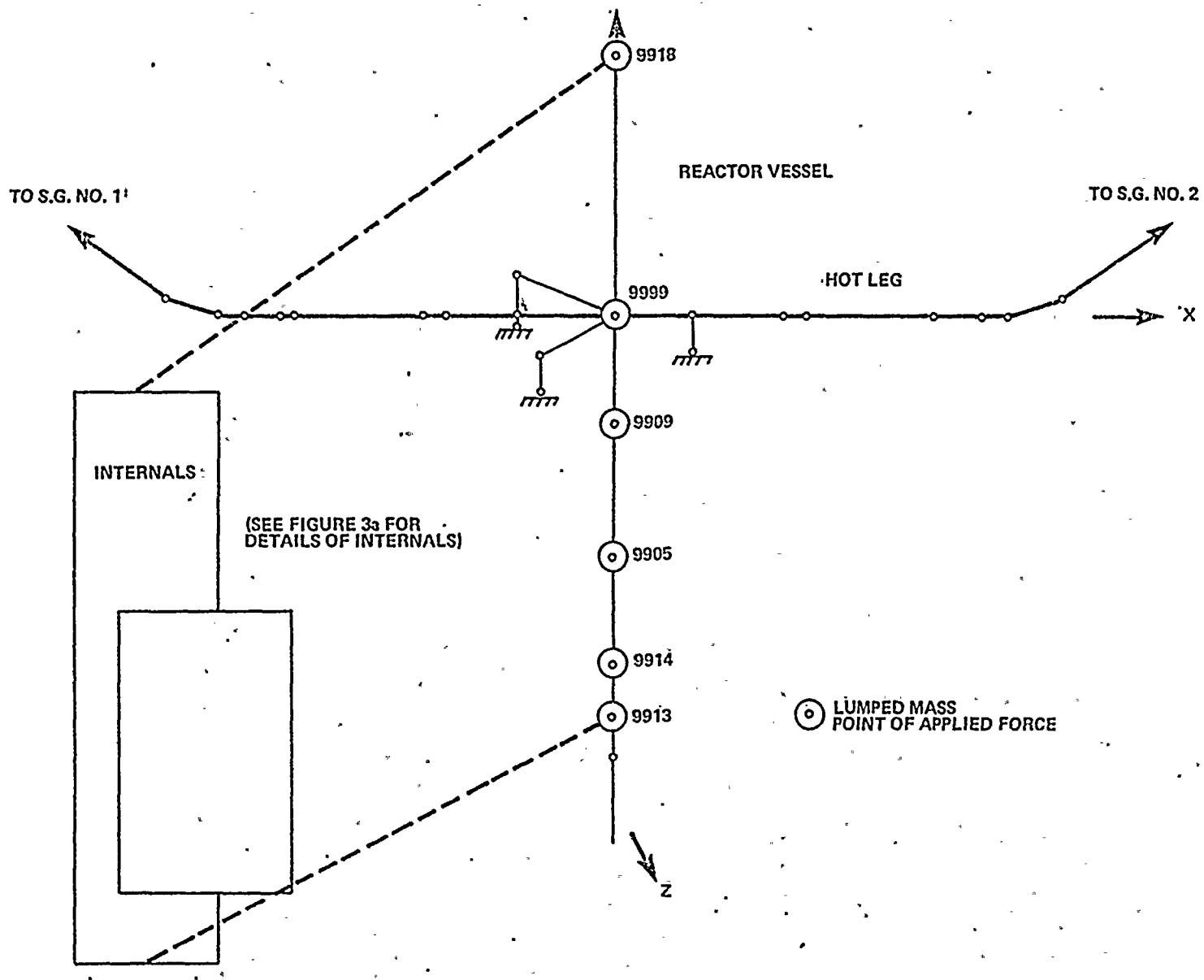
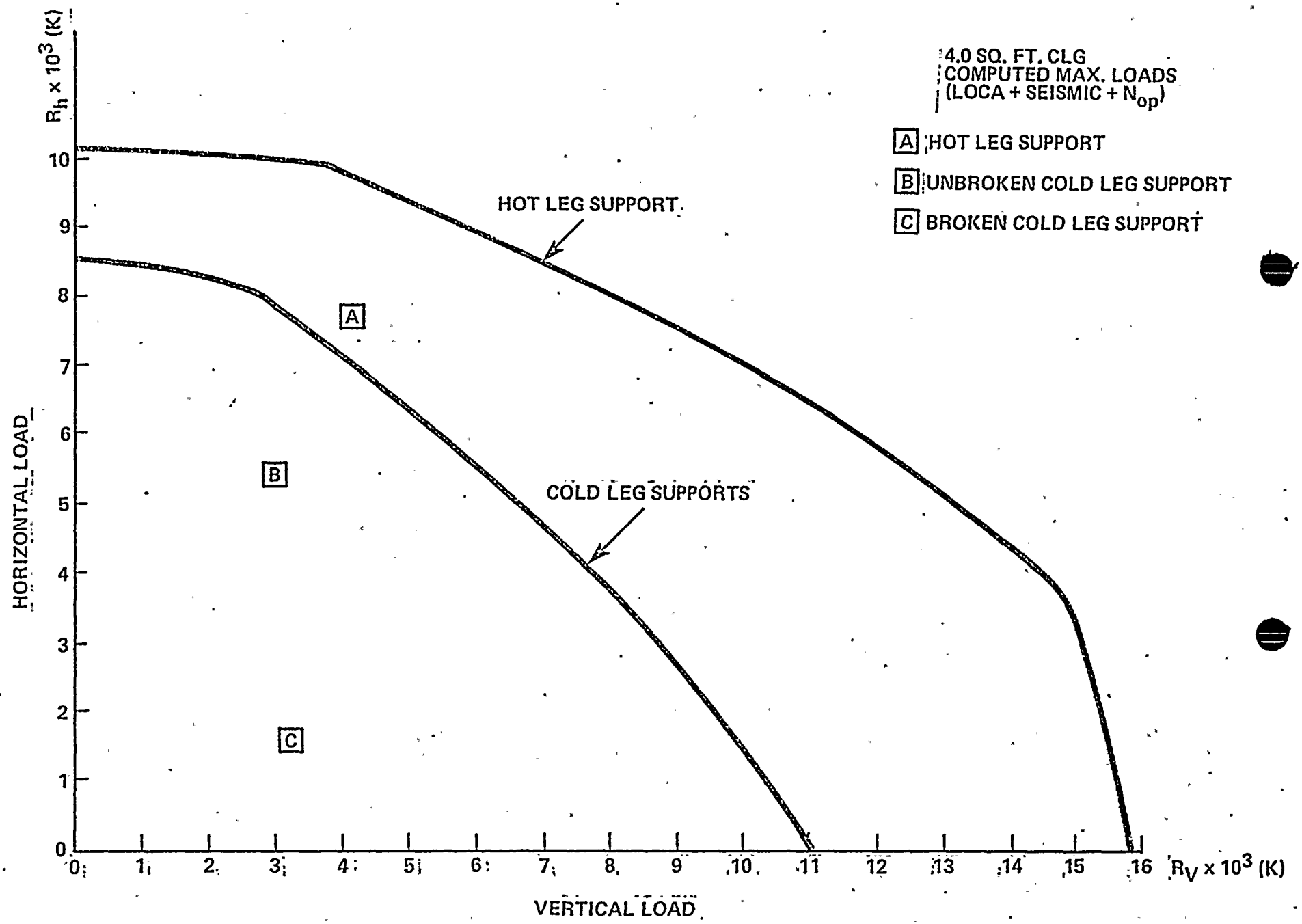


FIGURE 4 ST. LUCIE 1

REACTOR PRESSURE VESSEL SUPPORT PAD CAPABILITY



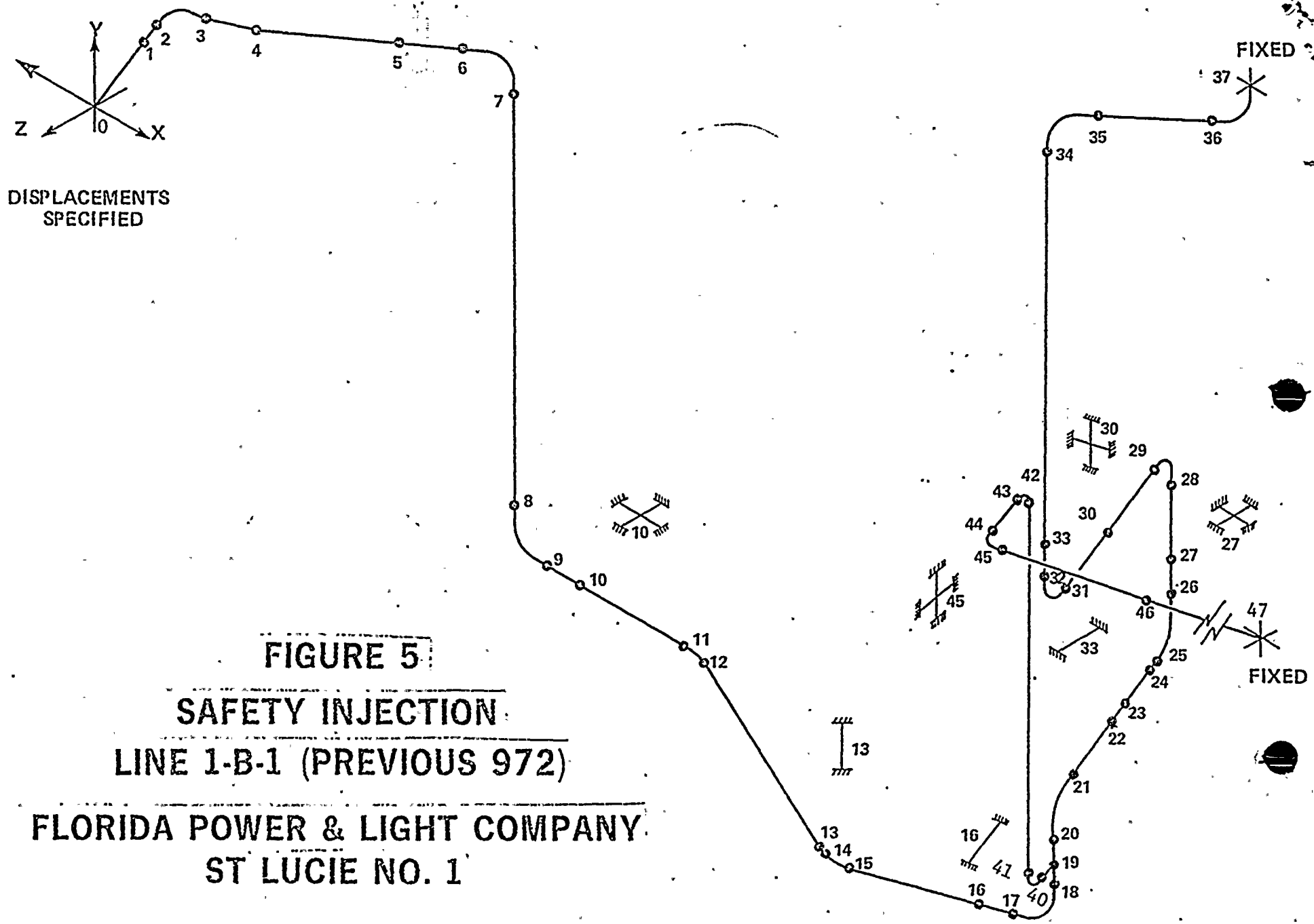


FIGURE 5

SAFETY INJECTION

LINE 1-B-1 (PREVIOUS 972)

FLORIDA POWER & LIGHT COMPANY

ST LUCIE NO. 1