VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

April 15, 1979

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
Attention: Mr. D. Vassallo
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 259
LQA:EAB/pwc

Docket No. 50-338

Dear Mr. Denton:

At the conclusion of our meeting on April 12th where representatives of Virginia Electric and Power Co. and Westinghouse Electric Corporation made technical presentations which demonstrated that the flow splitter plates on North Anna 1 are structually sound, you requested that we provide a written discussion based on a postulated failure of a splitter plate and an accompanying Safety Evaluation assuming such a failure would occur. This information is attached. Also enclosed is the written description of the analytical and ultrasonic examinations which we presented.

Based on the results of metallurgical and ultrasonic examinations, we have concluded that failures observed on Unit 2 were a result of high cycle fatigue which occurred early in life, and that this has not occurred, and will not occur on Unit 1. This unit has operated for approximately one year, during which time the accumulated number of cycles is on the order of 5×10^9 cycles, well beyond the 10^6 cyc[illegible] where fatigue failure would have occurred. Nevertheless, to confirm the con[illegible]nued integrity of these plates the same ultrasonic examination will be conducted during the second refueling of North Anna 1.

We feel this information which has been reviewed and unanimously approved by the Station and System Nuclear Safety and Operating Committees is responsive to your requests, and it further supports our position that North Anna 1 will operate safely and that it should be returned to service promptly. Your concurrence and prompt notification to Region II permitting power operation to resume will be very much appreciated.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President
Power Supply and
Production Operations

Attachment

cc: Mr. J. P. O'Reilly

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SAFETY EVALUATION

OF

REACTOR COOLANT

PUMP SUCTION

ELBOW SPLITTER

NORTH ANNA UNIT 1

April 15, 1979

INDEX

SAFETY EVALUATION

OF

REACTOR COOLANT PUMP SUCTION

ELBOW SPLITTER

1. DESCRIPTION OF ISSUE

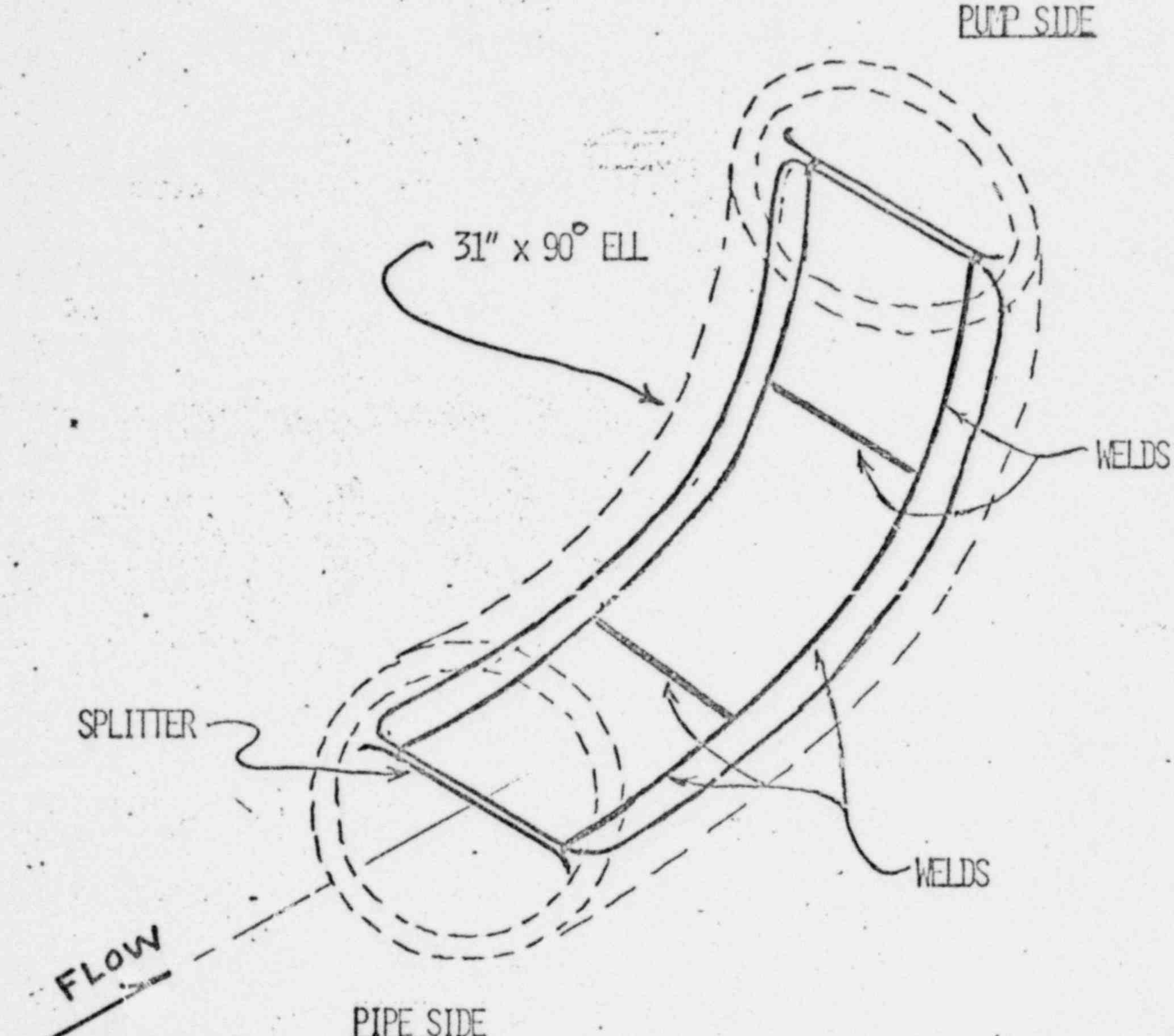
The purpose of this report is to evaluate the impact on plant safety and performance resulting from the postulated failure of the reactor coolant pump suction elbow splitter. The NRC has requested that this evaluation be performed as the result of an isolated splitter failure that occurred at North Anna Unit 2 Nuclear Plant some time during plant pre-start up testing.

The splitter element for which this evaluation is being performed is installed in a 31" x 90° elbow. (See Figure 1). The splitter element is fabricated and installed in three sections. The splitter element is full penetration welded along the axial length of the fitting and full penetration welded between the three sections. The splitter material is ASTM A-240 TP 304 cold rolled plate 1-1/16 inches in thickness. The elbow material is SA351 CF8M. The splitter/elbow weld is located approximately 1-1/2" from the the inside diameter of the pressure boundary.

For this report various sizes of failed plate material will be selected and evaluated to determine their impact on plant and equipment performance. The smallest size particle that could result from this postulated failure in quantities sufficient to affect plant/equipment performance will be determined by reviewing the fracture characteristics of the failed splitter. Intermediate size portions of a failed plate that could possibly pass through the impeller will be determined by a study of the pump geometry and interactions of the plate with the pump internals. The effect of large portions of failed plate that could lodge at the impeller inlet will also be considered.

Also included as attachments 2 and 3 are two additional analyses done by Westinghouse as support to the arguments presented in this report.

FIGURE 1
SPLITTER



2. FRAGMENT SIZE DETERMINATION

For this evaluation the largest size fragment to be considered will be that which would not pass through the Reactor Coolant Pump but would lodge at the pump inlet.

Intermediate size fragments will be determined by considering the largest size particle that would pass through the reactor coolant pump and eventually find its way into the Reactor Vessel. The possible size of this fragment will be discussed in a latter section of this report which describes the interaction of the pump internals with a failed splitter plate.

The smallest particle to be considered was determined by performing a detailed examination of the failed splitter plate. The nature of the fracture surfaces indicate that the most probable size of the smallest fragment could be defined by a 1-1/16" cube. Smaller size fragments are possible but these would be limited in number and their effects in the primary system would be inconsequential.

3. EQUIPMENT CONSIDERATIONS

(a) Splitter Elbow

The pressure retaining integrity of the elbow is not affected by the postulated splitter failure. The splitter is welded to an integrally cast transition area which is 1-1/2" removed from the elbow wall. The elbow as-cast has a minimum calculated wall thickness $t_m = 2.88"$ based on the ANSI B31.1.0 minimum design stress $S_m = 14,950$ psi for SA 351 TP CF8M material.

The applicable code for the elbow is ANSI B31.7 - 1969 which permits a design stress $S_m = 18,700$ psi resulting in a minimum calculated wall thickness $t_m = 2.25"$ using the code equation:

$$t_m = \frac{Pd}{2S - 1.2P}$$

This conservatism results in an excess calculated $t_m = 0.63$ inches.

Metallurgical studies performed on the cracked splitter samples conclusively show crack initiation in the weld area only. Due to high cycle fatigue, the crack propagated along the weld line and then due to the characteristics of the geometry and vibratory mode of failure, propagated towards the center of the plate away from the fitting wall.

In order to obtain an estimate of the integrity of the pipe wall impacted by a piece of the splitter plate, the gross assumption was made that the dislodged piece would be 31" x 19" x 1-1/16", weighing 175 pounds. For simplicity, the piece was assumed to be a 31" cylinder, 5" in diameter. The calculated impact force is 5,000 lbs, significantly less than the punching shear resistance of the elbow material, which is greater than 500,000 lbs.

(b) Reactor Coolant Pump

Consideration of hydraulic passage cross sectional areas and shapes in the pump impeller and diffuser lead to the conclusion that the largest piece of 1-1/16" plate which could conceivably pass through the pump could be no larger than about 9" by 9" square.

However, it's very unlikely that a piece anywhere near this size would pass cleanly from the impeller outlet to the diffuser inlet given the relatively high tangential velocity ($\approx$180 fps) of the impeller relative to the diffuser. It is estimated that objects with dimensions greater than 3" in the radial flow direction are likely to be pinched or sheared between the impeller and diffuser vanes. The extent to which this may damage the pump is discussed below.

Small Pieces

There is a 1.5" radial clearance between the impeller outlet and the diffuser inlet, so small objects (less than 1.5" on a side) will tend to pass through the pump hydraulics without pinching between the rotating and stationary parts and will at worst only locally dent the impeller and diffuser vanes as they bounce through. This would not significantly effect pump performance and is of negligible concern from the standpoint of the RCP.

Large Pieces

Of greatest concern, with regard to the RCP, is the largest piece which could pass through the impeller and be pinched or sheared between the impeller and diffuser vanes. A piece too large to enter the impeller or which some how becomes lodged in the impeller is of lesser concern since shaft vibration increase and a possible reduction in loop flow will serve to alert the operators and allow a pump shutdown before pump damage of a more serious nature, such as shaft failure, can occur.

The largest piece (9" x 9" x 1-1/6") which could reach the impeller exit would most likely cause very severe impeller and diffuser vane damage as it encountered the stationary diffuser vanes at approximately 180 fps before it cleared the impeller. Because of the relatively large diameter of the impeller this would cause a very large transient retarding torque to be applied to the pump rotating assembly as the piece was sheared or otherwise deformed by the impeller. Under such a torsional loading condition the weak link in the rotating assembly, by a factor of more than two, is the impeller key. A stainless steel piece with a shear area of 5 to 10 inches squared jamming at the impeller exit - diffuser entrance could generate enough torque to seriously deform and possibly fail the impeller key. Key deformation would cause a significant increase in pump vibration levels as the impeller shifted off its rotational center. Key failure would, of course, cause an immediate loss of loop flow. In either case, the broken or damaged pump parts would be expected to remain within the pump, and shaft seal failure would not be expected. Diffuser damage in all conceivable cases would be restricted to the locality of the inlet vane edges, and most likely be limited to deformation rather than fracture. No gross failure of the diffuser structure would be expected. Likewise, the impeller damage would most probably be deformation, possibly severe, but not fracture.

In all cases lateral vibration or bending of the shaft due to impacting of objects or imbalance is not considered to be a short term problem and would not be expected to cause shaft, bearing, or seal failure so long as operating time under such conditions was limited. This means that pump shaft vibration levels should be continuously monitored, and the pump should be shutdown immediately upon the detection of abnormally high shaft vibration levels.

(c) Reactor Vessel

General

Calculations were performed to estimate the velocity and kinetic energy of a 304SS object 9" x 9" x 1-1/6" in size at various positions in the reactor vessel. The total vessel flow rate assumed was 315,600 gpm. (This corresponds to the mechanical design flow rate for North Anna). The velocity in the cold leg nozzle is approximately 57 ft/sec and the kinetic energy of the object would be 1276 ft-lb_f. This corresponds to the energy imparted by the object to the core barrel. The object would then pass down the annulus between the thermal shield and reactor vessel. The velocity in this region is approximately 36 ft/sec and the kinetic energy of the object would be 506 ft-lb_f. The above corresponds to the velocity and kinetic energy of the object as it enters the lower plenum of the reactor vessel. In the lower plenum of the reactor vessel the flow velocity decreases. A velocity of 5.5 ft/sec would be sufficient to lift the object through the lower core support plate. The object would stop at the underside of the lower core plate. It should be noted that depending on the orientation of the object in the vessel lower plenum the object might come to rest at the bottom of the vessel. See Fig. 2, 3 & 4.

North Anna has a loose parts monitoring system. This system is capable of sensing objects with a kinetic energy of 0.5 ft-lb_f. Assuming that an energy of 1 ft-lb_f is impacted by a 304SS object traveling with a velocity between 20 and 35 ft/sec the volume of the object could not exceed 0.55 in^3, or approximately .15 lb. in weight. This implies that the loose parts monitoring system would be more than adequate to detect any object of appreciable size. A complete description of the loose parts monitoring system is provided in Attachment 4.

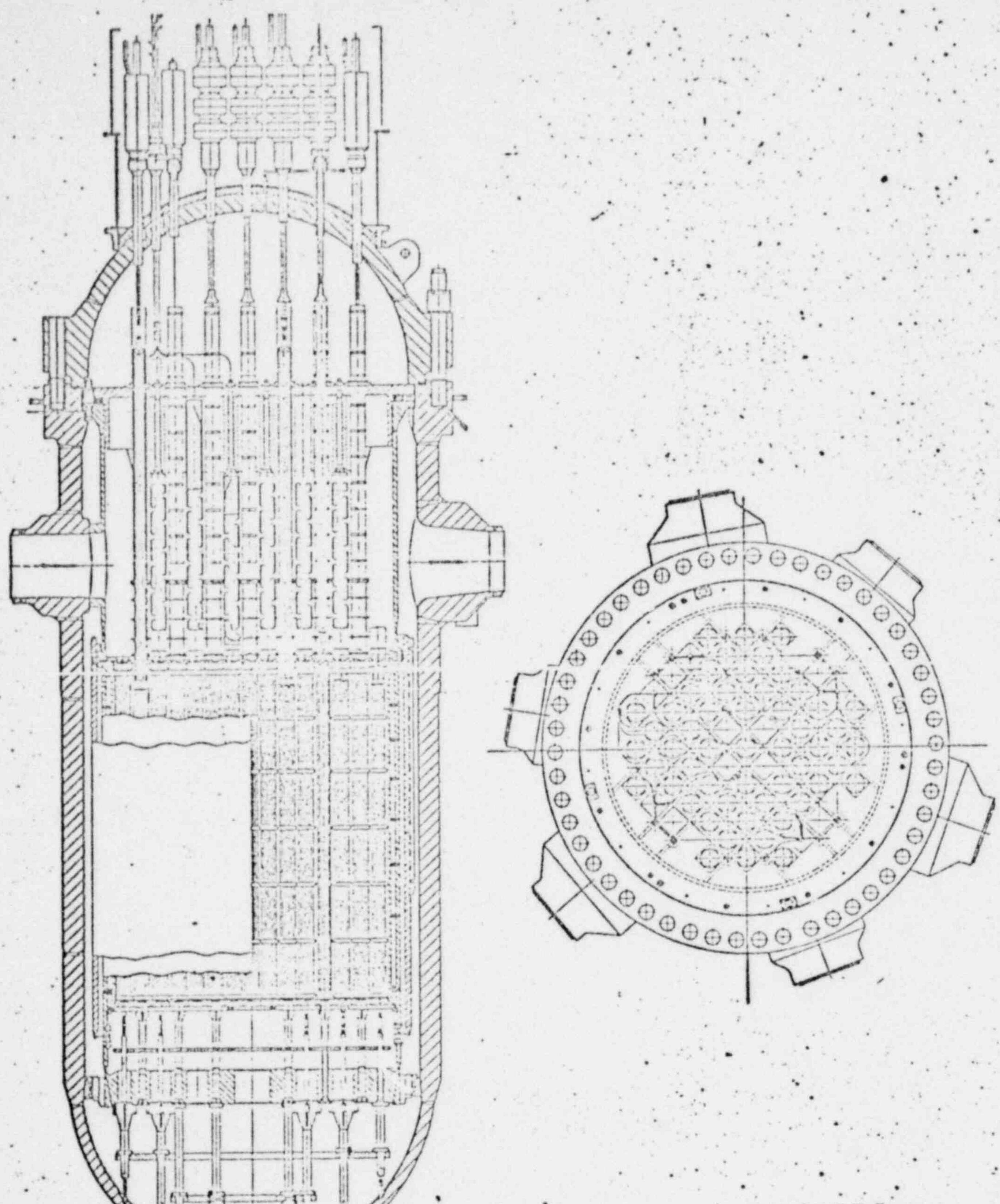
Internals

The following presents a summary of the analysis and results obtained by an analytical review of the effect on the reactor vessel and internals due to loose parts resulting from pieces of a flow splitter entering the reactor vessel at the cold leg.

The items reviewed relative to the internals and vessel fall into two main categories.

- Loose parts impacting on the lower internals structure.
- Loose parts lodged in areas of clearance and becoming wedged during periods of relative thermal growth.

FIGURE 2



REACTOR VESSEL GENERAL ASSEMBLY

FIGURE 3

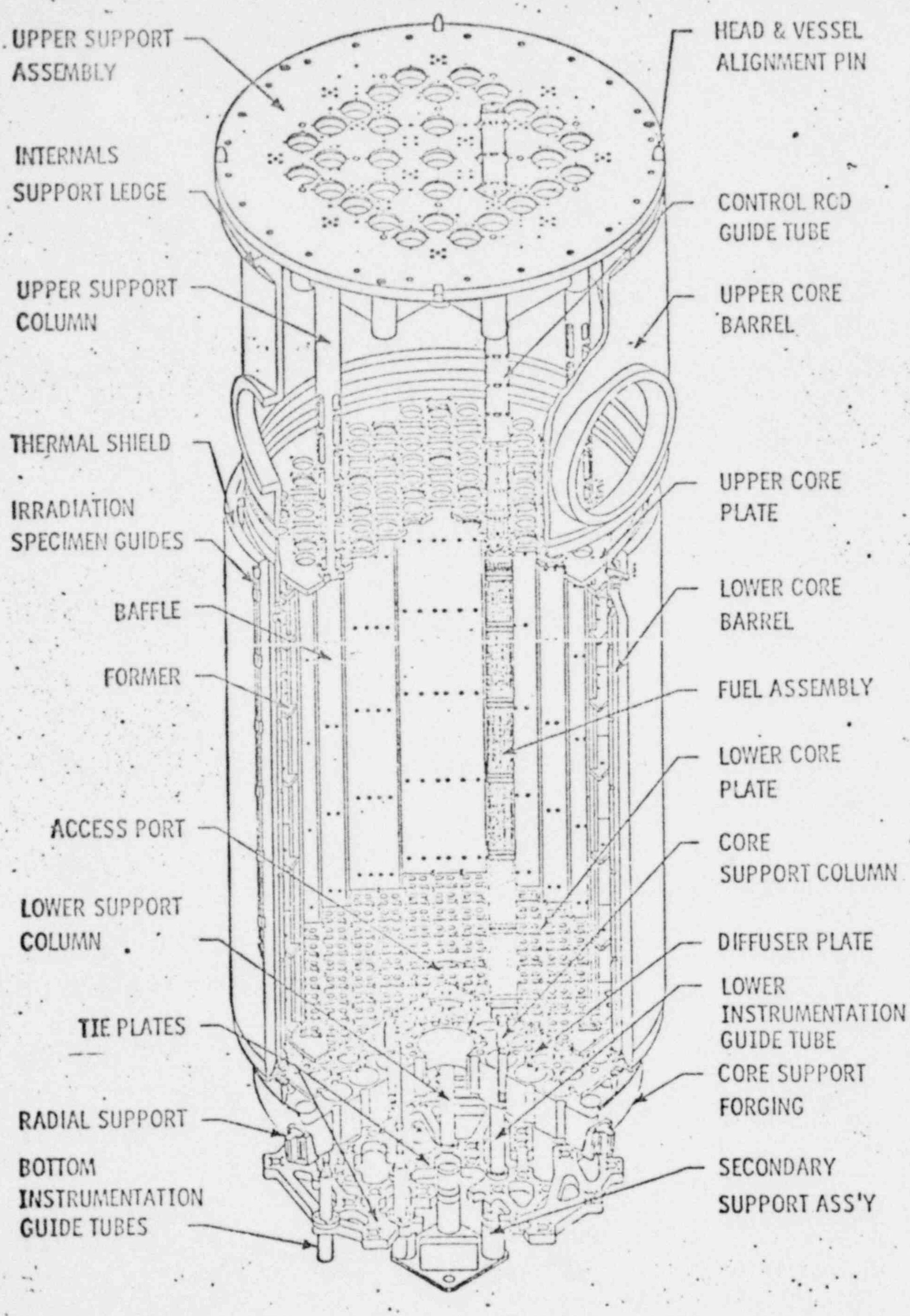
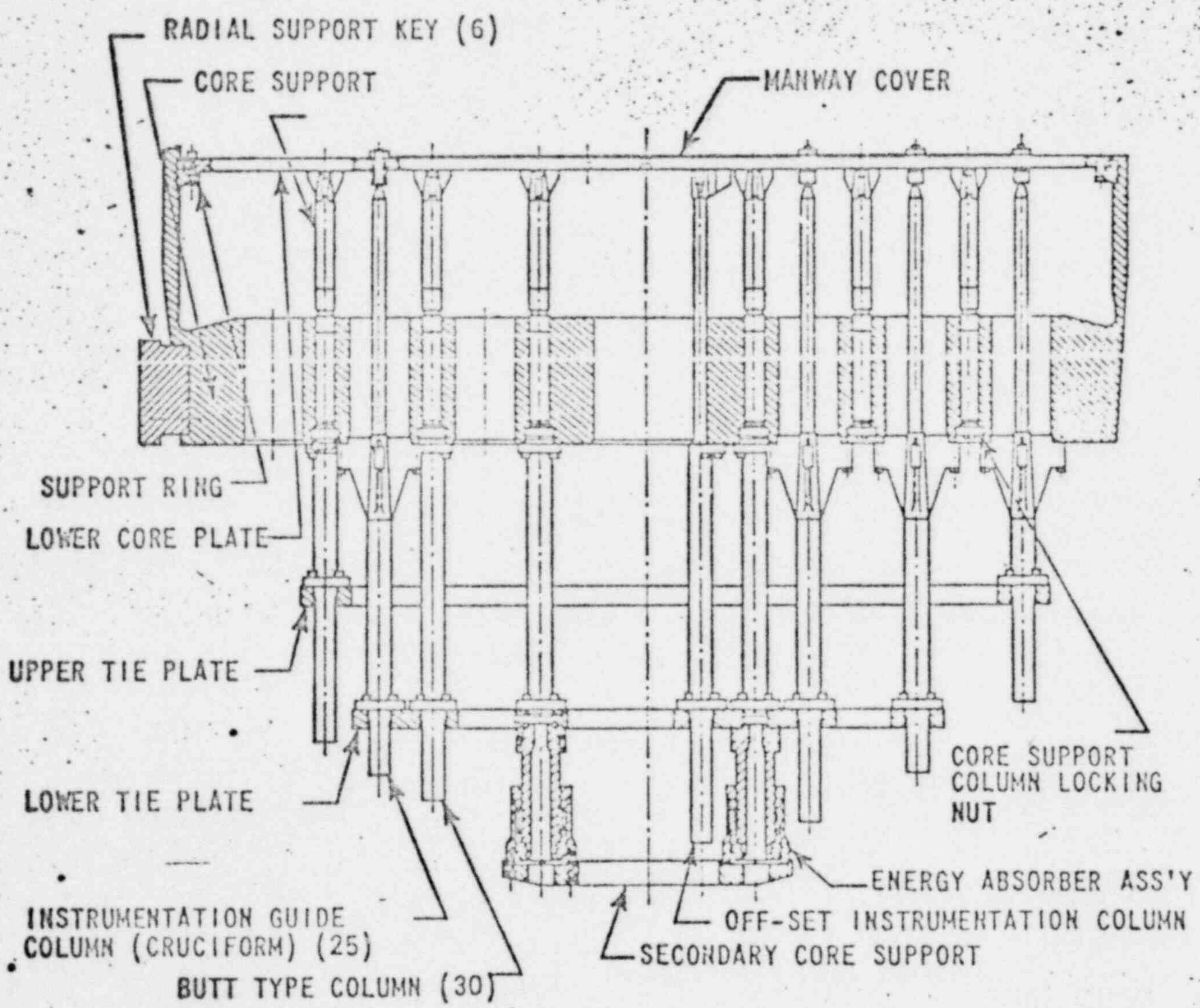

UPPER SUPPORT ASSEMBLY
INTERNALS SUPPORT LEDGE
UPPER SUPPORT COLUMN
THERMAL SHIELD
IRRADIATION SPECIMEN GUIDES
BAFFLE
FORMER
ACCESS PORT
LOWER SUPPORT COLUMN
TIE PLATES
RADIAL SUPPORT
BOTTOM INSTRUMENTATION GUIDE TUBES
HEAD & VESSEL ALIGNMENT PIN
CONTROL ROD GUIDE TUBE
UPPER CORE BARREL
UPPER CORE PLATE
LOWER CORE BARREL
FUEL ASSEMBLY
LOWER CORE PLATE
CORE SUPPORT COLUMN
DIFFUSER PLATE
LOWER INSTRUMENTATION GUIDE TUBE
CORE SUPPORT FORGING
SECONDARY SUPPORT ASS'Y

FIGURE 4



RADIAL SUPPORT KEY (6)
CORE SUPPORT
MANWAY COVER
SUPPORT RING
LOWER CORE PLATE
UPPER TIE PLATE
LOWER TIE PLATE
CORE SUPPORT COLUMN LOCKING NUT
ENERGY ABSORBER ASS'Y
OFF-SET INSTRUMENTATION COLUMN
SECONDARY CORE SUPPORT
INSTRUMENTATION GUIDE COLUMN (CRUCIFORM) (25)
BUTT TYPE COLUMN (30)

Specific areas judged to be critical are as follows:

(1) Impact on core barrel by flow splitter piece.

(2) Piece of flow splitter lodged into one radial support key and clevis

(3) Piece of flow splitter impact on bottom mounted instrumentation tubing.

(4) Piece of flow splitter lodges between the energy absorber of the internals and the reactor vessel.

Results:

The resulting conclusions are as follows:

(1) Impact on core barrel by a piece of the flow splitter.

A piece whose size 9" x 9" x 1.1/16" will not shear through the core barrel, but causes the normal and upset stress limit in the ASME Code Sub-section NG to be exceeded. This would result in local deformation of the core barrel.

(2) Piece of flow splitter lodged into one radial support key and clevis.

A size of 1 1/16" x .50 x 9.0 could lodge into the clearance area between the internals and the reactor vessel clevis, at cold conditions and then wedge when the plant is heated up. The resulting load imposed on the core barrel and reactor vessel causes the pre-load at the flange to be lost for the duration that the piece is wedged, and could result in yielding of the internals hold down spring. The interference load and pressure stresses produced on the internals and vessel would be within code allowable limits.

(3) Piece of flow splitter impacts on the vessel bottom mounted instrumentation tubing.

The impact force caused by a piece whose size is 9" x 9" x 1.06" would severely damage the tubing and associated instrument, and cause stresses in excess of the ASME Code allowable values for normal and upset operation. Plastic analysis of the tube indicates gross deformation. The pressure boundary would not be breeched.

(4) Piece of flow splitter lodges between the energy absorber of the internals and the reactor vessel.

A piece with a contact area of one square inch lodged between the absorber and vessel will result in the reactor vessel stresses to remain within the code allowable limits when a load of 400,000 lb or less is applied. As shown in the study for the typical three loop plants with steam generator plugs in the reactor vessel the load required to yield the energy absorber is between 147,500 lb for one column to 590,000 lbs. for four columns.

Fuel Assembly

Coolant flow blockage can occur with an assumed splitter plate piece entering the lower internals. The blockage can hypothetically occur by simultaneously covering all four lower core plate flow holes located directly below a fuel assembly with a piece approximately 9 inches square. Blockage can also occur from a smaller piece entering one of the core plate flow holes. In both cases, the flow blockage causes local reductions in coolant flow. The effects of the coolant flow blockage in terms of maintaining rated core performance, have been determined. With the reactor operating at nominal full power conditions, and the fuel assembly inlet nozzle completely blocked, the effects of an increase i.. enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR of 1.30 (reference Attachment 1 from RESAR-3S). In reality, a local flow blockage is expected to promote turbulance and this would not affect DNBR at all.

A piece of debris larger than the plate thickness could be considered. Such debris would enter from the lower internals. The debris would then permanently lodge in the fuel assembly bottom nozzle plenum. The bottom nozzle flow holes, being considerably smaller than 1-1/16, would prevent the debris from further movement through the fuel assembly and reactivity control components. Thus, proper functioning of these reactor components, which includes the control rod assemblies, is maintained.

4. SAFETY/ACCIDENT ANALYSIS

The mechanical and thermo-hydraulic effects of a foreign piece upon the reactor coolant system, reactor coolant pump and the reactor core and core structures have been discussed in the other sections of this paper. These concluded that there were no significant hazards to those components presented by the hypothesized/presence of a foreign piece.

From the safety/accident analysis viewpoint, three analyses are discussed. These are: partial loss of forced reactor coolant flow; complete loss of forced reactor coolant flow; and single reactor coolant pump locked rotor.

Considering an assumption of a foreign piece of a size sufficiently large to not pass through the pump rotor, yet small enough to be lifted up through the elbow by hydraulic forces, a situation with potential effects on pump flow can be hypothesized.

An orientation, where a foreign object is held against the impeller inlet, would reduce pump efficiency and very likely result in unbalance and consequently an increased amplitude of vibration. This, in turn, would necessitate a shutdown of the pump and a subsequent examination of cause. The effect on loop flow likely would not be of sufficient magnitude to generate a reactor trip on low flow; however, in any event, the flow reduction would be less than the partial loss of flow event, the results of which are shown to be satisfactory in Section 15.2.5 of the North Anna FSAR.

The Unit 1 splitter plate evaluations concluded that all plates were structurally sound. The sudden non-mechanistic failure postulated by the NRC would be considered to constitute a single passive failure in one loop.

In that event, there would not be a related failure or loss of flow in the other loops. The total loss of forced reactor coolant flow would not result and therefore, this accident analysis is not applicable.

A third situation was theorized by the NRC, that of a locked rotor. It would require a foreign object of appropriate geometry to enter but not pass through the impeller and to then extend beyond the outer diameter of the impeller to a sufficient degree to impact the pump diffuser. Considering the values of rotation initeria, relative to the structural strength of the object configuration, it is expected that the object would be deformed sufficiently to clear the rotational obstruction with lesser damage occurring on the pump components, and a sudden and immediate locked rotor is not anticipated. In a worst case, an abbreviated coastdown would be anticipated. An abbreviated coastdown occurred on Surry Unit 1 at approximately 80% power as a result of a sheared pump shaft. The later results and evaluation of that event showed the core did not approach an unsafe condition and the results were much less severe than the evaluation of a locked rotor event, as reported in Section 15.4.4 of the North Anna FSAR. The effect of a foreign object interacting between the pump impeller and diffuser would be expected to be less than those shown by experience with the sheared shaft.

Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV program. Inspection of the DNB correlation (Section 4.4.2.3 and Reference [44]) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The THINC-IV Code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on subchannel basis, regardless of where the flow blockage occurs. In Reference [63], it is shown that for a fuel assembly similar to the Westinghouse design, THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reference reactor operating at the nominal full power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR of 1.30.

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the Westinghouse cores cause flow perturbations which are local to the blockage. For instance, A. Oktsubo[82], et al., show that the mean bundle velocity is approached asymptotically about 4 inches downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked. P. Basmer[83], et al., tested an open lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 inches for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence, the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. The reduction in local mass velocity is then the main parameter which affects the DNBR. If the standard plants were operating at full power and nominal steady state conditions as specified in Table 4.4-1, a reduction in local mass velocity greater than 56 percent would be required to reduce the DNBR from 1.74 to 1.30. The above mass veloc- effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR at all.

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently

Summary of Results of Analyses Performed
to Support an Evaluation of the Effects of a
Small Loose Part Lodged Between the
Reactor Vessel and the Bottom Plate of the
Secondary Core Support

MARCH, 1978

CONTENTS

1. Introduction

A loose part was detected in the bottom of a typical W PWR reactor vessel by a Metal Impact Monitor* late in 1976. Subsequent measurements using the MIM detectors indicated that the part weight was less than 1.5 pounds or approximately 2" x 2" x 1-1/16" thick. As part of their evaluation of possible effects of this loose part, Westinghouse was requested to perform preliminary calculations to determine the effects of the loose part, assuming that it became lodged between the reactor vessel and the bottom plate of the energy absorber. Additional assumptions to be made were that 1) owing to the weight estimated from the MIM data, the part could be assumed to be under one of the four posts of the energy absorber, and 2) since the identity of the part has not been established, preliminary calculations should include those assuming that the part was rigid. Other conservative assumptions such as the assumption that the part perfectly conformed to the surfaces of the vessel clad and lower plate, and that these surfaces remain elastic were included in the calculations.

The forces computed from the structural model were used to evaluate the net holddown force at the core barrel flange and the stresses in the reactor vessel and internals. Vessel stresses were computed with handbook formulas and include pressure, thermal and loose part induced stresses. The results were compared with allowable stresses for normal operation. Internals stresses were found to be within allowable values by comparison of the loads imposed by the loose part with loads considered in design calculations.

Initial calculations were done to scope the effects of the part becoming lodged during heat-up or power escalation of the plant. Calculations were also performed to determine a preliminary basis for heat-up (i.e., heat-up rate and frequency of ascertaining that the part had not become lodged). The later calculations were done with a more detailed structural model.

The analytical methods and results for the most important cases considered are discussed in subsequent sections of this report. The results were obtained using the analytical model described in the next section.

* Comparable to Vepco Loose Parts Monitor

2. Analytical Model Used for Force-Deflection Computations

A flexibility model of the reactor vessel and internals was developed to compute forces and displacements resulting from differential thermal growth between the reactor vessel and internals, hydraulic flow forces and weight with a loose part wedged between the bottom plate of the energy absorber and the bottom head of the reactor vessel. In addition to cross-sectional stiffnesses of the various sections of the reactor vessel and internals, the model includes estimates of the flexibilities introduced by local loading of the vessel, local loading of the internals (through one post of the energy absorber) and contact stiffnesses above and below the loose part (assuming a 5.2 square inch contact area).

Since the area of the part is small (based on metal impact monitor results) relative to the size of the botttom plate of the energy absorber, it was assumed that the part was lodged under one post of the absorber. On this basis, the stiffness of the energy absorber used in the model was comprised of the stiffness of one post (and the associated cylinders) and the stiffness resulting from bending the bottom plate of the energy absorber with the conservative assumption that the other three legs of the energy absorber were rigid.

When indicated by the force levels, an approximate elastic-plastic force strain curve was used to calculate energy absorber stiffness.

3. Calculation Technique

The analytical model was used to determine the forces and deformations resulting from a loose part becoming lodged between the bottom plate of the energy absorber and the lower head of the reactor vessel during heat-up or power escalation. For each case considered, a part exactly fitting the gap was assumed to become lodged just before the transient was begun. The reduction in the gap resulting from the change in temperature of the structure was estimated and the resulting deformation applied to the analytical model to determine the forces acting between the vessel and the internals through the loose part and to determine the net force between the lower surface of the core barrel flange and the adjacent ledge of the reactor vessel. If yielding of the energy absorber was indicated, this process was done iteratively. The force introduced by the loose part taken together with the weight, spring (fuel assembly and core barrel holddown springs) and hydraulic forces was used to determine net holddown force that would exist at the core barrel flange.

To compute structure temperatures during heat-up transients, the temperature distribution through the vessel wall was calculated from data in the literature for a linear temperature increase of one surface of a plate. Since the core barrel wall was found to be close to the fluid temperature for the transients considered, it was conservatively assumed to be at the fluid temperature.

4. Force and Deformation Results for Several Transients

The following assumptions were used in the calculations leading to the results for all cases listed below:

a. Loose part contact surface area of 5.2 square inches (each side)

b. A minimum value of core barrel holddown spring force

c. No yielding of the loose part or adjacent surfaces

d. The hydraulic forces were those for four pump flow at 70°F or 550°F.

The limiting criteria was that the minimum net force between the core barrel flange and vessel ledge is not less than 100,000 pounds.

Case 1: Part becomes lodged at the beginning of a heat-up transient that starts at 70°F.

1. The minimum holddown force of 100,000 pounds is reached when the differential thermal growth reaches 0.038 inches.

2. A temperature increase of 31°F will result in an 0.038 inch relative growth if the heat-up rate is sufficiently slow that all structures are at the coolant temperature.

3. A temperature increase of 20°F at a rate of 20°F/hour will result in a growth of 0.035 inches (i.e., the minimum hold-down force would be reached after 1 hour at 20°F/hour).

4. The force acting across the loose part after the transient listed in (3) and after all four pumps have been shut down, will be approximately 320,000 pounds (361,000 pounds, if the hydraulic force acting at the core barrel flange is conservatively assumed to act at the lower support plate).

Case 2: Part becomes lodged at the beginning of a heat-up transient that starts near the full temperature.

For this case the energy absorber yields before the minimum holddown deflection is reached. As an example of the approximate results, if a heat-up rate of 37°F/hour occurs for one hour, the maximum forces on the vessel will be 270,000 pounds and the energy absorber deflection will be approximately 0.030 inches.

Case 3: Part becomes lodged just after the escalation from zero power to full power is begun (assuming original design conditions);

1. Relative thermal growth (assuming no thermal lags) of 0.053 inches occurs.

2. The system forces across the loose part are calculated to be just above the yield point of the energy absorber so that a force of approximately 220,000 pounds will exist across the loose part with four pumps in operation.

5. Reactor Vessel Stresses

Using handbook formulas, it was determined that the combined primary, secondary and shear stress allowable values for the reactor vessel are met when a load of 400,000 pounds is applied by a loose part over a one square inch area (larger allowable loads result if a larger area is used). The analysis is based on the load over this small area causing a local membrane stress and the membrane and shear stresses across the vessel wall being secondary or self-relieving. Therefore, allowable stress limits of 40,000 psi (1.5 Sm) were used for the membrane stress intensity and 80,000 psi (3.0 Sm) were used for the stress intensity developed from the combined primary, secondary and shear stresses.

The initial high bearing stress from this displacement (deformation) controlled load over this small area was considered a local hertz contact stress and would be reduced to allowable code limits after a local deformation of the vessel clad on the order of 0.040 inches. If the code allowable bearing stress limit of 1.5Sy (61,000 psi) must be met, the required contact for the load must be approximately 6.5 square inches for a 400,000 pound load.

6. Discussion

With the assumptions used in the calculations, a heat-up rate of 20°F/hour with checks for loose part freedom once an hour can be used without reducing the core barrel holddown force to less than 100,000 pounds.

The results also indicate that higher heat-up rates and/or longer intervals between checks for looseness are possible if material properties and hydraulic forces at intermediate temperatures are used, if part of the heat-up is done with less than four pumps in operation, or if flexibility of the loose part is included. The higher heating rates will cause increased plastic deformation of the energy absorber and somewhat higher forces on the vessel.

The differential thermal expansion that occurs during power escalation results in forces at the loose part that are slightly higher than the force required to yield the energy absorber. Oscillatory stresses in the vessel and internals during normal operation with the part lodged and changes in stresses due to seismic and loss of coolant events, have not been evaluated.

S. G. TUBE PLUGS IN TYPICAL 3-LOOP VESSEL

Introduction:

There are two possible consequences that could arise from the existence of steam generator tube plugs in the reactor vessel:

1. Loose pieces impacting upon the lower internals components.

2. The plugs becoming wedged between clearances and causing high forces during periods of relative thermal growth.

These plugs are approximately 3/4 inch in diameter x 6 inches long. The first consequence was examined using known or estimated flow velocities in the lower vessel plenum and was judged to not be a serious problem, provided the plugs are not left in indefinitely. The plugs are too large to enter either the core region or the drive line area, and the chance of small pieces breaking off and migrating upward is judged to be remote.

For the second consequence, several areas were identified where the possibility of tube plugs becoming wedged was considered. Of those studied, only one area was subsequently judged to have any real probability of occurring - and that is the area at the bottom of the vessel, where a clearance exists between the vessel and the secondary core support base plate (see Figure 1).

A close clearance exists between the underside of the base plate, at its periphery, and the reactor vessel bottom head. This clearance is 1.00 inch (nom.) cold. At the end of normal heatup, this gap has closed to 0.375 inch, and eventually stabilizes at 0.500 inch during steady state operation. Therefore, if steam generator tube plugs (approximately 0.72 inch at solid end) were to become wedged between the base plate and vessel before or during heatup, the constriction against thermal growth would cause high forces to exist.

Results of Analysis:

A study was performed to determine:

1. The magnitude of forces produced by a number of wedged tube plugs,

2. The possible consequences of these forces upon the reactor vessel and internals.

A test was performed to determine the forces that would exist at given deflections. For this test, the solid end of a tube plug was compressed between two flat 304 SS plates in a load machine. ' e attached Figure 2 indicates that forces of 49,000 lbs. and 72,000 lbs. would exist at deflections of 0.250 inch and 0.375 inch, respectively. The 0.250 inch deflection represents the remaining vertical growth of the internals (relative to the vessel) once contact has been made with the tube plug, while 0.375 inch is for the end of heatup condition. Correcting for operating temperature reduces the above loads to 42,000 lbs. and 62,000 lbs., per wedged plug.

If more than one plug were wedged between the vessel and base plate, the load generated would increase accordingly. Thus, if in the worst case, all eleven tube plugs were wedged beneath the base plate, the maximum theoretical load that could be generated would be 682,000 lbs., based upon the results of the test.

To assess the possible consequences of these forces, the following areas were studied:

1. Stresses in vessel bottom head

2. Load capacity of secondary core support energy absorber

3. Load capacity of internals hold down spring

4. Longitudinal stress in vessel shell

5. Stresses in internals core support, core barrel and core support columns

Of these five areas, only the first three were found to be significantly affected.

Analysis of the secondary core support energy absorber assembly, indicates that assembly comprising four energy absorber columns will yield at forces in the range of 430,000 to 590,000 lbs., depending on the actual yield strength properties of the material used. Any one column would yield at between 107,500 and 147,500 lbs. Yielding of the energy absorber assembly would impair its ability to limit the force produced by a postulated core drop accident (core barrel failure) to within prescribed values. Yielding of a single energy absorber column (as a result of eccentric loading under the base plate), while not desirable, [illegible] v[illegible]wed to be as serious as the general yielding case.

The load capacity of the internals hold down spring was examined for two conditions - steady state operation (mechanical design flow) and the hot pump overspeed condition. During steady state operation, a contact force of 512,000 lbs. exists [illegible]ween the core barrel flange and the vessel ledge (Figure 3), while a contact force of 396,000 lbs. exists during hot pump overspeed (Figure 4). Current design practice is to consider 100,000 lbs. of the contact force as margin against uncertainties, which leaves 412,000 lbs. and 296,000 lbs. as reserve contact force during steady state operation and hot pump overspeed, respectively. Any force acting upward through the base plate (due to wedged tube plugs) would act to reduce the reserve contact force described above.

If this contact force were overcome by the upward force of the wedged steam generator tube plugs, the consequences to the internals could be serious. As contact is lost at the vessel core support ledge, flow through the resulting gap would tend to equalize the pressure above and below the core barrel flange. This in turn might cause the lower internals to slam down upon the ledge, where the process could repeat.

Conclusions:

The loads discussed above are summarized in the following table.

Component	Allowable Load
1. Vessel bottom head	450,000 lb. to 500,000 lb.
2. Energy absorber assembly	430,000 lb. to 590,000 lb.
3. Hold down contact force	296,000 lb. to 412,000 lb.

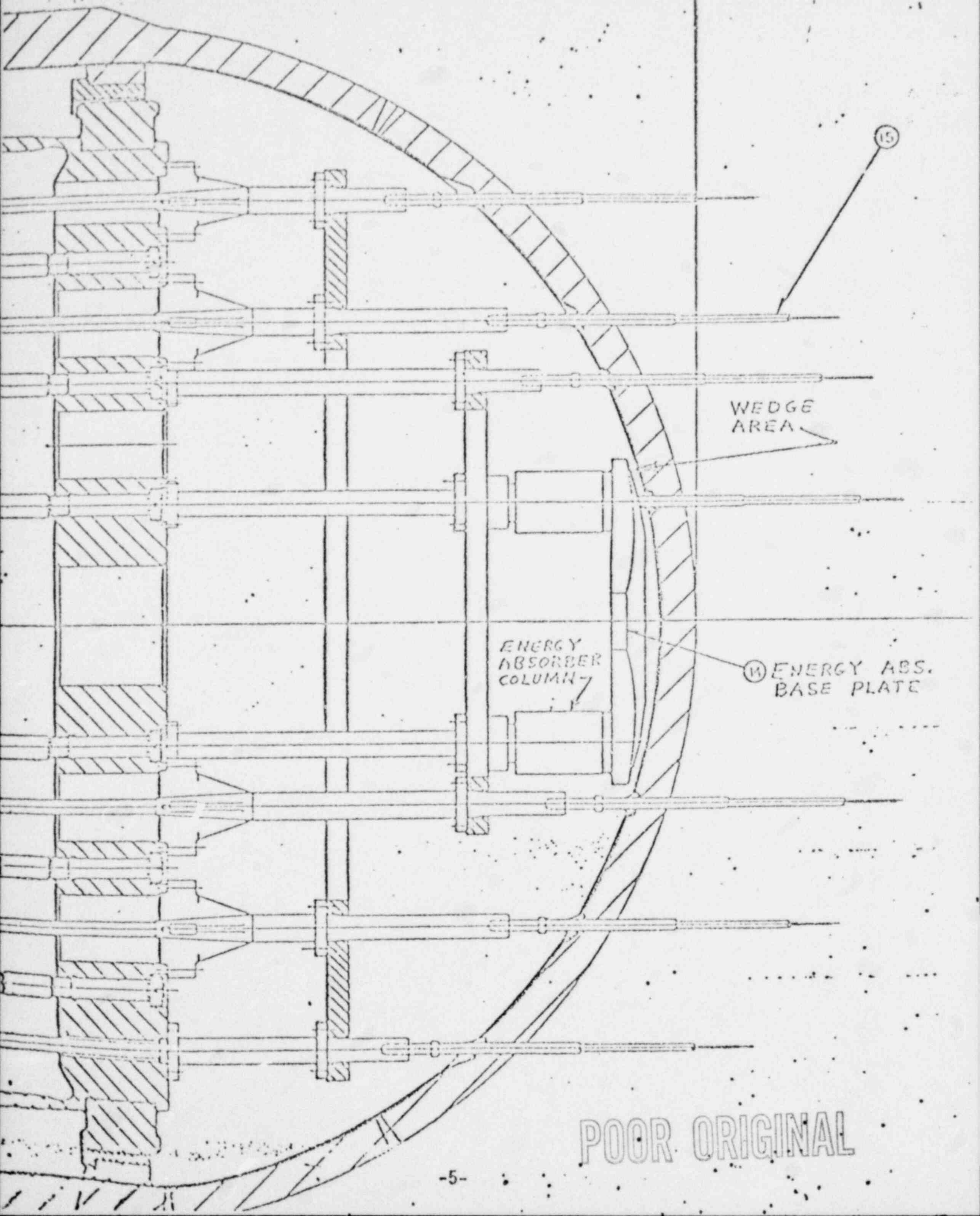
Thus, during steady state operation, the limiting load that can be tolerated is 412,000 lbs. If the hot pump overspeed condition is considered, the limiting load reduces to 296,000 lbs.

The forces generated by wedged tube plugs are summarized below.

Condition	Allowable Force	Allowable Number of Plugs
Steady state operation:	412,000 lb. ÷ 42,000 lb.	= 9.80 plugs
Hot pump overspeed:	296,000 lb. ÷ 42,000 lb.	= 7.04 plugs
End of normal heatup:	412,000 lb. ÷ 62,000 lb.	= 6.64 plugs
Heatup + pump overspeed:	296,000 lb. ÷ 62,000 lb.	= 4.77 plugs

From the above, it can be seen that if a hot pump overspeed condition is considered to occur at the end of a normal heatup, no more than four (4) wedged steam generator tube plugs can be tolerated beneath the energy absorber base plate. If the above-postulated transient is not considered viable, then the number of wedged tube plugs that can be tolerated increases to six (6).

FIGURE 1



15
WEDGE AREA
ENERGY ABSORBER COLUMN
14 ENERGY ABS. BASE PLATE

STEAM GEN TUBE PLUG - LOAD vs DEFLECTION — T=70°F

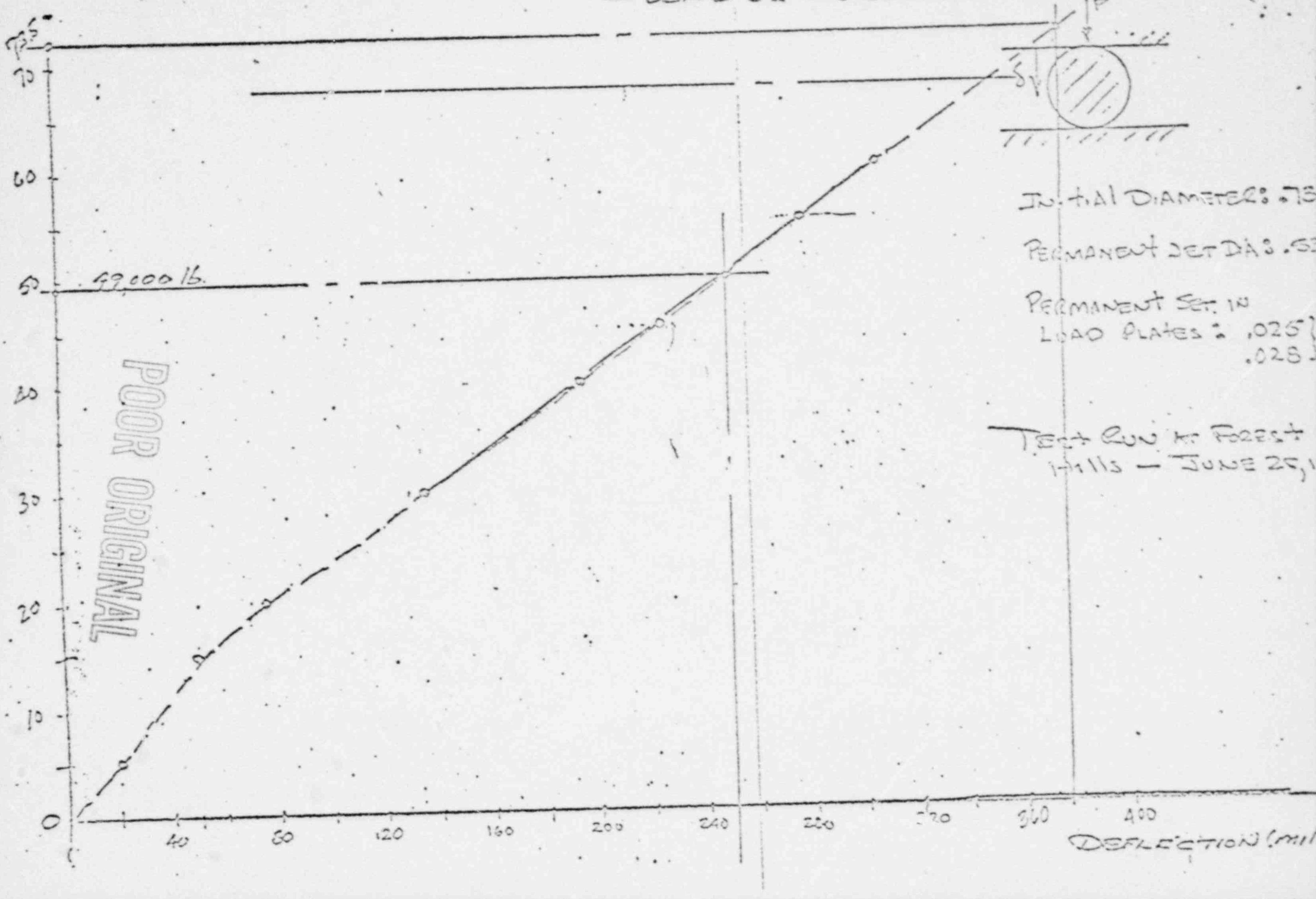


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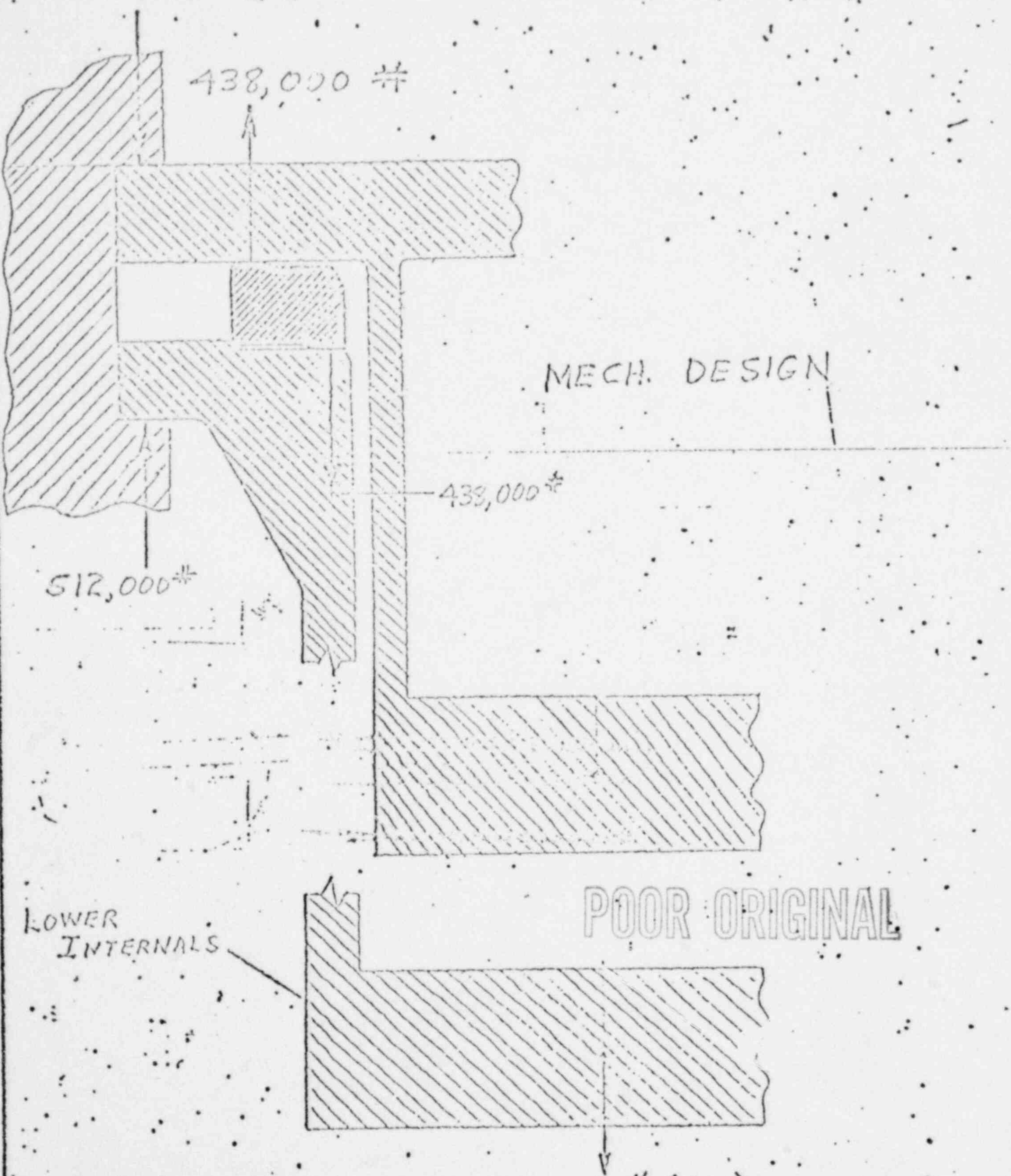
FIGURE 3



POOR ORIGINAL

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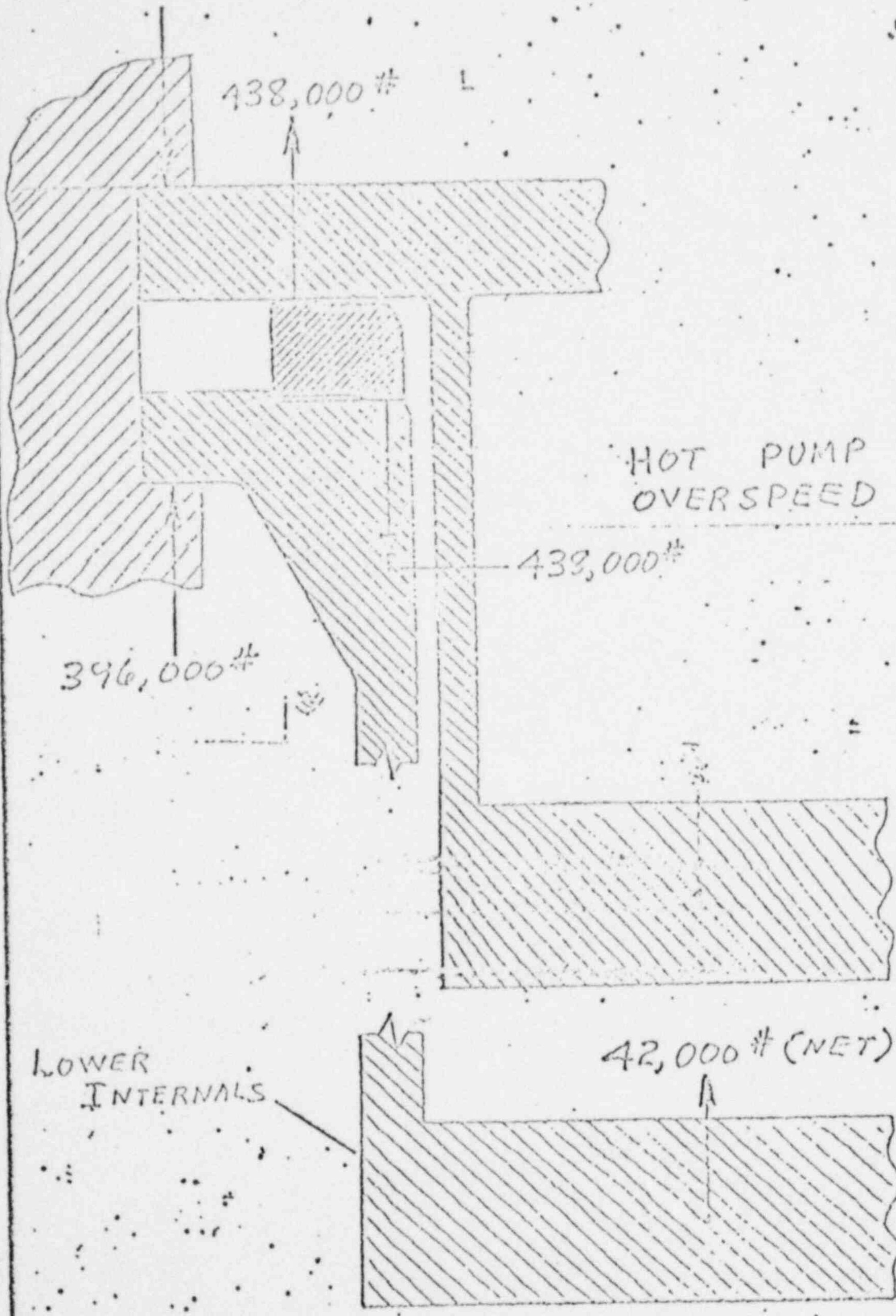
FIGURE 4



LOOSE PARTS MONITORING

The Loose Parts Monitoring System, installed in North Anna Units 1 & 2, consists of a total of 10 transducers per unit, five active and five passive (installed spares). There are 2 transducers on each Steam Generator, one on each manway on the lower section of the generator, with one active and one passive. There are 2 transducers in the reactor vessel flange area and 2 on the lower reactor vessel hemisphere. One transducer in each location is active.

The basic sensitivity of the transducers is .05 ft.-lb of impact energy at the transducer location. Of course, if the impact of a mass is not exactly at the transducer the signal will be attenuated as it travels through the materials in the coolant system and finally reaches the transducer. After the impact noise is detected, it is then transmitted to the control room where it is amplified for readout and alarming functions.

During initial installation and checkout of the system, the attenuated factors at various locations around the transducers is checked using a tool which imparts a known impact to the materials being tested. The data from the initial test makes it possible for the system to be used to evaluate any future signals as to size and location of a loose piece of material in the Reactor Coolant System.

The impact energy of an object is basically dependent on its velocity and mass. Therefore, calcuations were made to determine the minimum particle size that the Loose Parts System can detect in its present configuration. The following flow velocities were calculated using the minimum allowable Reactor Coolant flow rate of 92,800 G.P.M.

North Anna Units 1 & 2 Flow Velocities

Coolant Temp. - 547°F

Coolant Pressure - 2235 PSIG

Coolant Flow (GPM) - 92,800 GPM

R. C. Pump Discharge	- 27 1/2" I.D.	-	50.18 ft/sec
Reactor Outlet	- 29" I.D.	-	45.14 ft/sec
R. C. Pump Suction	- 31" I.D.	-	39.46 ft/sec

Calculations Done By Rockwell, Loose Parts Monitoring System Manufacturer.

Coolant Temp. - 547°F

Coolant Pressure - 2235 PSIG

Coolant Flow (GPM) - 93,800 GPM

Nominal Pipe Size - 30" I.D. - 42.5 ft/sec

The calculation of impact energy was based on a weight of 1 lb. traveling at 42.5 ft/sec hitting the core barrel. An object of this size will generate an impact equivalent to 28.3 ft-lb. Initial test data indicates that, based on transducer location and attenuated factors, the signal would be attenuated by a maximum of 10 lb. or by a factor of 3.16. The resultant signal at the transducer would therefore be $\frac{28.3 \text{ ft-lb}}{3.16}$ or 8.95 ft-lb. which is well above the alarm setting of 0.5ft-lb. Since the flow velocity in the coolant system is constant a correlation can be made to show the smallest mass that will cause an alarm.

$$\frac{1 \text{ lb.}}{8.95 \text{ ft-lb}} = \frac{x \text{ lb.}}{.5 \text{ ft-lb}}$$

x lb. = .055 lb. which is < 1oz.

1 oz. will impart .56 ft-lb. of energy; therefore, we feel that a particle of this weight could be detected and that it is feasible that it would travel at the same velocity as the coolant. Since the start-up test data for North Anna shows that a flow rate of approximately 105,000 GPM exists in the coolant loops, we feel even more confident in making this statement. Of course it is not realistic to assume that a mass of 1 lb. would travel at coolant flow velocities; however, it would only have to travel at 1/4 of those velocities to generate enough impact energy to cause alarms.

The Loose Parts Monitoring System functional capability can be verified during operation in two ways. First of all, transducer operation may be checked by comparing initial background noise profiles with present profiles using the vibration made of operation of the system. In general, background noise in a mechanical system tends to increase with time; therefore, a noticeable decrease in background noise would warrant further investigation into the system functional capability. On the other hand, a rapid increase in background noise level, even below alarm settings, would also be indicative of a monitoring system problem or possibly a mechanical component problem. In either case a periodic check of each channel should reveal such occurrences. The second method would be used to determine that the electronics were working properly using built in test signals.

By using frequent surveillance intervals, we believe that the Loose Parts Monitoring System can provide us with an early indication of metallic particles in the coolant system that are as small as 1 oz. in weight.

ANALYTICAL EVALUATIONS

I. INTRODUCTION

In this section will be presented results of analytical evaluations conducted to identify the cause of cracking observed in Unit 2 elbow C and analytical predictions of subsequent crack behavior had operation been continued. A brief summary of pertinent facts and analytical results follows Appropriate details are presented in Figures and Tables.

A. Observations

1. Pattern

The cracking observed in the splitter plate is characterized dominantly by two large cracks located at opposite ends of the plate (leading and trailing edges) extending approximately 19 inches along the plate (leading edge) and 15 inches along the plate (trailing edge). The two cracks are diagonally opposite one another in the plate. Both cracks started in the welds, follow the weld for about 10" and then hook in towards the center of the plate and towards each other.

2. Modes of Failure Investigated

The following causes of cracking were investigated.

a. Crack extension of an initial flaw by application of a large load
No large, abnormal (non-cyclic) load source could be identified nor could any evidence of a pre-existing flaw be found.

b. Stress corrosion cracking. No evidence of stress corrosion cracking was found.

c. Material properties. Results of material investigations revealed no abnormal or unacceptable material properties.

d. Welding practice. No abnormalities in welding practice were uncovered.

e. Fatigue. Fractographic analysis of the leading edge crack at several locations clearly established that cracking occurred by fatigue. Striation measurements were in the range of 1 - 5 micro-inches per cycle and yield an estimated range of cycles to failure of 110,000 to 530,000 cycles. These results support ruling out causes a, b, c and d, above and suggest a vibration source of loading stemming from either structural induced or flow-induced vibrations. Striation measurements indicate high stress levels ranging from 20 to 60 ksi. (See Table 1).

B. Sources of Fatigue Loading

1. Structural Induced Vibrations

Experimental vibrations measurements were assumed to apply to the North Anna Unit #2 loop C and vibration and stress analysis results on the splitter plate show that the cracking was not due to structural induced vibrations because of very low vibration stress amplitudes ($\pm$ 10 psi).

2. Flow Induced Vibrations

a. Natural frequency of flow splitter plate. The first two natural frequencies of the flow splitter plate were determined by finite element analysis to be 148.15 and 148.18 cycles/sec. resulting from the first two closely spaced modes. Mode shapes are shown in Figures 1 and 2.

b. Natural frequency of elbow. The natural frequency of the elbow is determined to be much less than that of the splitter plate. Estimates are 57 cycles/sec, first mode.

c. Forcing frequency due to pump induced vibrations. Experimental vibration measurements made show that the predominant frequencies related to pump operation are 20 Hz, 140 Hz and 273 Hz. The first frequency is associated with the shaft rotation at 1200 rpm. The second and third frequencies are associated with the blade (7) passing frequencies. The second frequency (140 Hz) suggests a possible resonance of the plate (148 Hz), however as mentioned above stress amplitudes were insufficient to cause the observed cracking.

d. Excitation frequency due to vortex shedding. Considerations of elbow and plate geometry and flow and fluid conditions result in excitation frequencies of 98 Hz, (lower bound) and 136 Hz, (upper bound), again indicating possible resonance of the splitter plate. Results of the modal analysis (Figures 3 and 4) show that (1) sufficient stress amplification (± 32 to ± 72 ksi) could occur to cause the observed cracking (See Table 2); (2) the stress distribution in the plate is consistent with the cracking pattern observed in elbow C, namely, the cracks start at either leading or trailing edges, follow the weld and then turn into the plate center (Figures 3 and 4); (3) after progression of cracking the natural frequency o. the plate would drop to 64 Hz and a reduction in stress amplification, eliminating further crack extension (See Figures 5 and 6). Figures 5 and 6 also show that the crack would turn towards the center of the plate.

C. Summary

The information presented herein shows that a postulated resonance in the flow splitter plate induced by fluid flow is a plausible explanation of the observed cracking in the splitter of Unit 2, and that this cracking would not be expected to progress. Furthermore, because of the low natural frequency of the elbow, and the geometry of the cracks (hooking into the plate), this mechanism is not expected to have any effect on the pipe wall. This mechanism also indicates that the cracking observed occurred over a period of 12 to 60 minutes, significantly less than the time of operation in Unit #2 (17 days) and considerably less th n the operating period of Unit #1, 1 year. Had cracking occurred in Unit #1, (1) it would have been detected, and (2) it would have quickly progressed and stopped due to the drop in natural frequency of the plate and thus eliminating the assumed source of loading.

Irrespective of the loading source, given the plate geometry and constraints, the maximum stresses can only occur in the diametral direction. Therefore the fracture mode cannot be other than the one observed in North Anna Unit #2, Loop C.

TITLE: TABLE 1 FATIGUE STRESS ESTIMATES

PAGE ___ OF ___

PROJECT	AUTHOR	DATE	CHK'D. BY	DATE	CHK'D. BY	DATE
NORTH ANNA UNITS 1/2	S.S. PALUSAMY	4/13/79	Y. S. Lee	4/14/79	N/A	

S.O.	CALC. NO.	FILE NO.	GROUP
VGBM-10098	N/A	N/A	NTD/MMT/SMC

$\frac{da}{dN}$	a_i	a_f	N (NO. OF CYCLES)	σ_{max}	σ_{min}	$\sigma_a = \frac{\sigma_{max}-\sigma_{min}}{2}$	R
1×10^{-6}	.030	.53	530,000	40	0.	±20	0
3×10^{-6}			180,000	51	0	±25.5	0
5×10^{-6}			110,000	57	0	±28.5	0
1×10^{-6}	.030	.53	530,000	44	31	±6.5	0.7
3×10^{-6}			180,000	56	39	±8.5	0.7
5×10^{-6}			110,000.	62	44	±9.0	0.7

TABLE 1 ESTIMATES OF MAXIMUM, MINIMUM AND ALTERNATING (σ_a) STRESSES DUE TO THREE DIFFERENT $\frac{da}{dN}$ VALUES

REV. NO.	REV. DATE	AUTHOR	DATE	CHK'D. BY	DATE	CHK'D. BY	DATE

TITLE			PAGE
TABLE 2 RESONANT STRESS ESTIMATES			OF
PROJECT: NORTH ANNA UNIT 1/2	AUTHOR: S. S. PALUSAMY DATE: 4/13/79	CHK'D. BY: Y. S. Lee DATE: 4/13/79	CHK'D. BY: N/A DATE:
S.O.: VGBM-10098	CALC. NO.: N/A	FILE NO.: N/A	GROUP: NTD / MMT / SME

VORTEX SHEDDING INDUCED RESONANCE

RESONANT DISPLACEMENT

$$Y = \pm\left[\frac{1}{8\pi^2}\right]\left[\frac{\rho d^3 l}{m_1 S^2}\right]\left[\frac{1}{2\zeta}\right]$$

WHERE

ρ =	FLUID DENSITY, $lb\ sec^2/in^4$	=	.000044
m_1 =	GENERALIZED MASS, $lb\ sec^2/in$	=	.465
d =	TIP THICKNESS, IN	=	1.063
l =	LENGTH OF TIP, IN	=	31
S =	STROUHAL No.	=	~0.2 TO 0.145
	[illegible]		
ζ =	FRACTION OF CRITICAL DAMPING	=	0.01, 0.02

$$Y = \pm\frac{1}{8\pi^2}\left[\frac{.000044 \times (1.063)^3 \times 31}{.465 \times (.2)^2}\right]\left[\frac{1}{2\times .01}\right], \quad \zeta = 0.01$$

$$Y = \pm .119$$

FINITE ELEMENT CALCULATIONS BASED ON UNIT LOAD ANALYSIS GIVE

$.002 = .6$ KSI

$$\text{FOR } \pm .119 = \frac{.6 \times .119}{.002} = \pm 36 \text{ KSI}$$

PEAK ALTERNATING STRESS AMPLITUDE IF THE STRESS CONCENTRATION IS ASSUMED TO 2 $\}$ = ± 72 KSI

REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

TITLE			PAGE
TABLE 2 (CONTINUED)			OF
PROJECT: NORTH ANNA UNIT 1/2	AUTHOR: S.S. PALUSAMY DATE: 4/13/79	CHK'D. BY: Y. S. Lee DATE: 4/13/79	CHK'D. BY: N/A DATE
S.O.: VGBM-10098	CALC. NO.: N/A	FILE NO.: N/A	GROUP: NTD/HMT/SMC

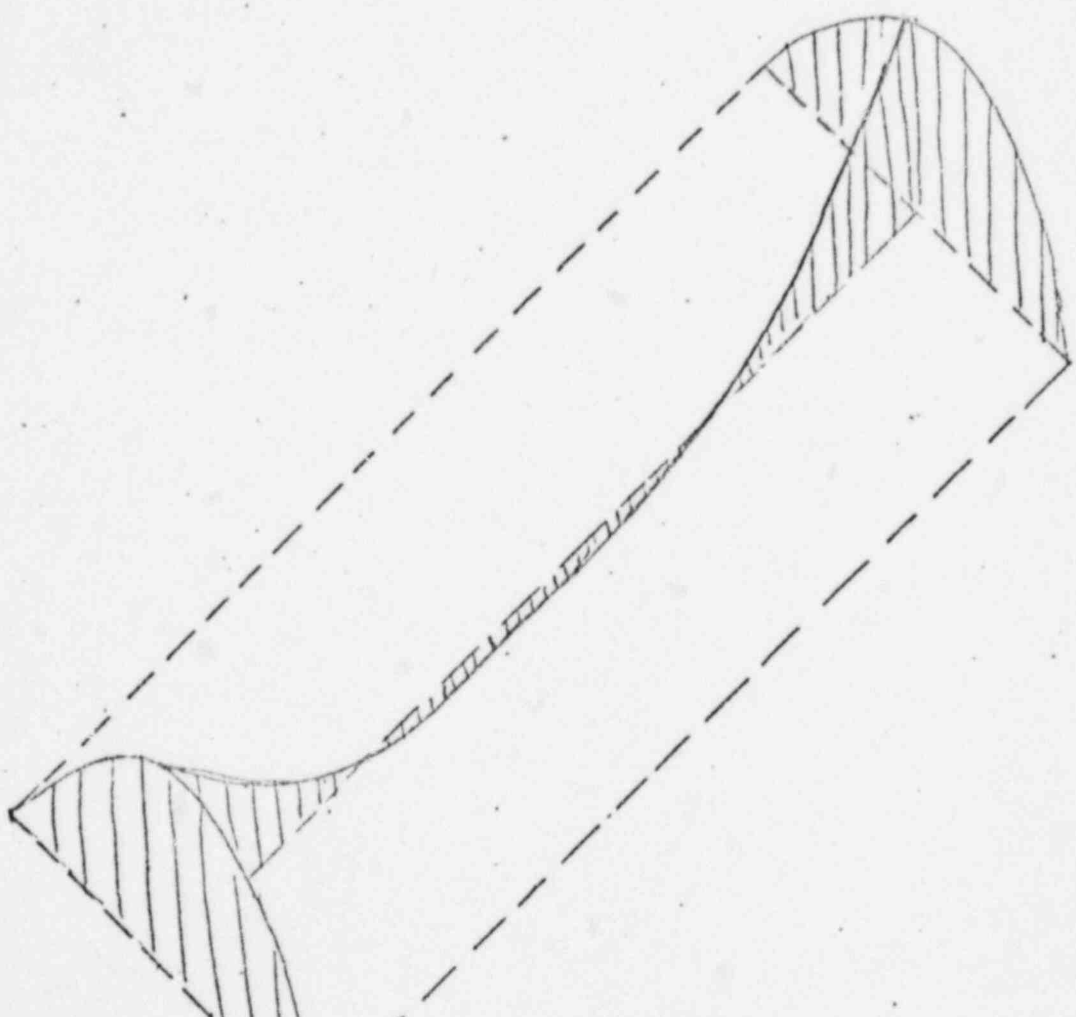
SIMILARLY PEAK ALTERNATING STRESS DUE TO $y = 0.02$ CAN BE SHOWN TO BE

$$\sigma_a' = \pm 36 \text{ ksi}$$

REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

WESTINGHOUSE PWR SYSTEMS DIVISION

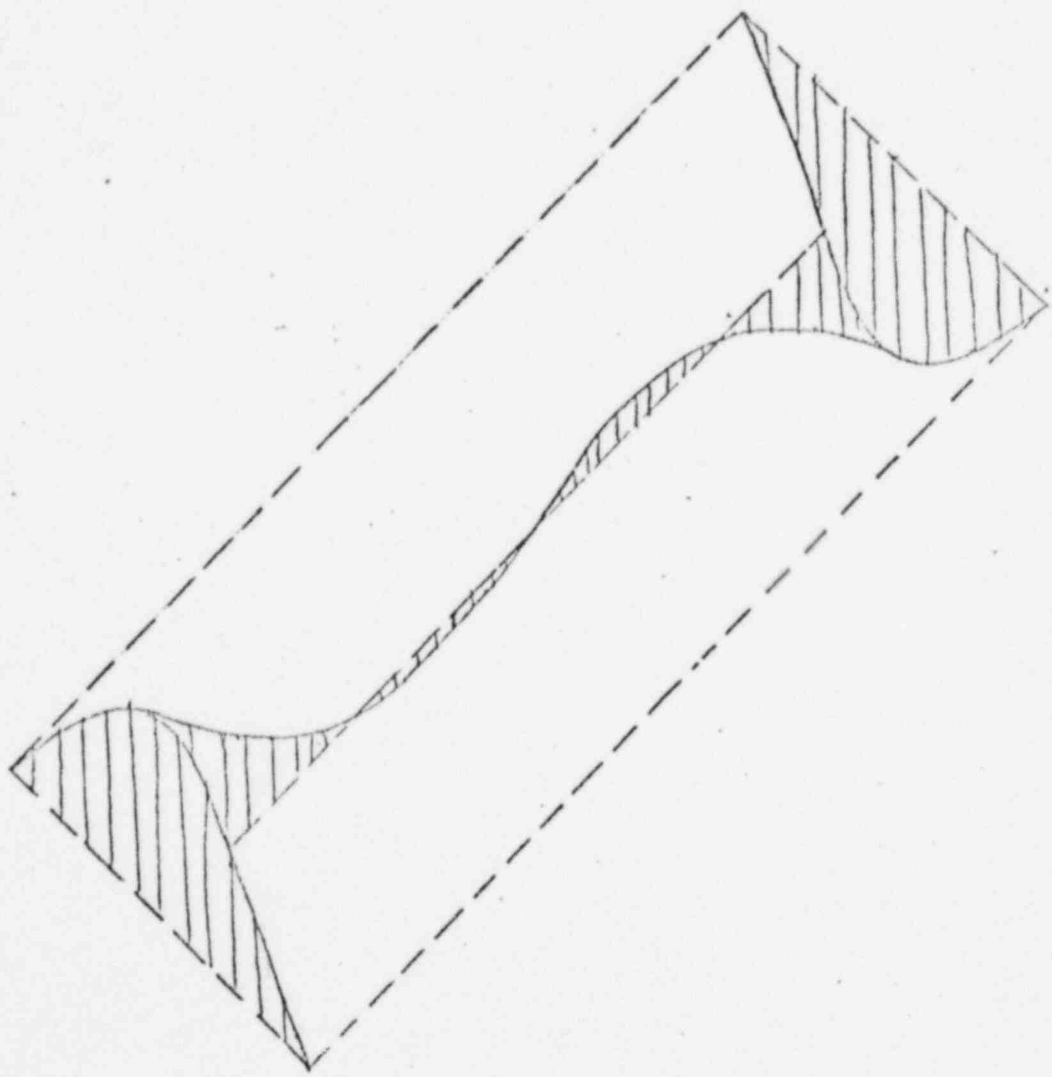
TITLE FIGURE I FIRST MODE SHAPE, f = 148.15 HZ			PAGE OF
PROJECT NORTH ANNA UNITS 1/2	AUTHOR S.S. PALUSAMY DATE 9/12/79	CHK'D. BY Y.S. Lee DATE 9/13/79	CHK'D. BY N/A DATE
S.O. VGBM-100 98	CALC. NO. N/A	FILE NO. N/A	GROUP NTD/MMT/SMG



REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

WESTINGHOUSE FORM 55213C

TITLE FIGURE 2 : SECOND MODE SHAPE, f = 148.18 HZ			PAGE OF
PROJECT North Anna Unit 1/2	AUTHOR S.S. Palusamy DATE 4/13/79	CHK'D. BY Y.S. Lee DATE 4/13/79	CHK'D. BY N/A DATE
S.O. VGBM-10098	CALC. NO. N/A	FILE NO. N/A	GROUP NTD/MHT/SMG



REV. NO.	REV. DATE	AUTHOR	DATE	CHK'D. BY	DATE	CHK'D. BY	DATE

Fig. 2

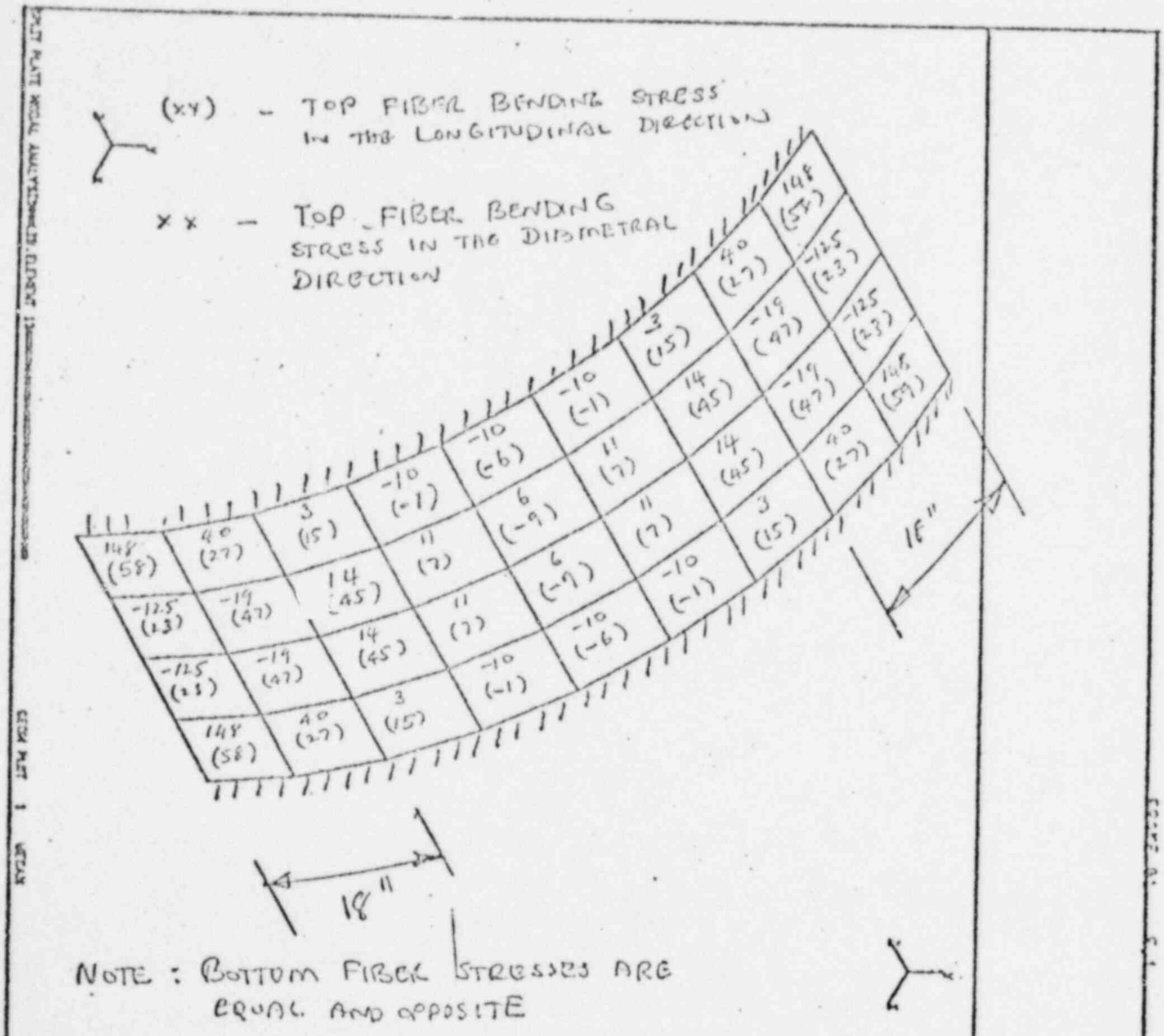


FIGURE 3 FIRST MODAL STRESSES

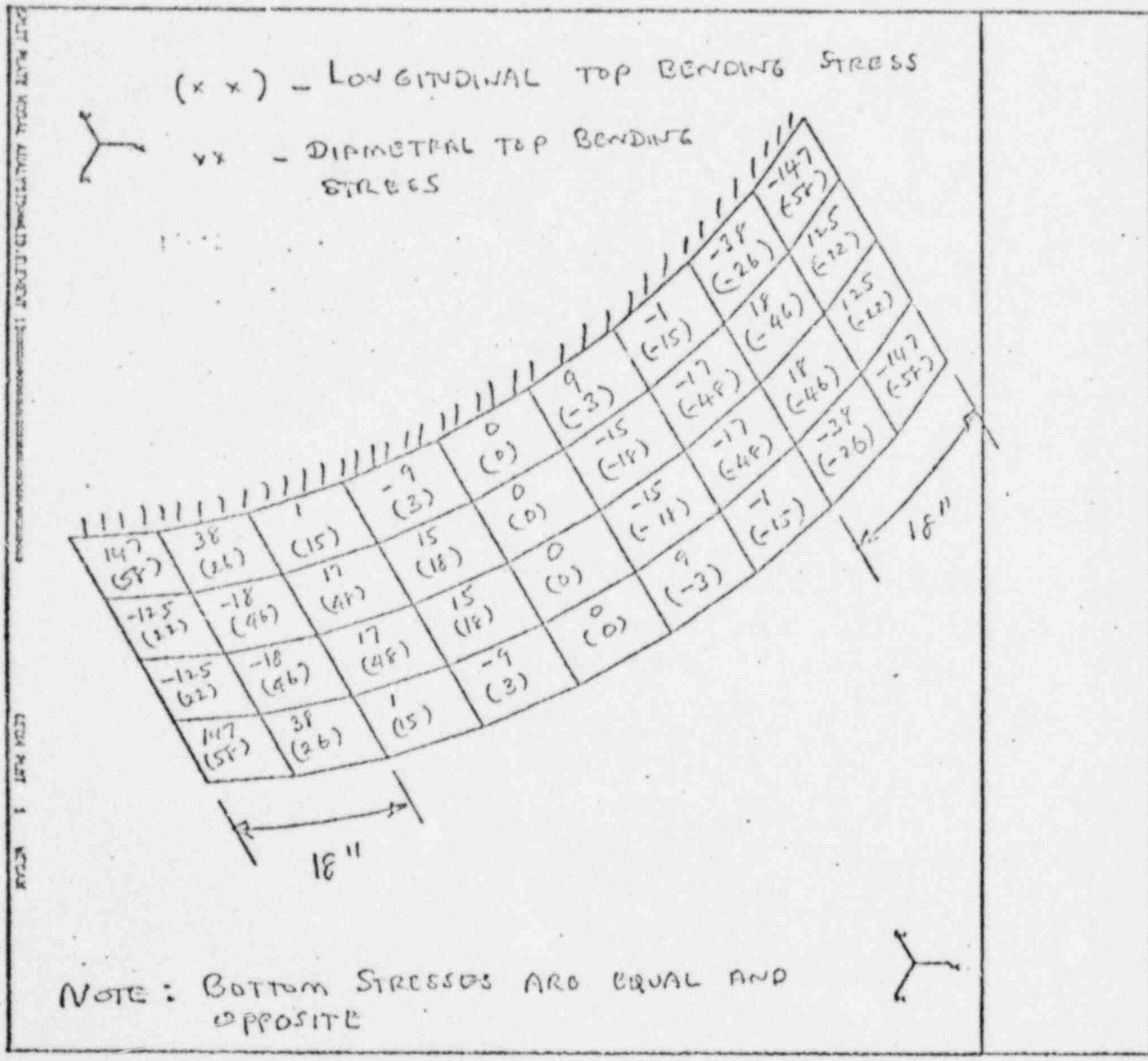


FIGURE 4 SECOND MODAL STRESSES

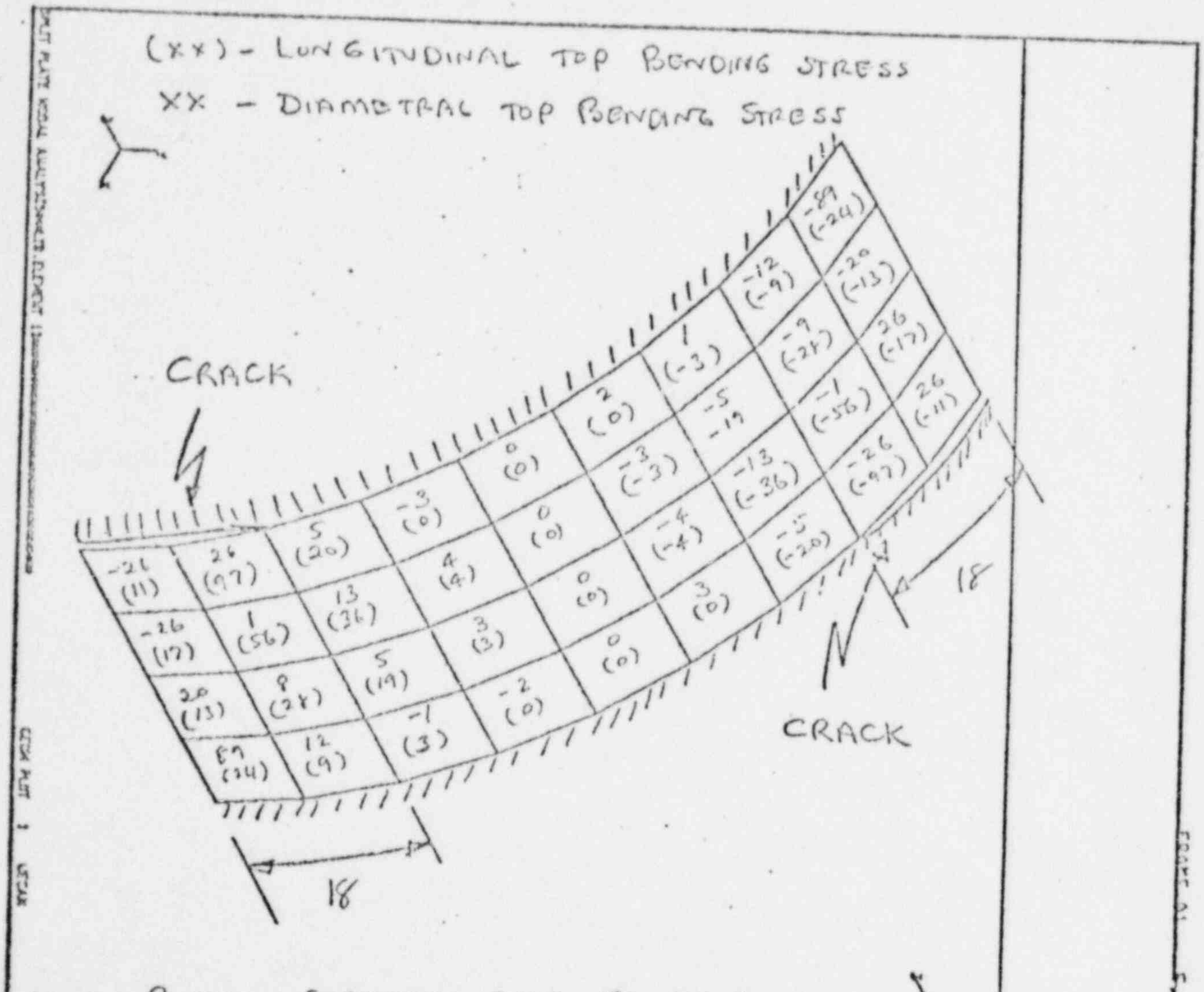
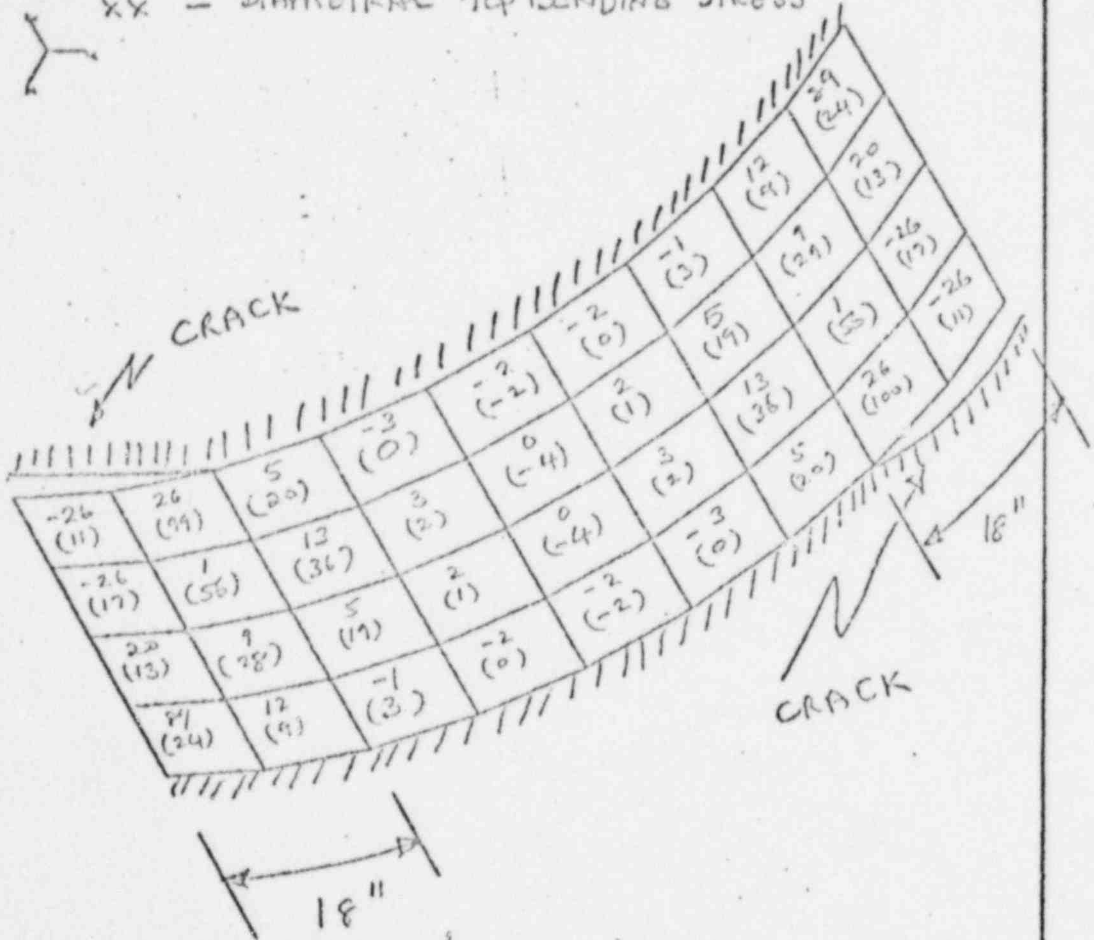


FIGURE 5 SECOND MODAL STRESSES, f = 64.31 HZ

(XX) - LONGITUDINAL TOP BENDING STRESS

XX - DIAMETRAL TOP BENDING STRESS



NOTE: BOTTOM STRESSES ARE EQUAL AND OPPOSITE

FIGURE 6 FIRST MODAL STRESSES, $f = 64.28$ HZ

ULTRASONIC EXAMINATION

As a result of the cracks observed on Unit 2, an ultrasonic examination (UT) procedure was developed to enable a volumetric examination to be made. The objective of this procedure was to examine the splitter plate to elbow weld from the outside diameter (O.D.) of the pipe. A straight beam technique utilizing a 1" transducer and a 1 megahertz signal was employed. The procedure was used on Loop C of Unit 2 which experienced the cracks. This test provided excellent results and clearly demonstrated that cracks of the nature found on Unit 2, would be detected by the procedure.

After the procedure was developed, a UT examination of each splitter elbow on Unit 1 was performed. As a result of this examination it was clear that the flow splitter was structurally sound and could perform its intended function. No defects as found on Unit 2 existed.*

It should be noted that the procedure employed would detect any crack which occurred in the weld area and approximately 4 inches deeper into the splitter plate (10 inches from O.D.). The only stipulation is that the crack surface must be perpendicular to the transducer. As shown in the analytical evaluations the only cracks that would be anticipated to occur in the flow splitter would be identical to those observed in Unit 2, and would be detectable by the UT procedure.

---

* It should be noted that on Loop B of Unit 1, two reflectors were found as a result of the ultrasonic examination. The first reflector was approximately 22" from the leading edge from the splitter plate and was approximately three inches in length. The second reflector was approximately two inches from the previous reflector and was 1/4 inch in length. These reflectors occurred in the weld metal at the junction of the longitudinal and lateral weld of the splitter plates. It was obvious from our inspection that these reflectors were not of the type found in Unit 2. Based on their location it could be expected that these reflectors

would be slag inclusion, lack of fusion, or some other type of weld imperfection due to the high amount of weld material deposited in this particular area. It should be noted that if a crack occurred in this particular area it would r t cause the type of failure which resulted in Unit 2 because of its location (approximately 22 inches from the leading edge of the splitter plate).