

STRUCTURAL EFFECTS
OF
ATWS OVERPRESSURE TRANSIENTS
IN
BYRON JACKSON PRIMARY PUMPS

BABCOCK & WILCOX CO.

NPGD

FEBRUARY 1, 1980

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SUMMARY

The effect of an ATWS transient on the structural integrity of primary pumps manufactured by Byron Jackson for Babcock & Wilcox nuclear reactor is evaluated in this report. The primary effect of the transient on the pump structure is to cause a pressure surge that peaks at less than 3750 psi.

The effect of the transient is evaluated by examining stresses caused by a static pressure of 3750 psi. Also, a quantitative estimate of the deformation is made to determine the potential for interference with rotation of the impeller.

It was found that stress in the flanges, the closure, and those portions of the pump in which the bearings are mounted are within allowable stresses in accordance with ASME Nuclear Pressure Vessel Code for Level C incidents. Stress in portions of the volute and the diffuser vanes do exceed allowable stress for Level C incidents. The location and magnitude of these stresses are presented in this report. These stress levels are well below material rupture strength, and closure bolts are within allowable stress limits; therefore, it is concluded that the structural pressure boundary will not be violated.

An evaluation of deformation resulting from these stresses shows that displacements which could affect the rotating impeller are small compared to available clearances.

This provides assurance that the structural integrity of the primary pump will be retained, and the post-ATWS function of the primary pumps will not be impaired by an ATWS pressure pulse, which was determined in accordance with current NRC guidelines.

INTRODUCTION

SCOPE

This document addresses the effect of an ATWS pressure transient on the structural integrity of the primary reactor coolant pumps built for the reactors listed in Table 1.

TABLE 1

REACTORS USING BYRON JACKSON PUMPS

<u>UTILITY</u>	<u>REACTOR</u>
Consumers Power Co.	Midland 1 Midland 2
Arkansas Power & Light Co.	ANO-1
Florida Power Corp.	Crystal River 3
Toledo Edison Co.	Davis Besse 1

A description of the pump and its stress analysis is included. A comparison of stress in the pressure boundary caused by the worst ATWS transient is made with allowable stress for Level C incidents. A quantitative estimate of distortion is included to determine potential effects on impeller clearances. Flange seal capabilities are also evaluated.

Earlier evaluation of the pump is included in topical report BAW 10099. That report determined the pressure at which stress reached allowable values for emergency conditions. The re-evaluation herein is based on current revised guidelines from NRC for ATWS evaluation.

ASME Allowable Stress

Allowable stress for Level C service limits are given in Subsection NB of Section III, Division 1 of the ASME Pressure Vessel codes. The

limiting requirements for this application requires that primary membrane stress not exceed $1.2 S_m$, and that primary membrane plus primary bending stress not exceed $1.8 S_m$.

The pump is made from austenitic stainless steel casting per the requirements of ASTM A-351 Grade CF8M. The value of S_m for this material at $680^{\circ}F$ is 16,400 psi. Therefore:

Primary membrane allowable stress = 19700 psi

Primary membrane + primary bending allowable stress = 29500 psi

Irradiation Effects

The primary pumps are sufficiently distant from the reactor to make irradiation effects negligible, therefore material properties of structural components in the pump do not include any modification for irradiation effects.

ATWS Transient

The pressure transient resulting from an ATWS event has been determined using the NRC guidelines of Alternate Number 3 specified in Volume 3 of NUREG 0460. The results of this study are included in a letter report bearing the title "Analysis of B&W NSS Response to ATWS Events", January 1980.

The above report shows the pressure transient peaks at a pressure less than 3750 psi, and that the coolant temperature does not exceed $660^{\circ}F$. The pump stresses are evaluated at 3750 psi, and the allowable stress for the pump materials is based on a temperature of $680^{\circ}F$.

PUMP DESCRIPTION

The Byron Jackson pumps are vertical, single stage, single bottom suction, horizontal discharge, centrifugal diffuser casing units with controlled leakage mechanical seals. They are driven by squirrel cage, vertical

AC induction motors. Some of the design features of this pump are listed below.

TABLE 2 - PUMP DESIGN PARAMETERS

Design Pressure	2,500 psig
Design Temperature	650°F
Operating Pressure	2,250 psig
Rated Flow - 4 Pumps Operating	88,000 gpm
Maximum Flow	130,000 gpm
Developed Head - 4 Pumps Operating	327 feet
Suction Nozzle Inner Diameter	28 inches
Discharge Nozzle Inner Diameter	28 inches
Power Source	6,600 volts, 3 Phase, 60 Hz
Motor Horsepower at Operating Temp.	6,090 HP
Rotating Speed	1,185 rpm

The pump assembly consists of the motor, the driver mount, the rotating element (coupling, impeller, and shaft), the heat exchanger, the seal cartridge (and mechanical seals), the cover plate, and the case.

Each reactor contains four of these pump assemblies for the purpose of recirculating primary reactor coolant. The pumps are installed in the cold leg of the piping system between the steam generator and the reactor vessel.

The general arrangement of the pumps and sketches showing pertinent details of the pump are shown in Figures 1, 2 and 3.

GENERAL DESCRIPTION OF MATHEMATICAL MODEL

The pump is modeled in two parts: pump case and case closure. The pump case model is a three-dimensional finite element model. This model uses solid elements for the flanges and diffuser vanes. Shell elements are used for the volute, and for suction and discharge nozzles.

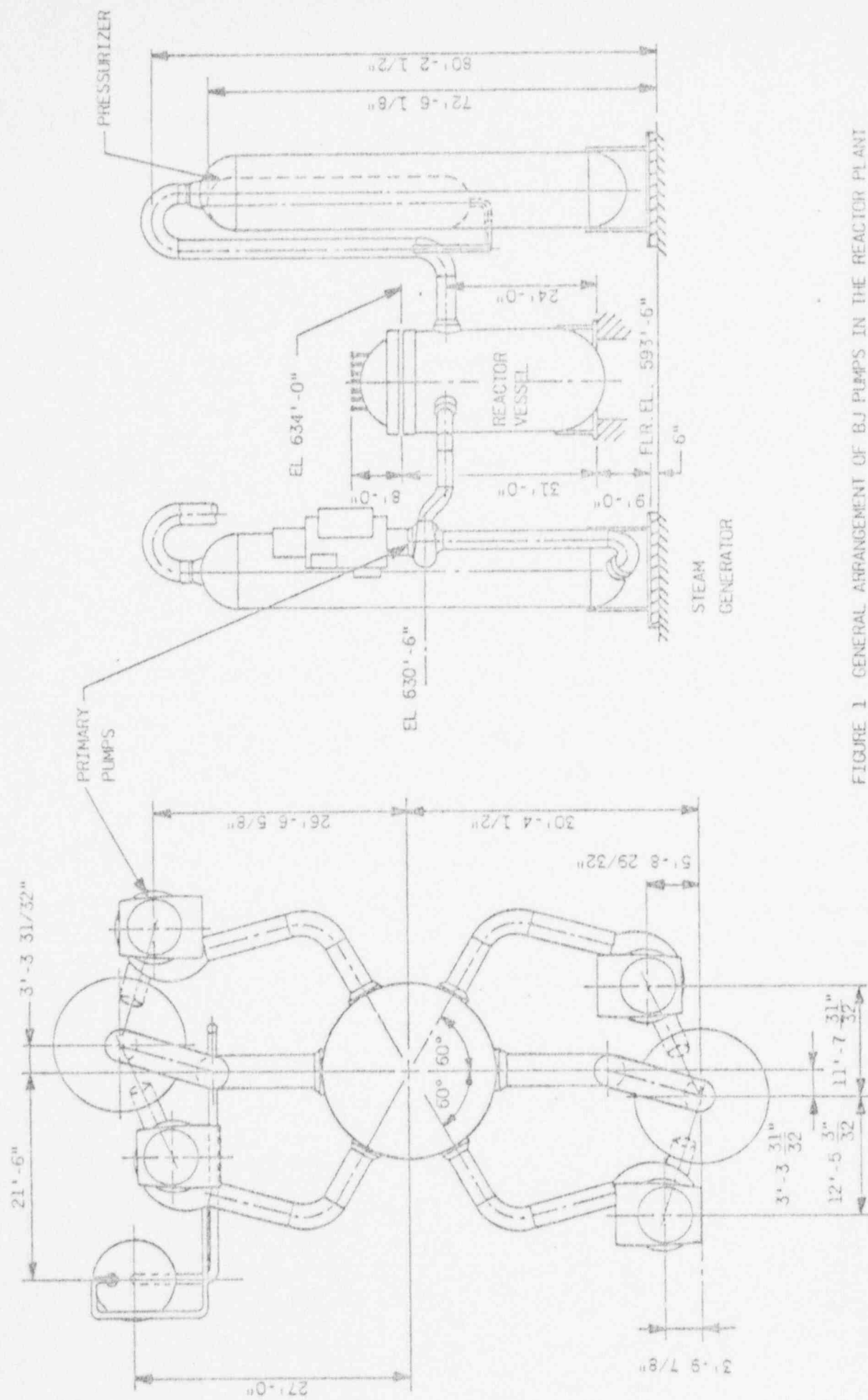


FIGURE 1 GENERAL ARRANGEMENT OF BJ PUMPS IN THE REACTOR PLANT

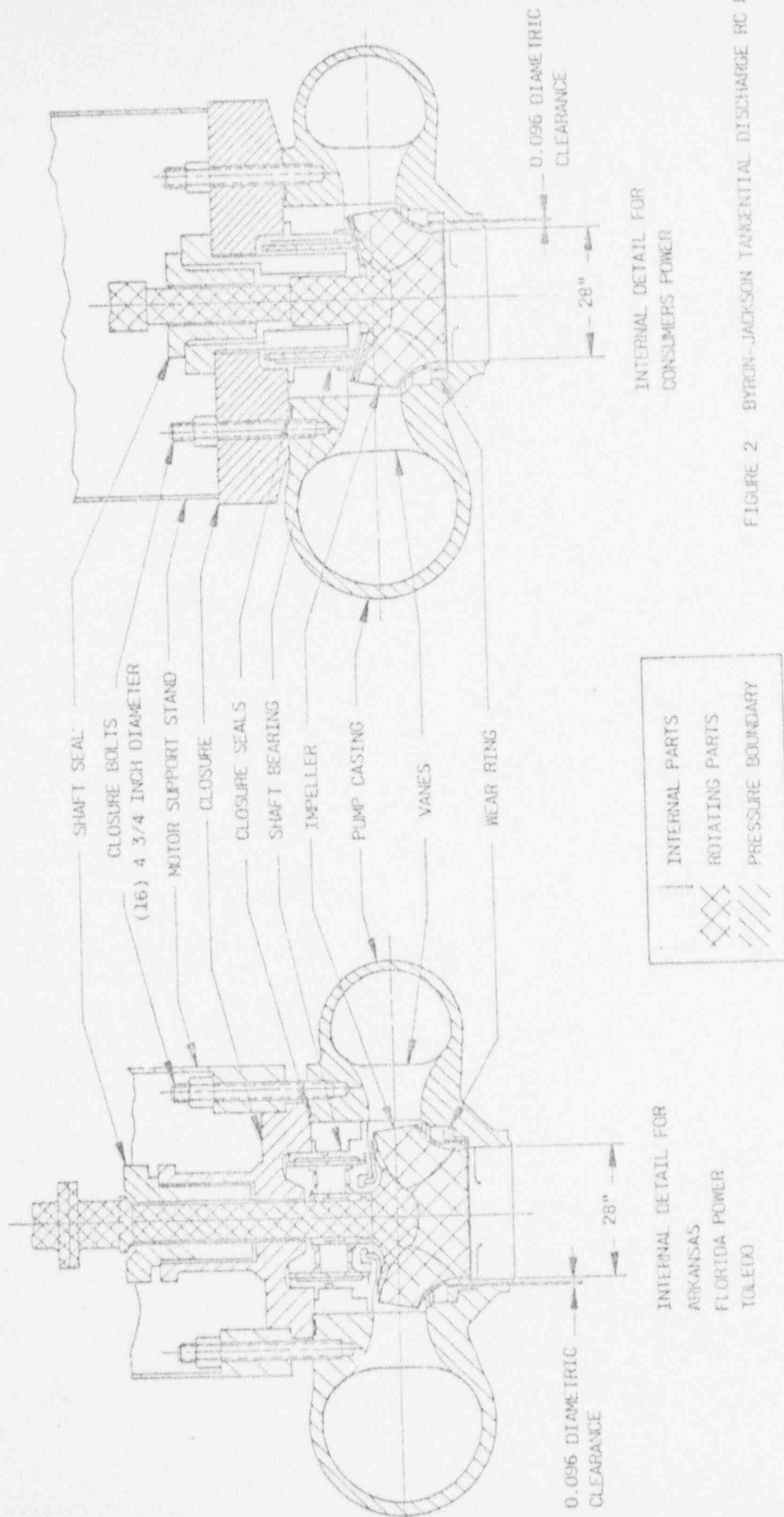


FIGURE 2 BYRON-JACKSON TANGENTIAL DISCHARGE RC PU

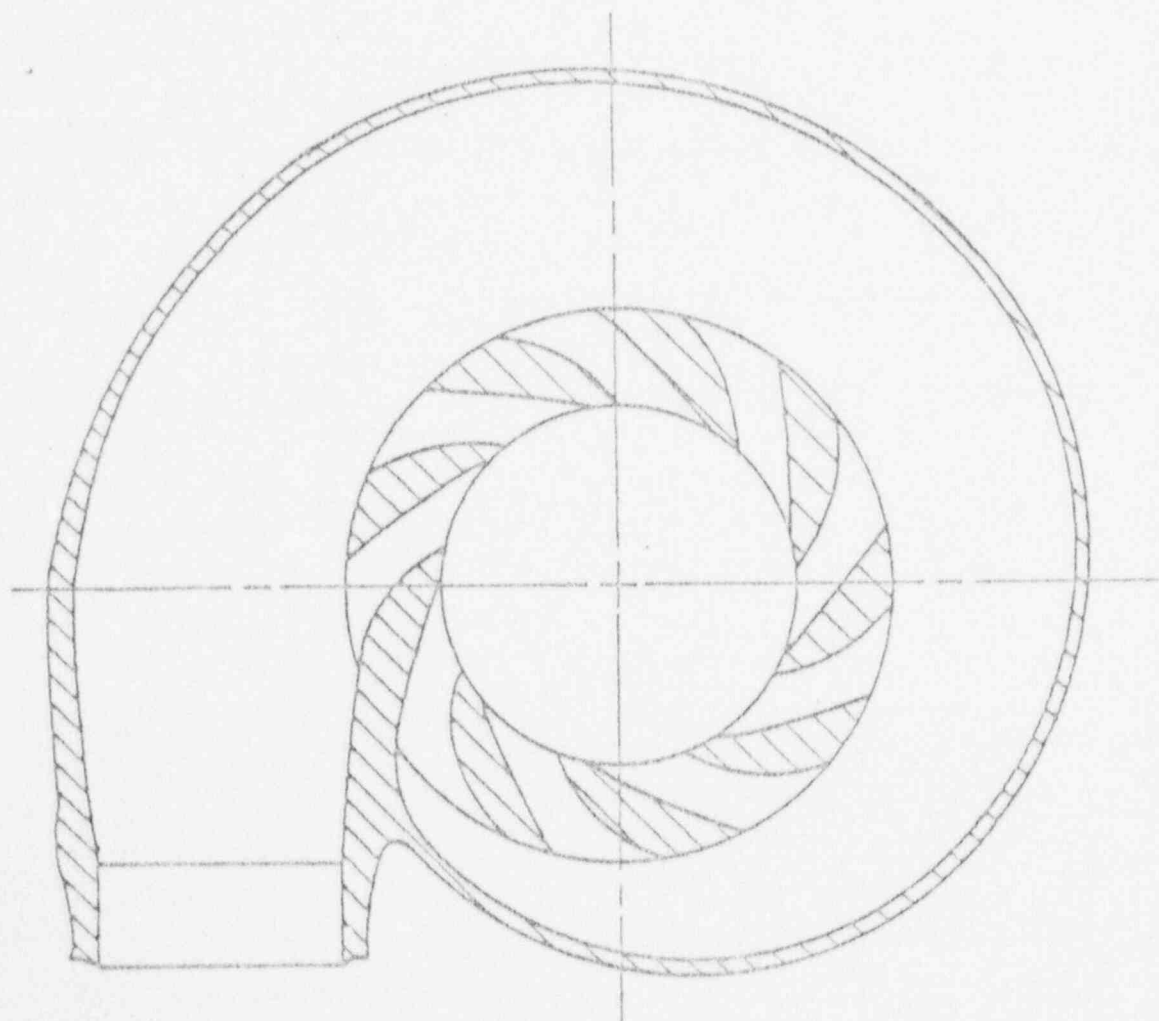
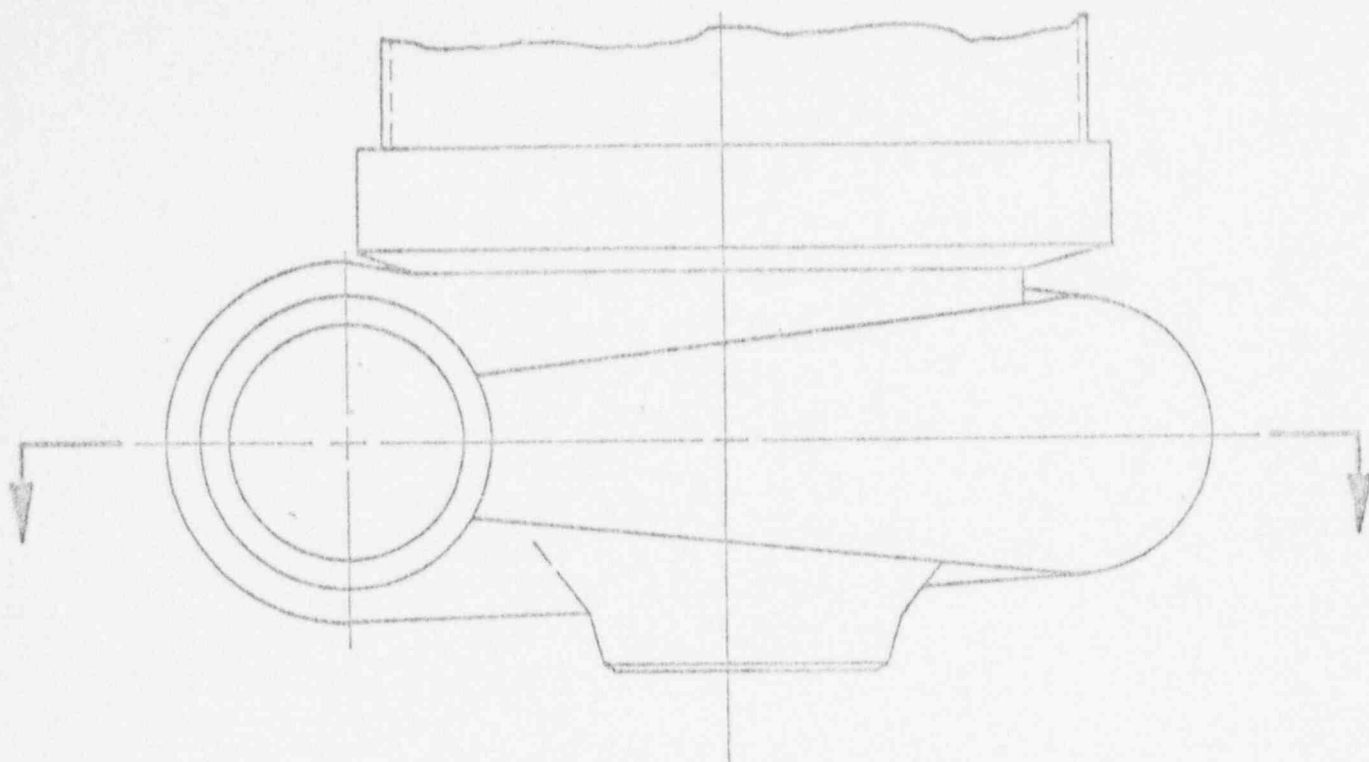


FIGURE 3 BYRON JACKSON PUMP CASING

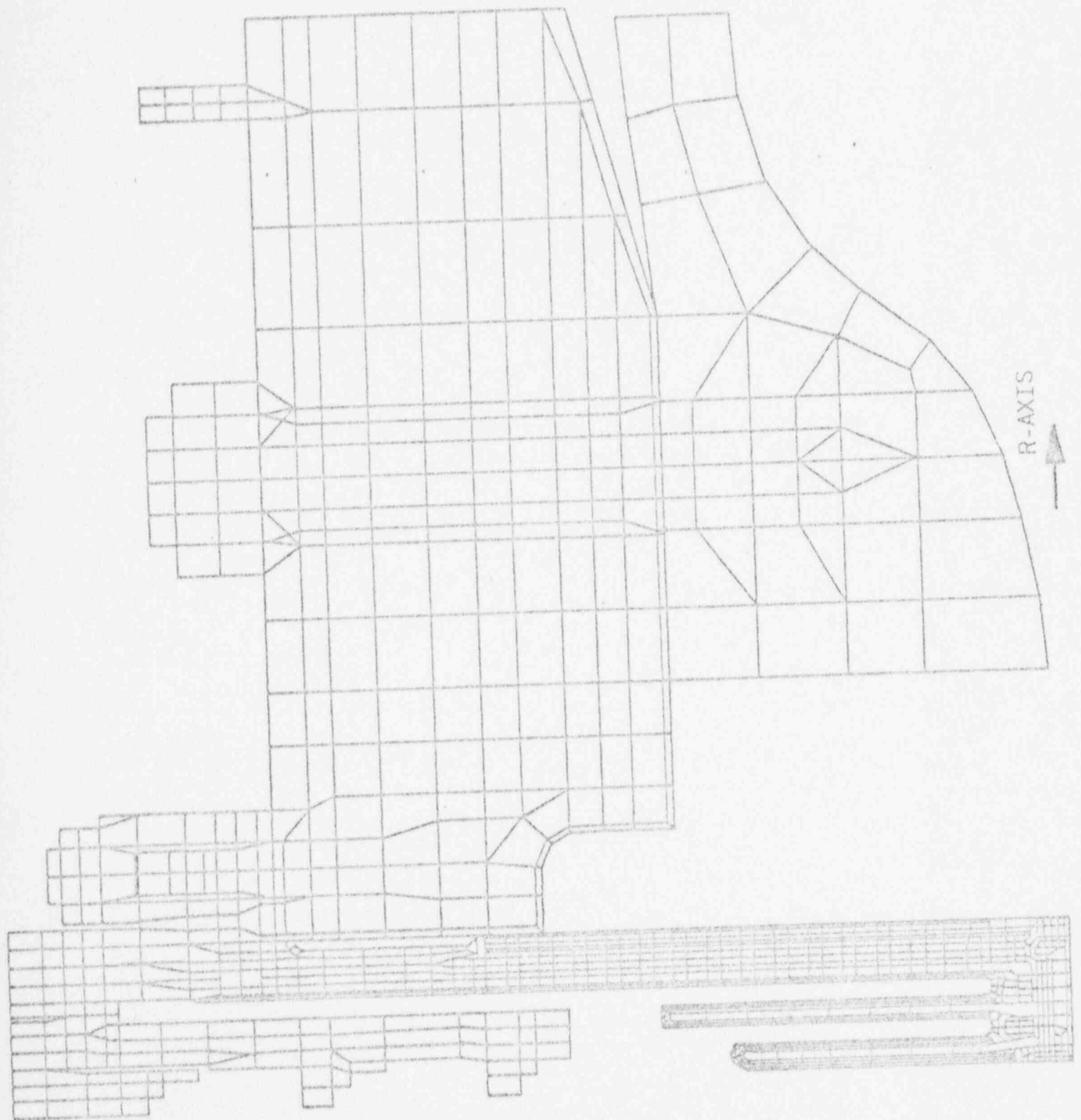


FIGURE 4 FINITE ELEMENT MODEL

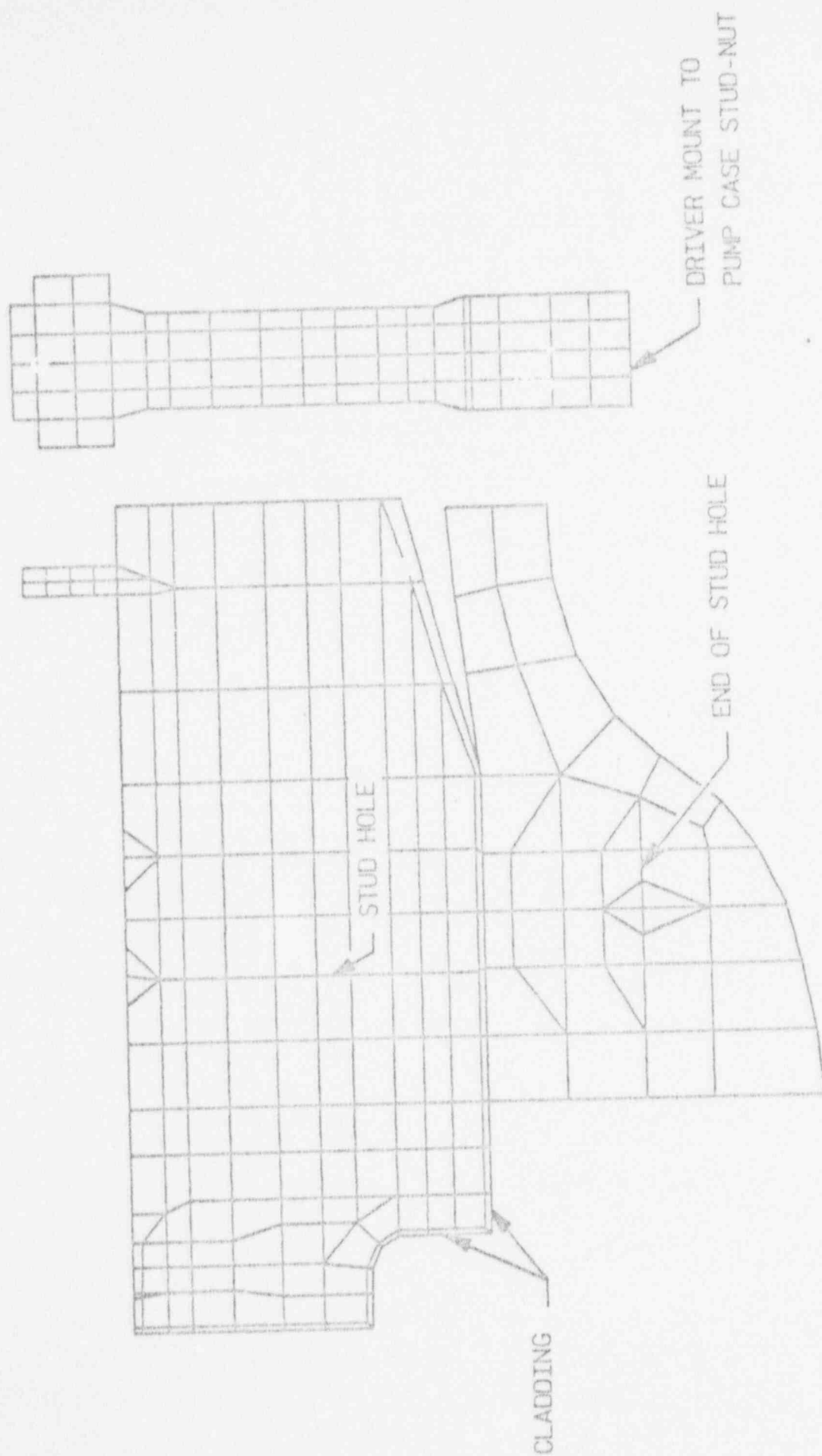


FIGURE 5 EXPLODED VIEW OF CASE CLOSURE MODEL

evaluate the stresses throughout the case closure for internal pressure and bolt preloading. The solution consists of nodal deflections and element coordinate stresses calculated at the element centroid.

Conditions Analyzed

Several conditions are evaluated. These are referred to as Basic Conditions. They include a bolt preload condition, a design pressure condition and a normal operation pressure condition.

STRESS EVALUATION

The Byron Jackson pump evaluated in this report is used in several of the B&W Reactors. A listing of these reactors is given in the Introduction to this report. The peak pressure during an ATWS transient for these reactors has been determined to be slightly less than 3750 psi (see page 3). The allowable stress intensity for ATWS transients are based on ASME Section III rules for Level C conditions, (previously "emergency conditions"). Primary membrane stress is compared to $1.2 S_m$ (19700 psi at 680°F). Primary membrane plus bending is compared to $1.8 S_m$ (29500 psi at 680°F). (For additional discussion of allowable stress, see page 2.)






Stress resulting from a pressure of 3750 psi has been compared to allowable values. Calculated stress intensities which exceed allowable values are summarized in Table 3. The number of elements whose stress exceed the allowable are tabulated as a percentage of the elements in the pump case. The stress values are those at the centroid of each element, and may be used as a conservative evaluation of primary membrane stress. (Conservative, because true membrane stress for vanes is the average stress across the section. No attempt was made to remove this conservatism, because stress intensities which include 6 components of stress cannot be averaged directly.)

The location of these elements are presented in Figures 6 through 11. The highest stress intensities occur in the root sections of vane Number 9, which is adjacent to the crotch. (Vane 9 is an extension of the crotch.)

CENTROID STRESS INTENSITIES

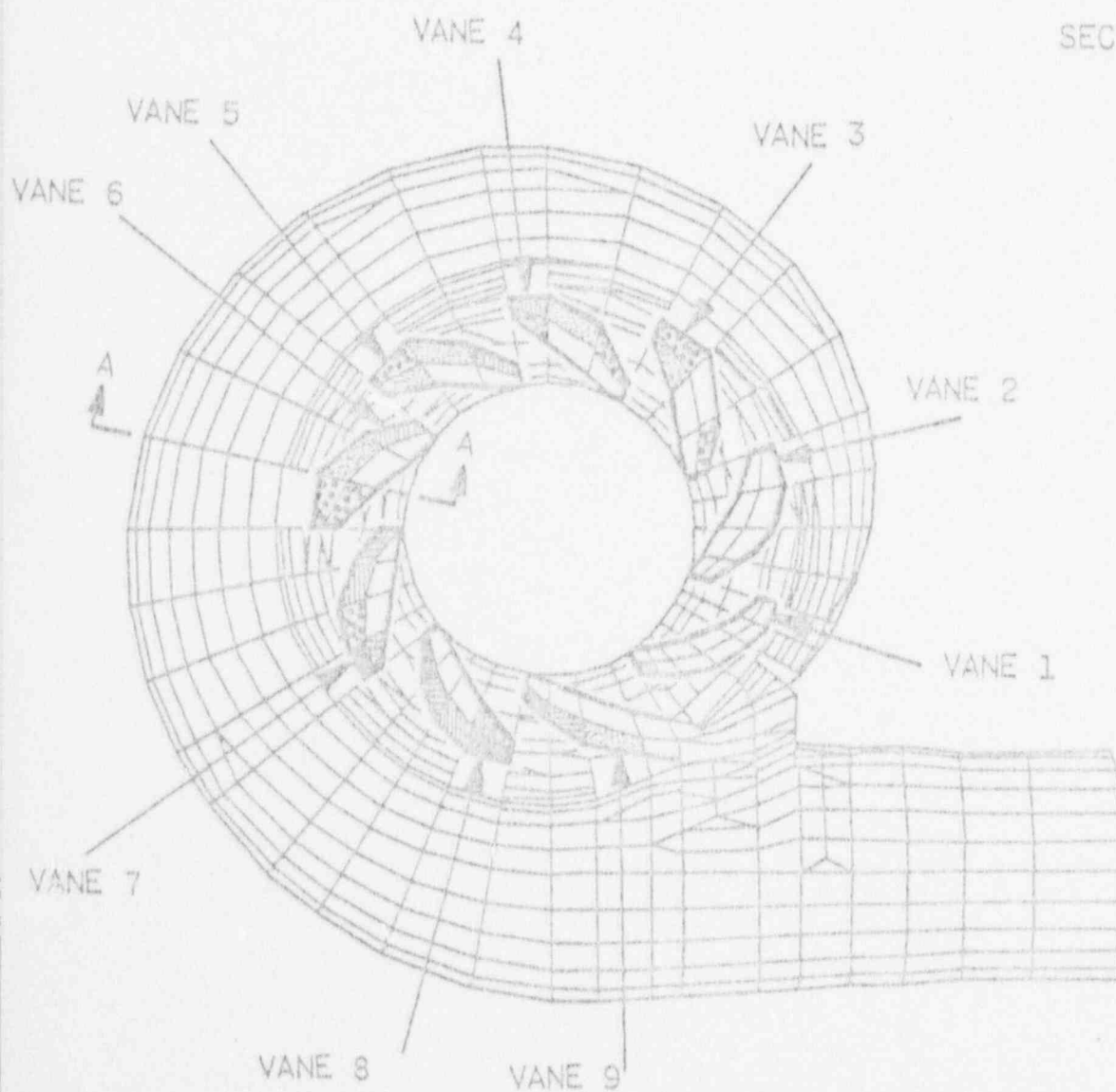
VANES - PLANE A

FIGURE 6

	% 1-2 S_M	% ELEMENTS
	100-110	0.2
	110-120	0.3
	120-130	0.3
	130-140	0.2
	140-150	0.2

PLANE A




SECTION A-A



CENTROID STRESS INTENSITIES

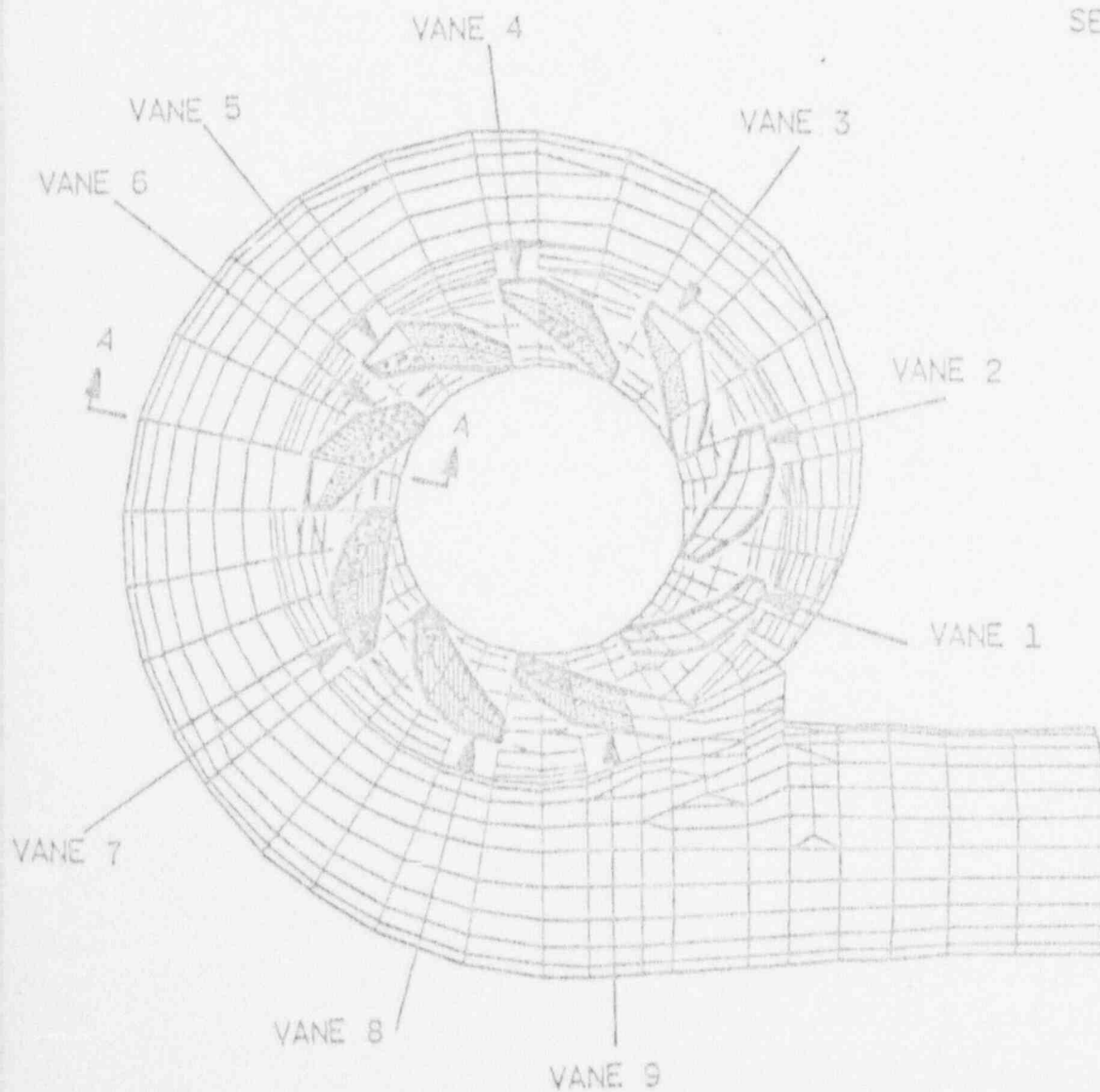
VANES - PLANE B

FIGURE 7

	% 1.2 S _M	% ELEMENTS
	100-110	0.6
	110-120	0.6
	120-130	0.4

PLANE B





SECTION A-A



CENTROID STRESS INTENSITIES

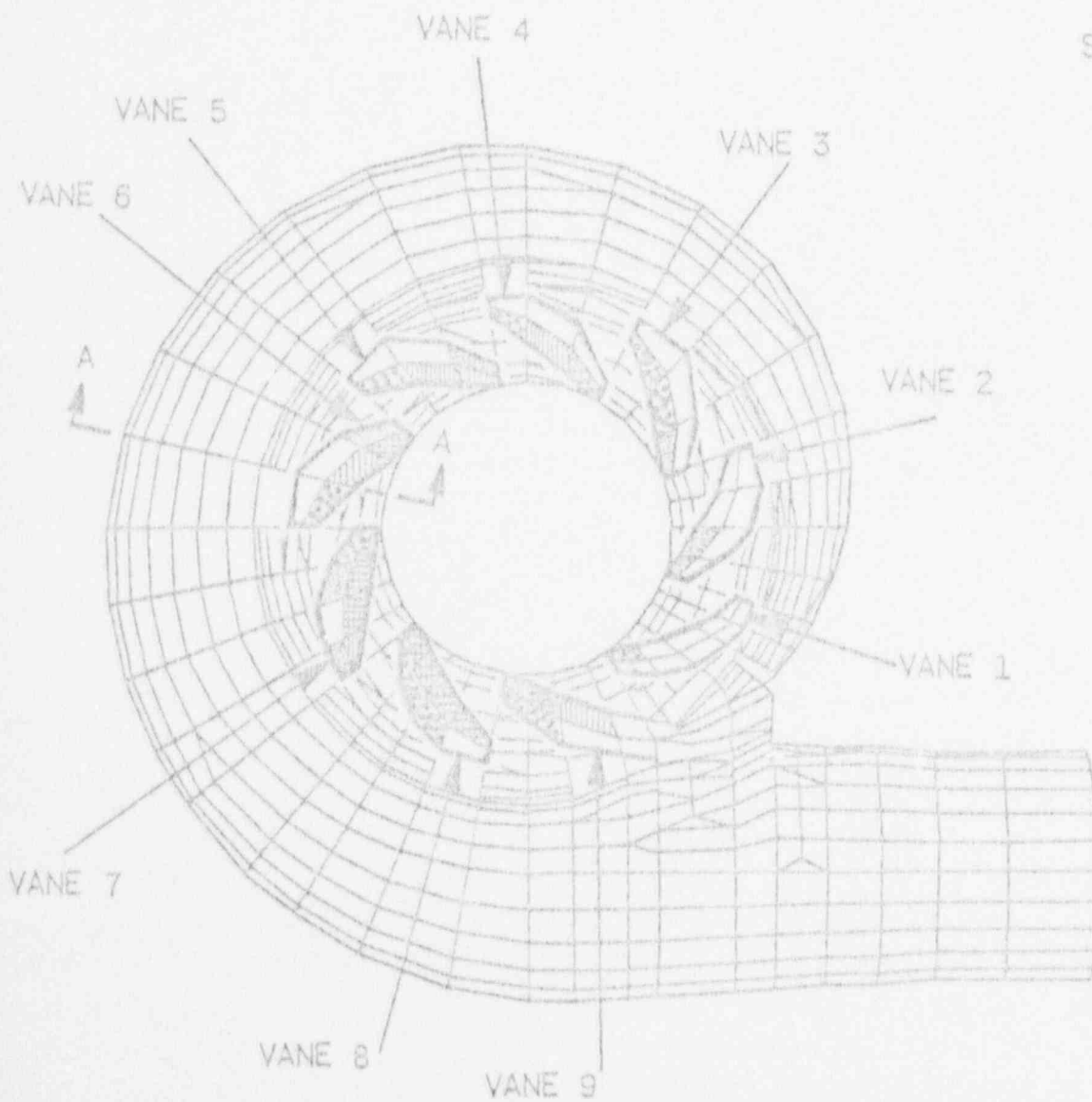
VANES - PLANE C

FIGURE 8

	% $1.2 S_M$	% ELEMENTS
	100-110	0.3
	110-120	0.5
	120-130	0.4
	130-140	0.4

PLANE C

SECTION A-A



CENTROID STRESS INTENSITIES

VANES - PLANE D

FIGURE 9

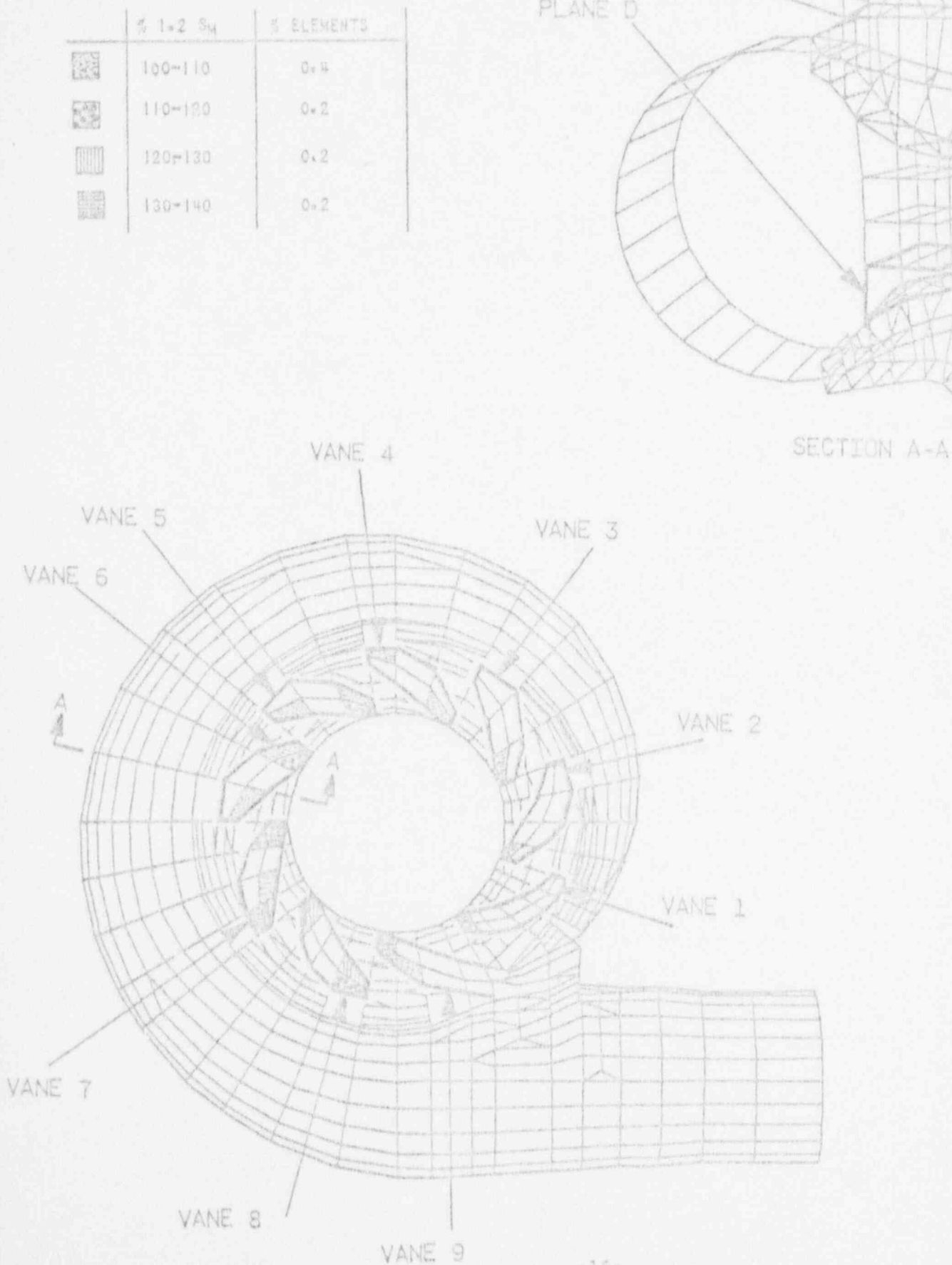
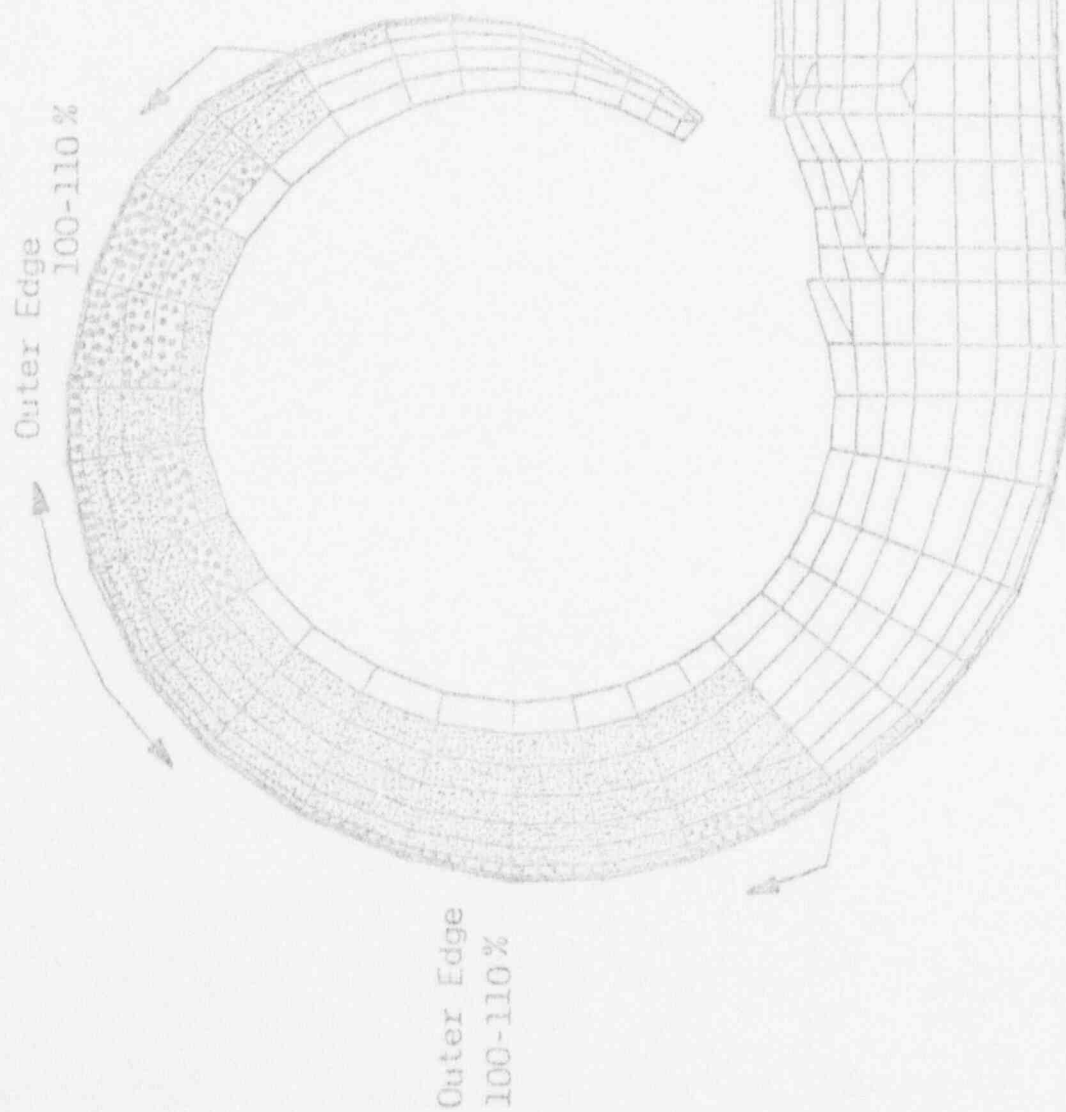


FIGURE 10. CENTROID STRESS INTENSITIES
TOP HALF OF VOLUTE



$\frac{1}{2}$ 1-2 S_M	% ELEMENTS
100-110	3.0
110-120	1.3

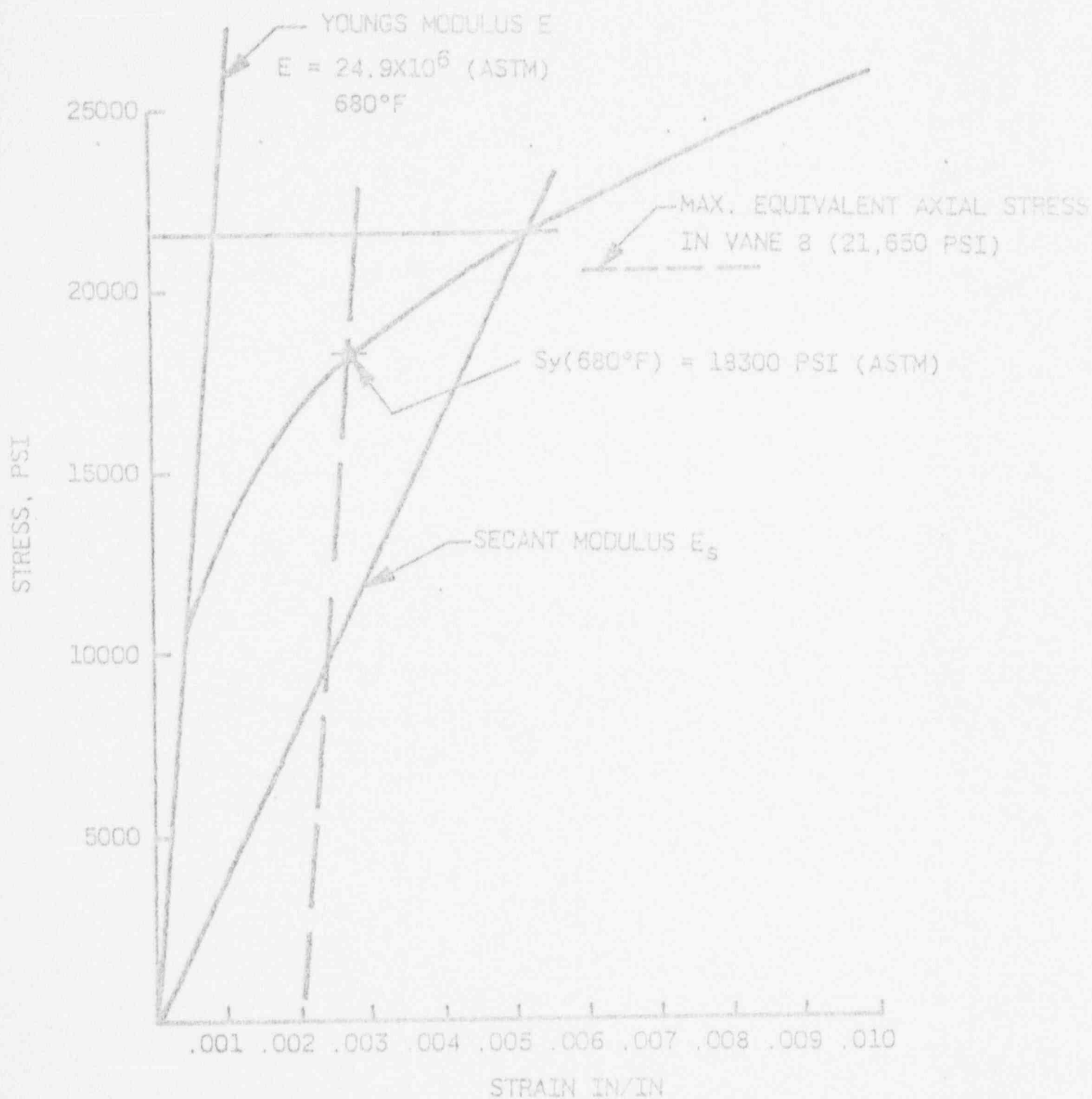


FIGURE 12 DETERMINATION OF INELASTIC STRAIN RATIO

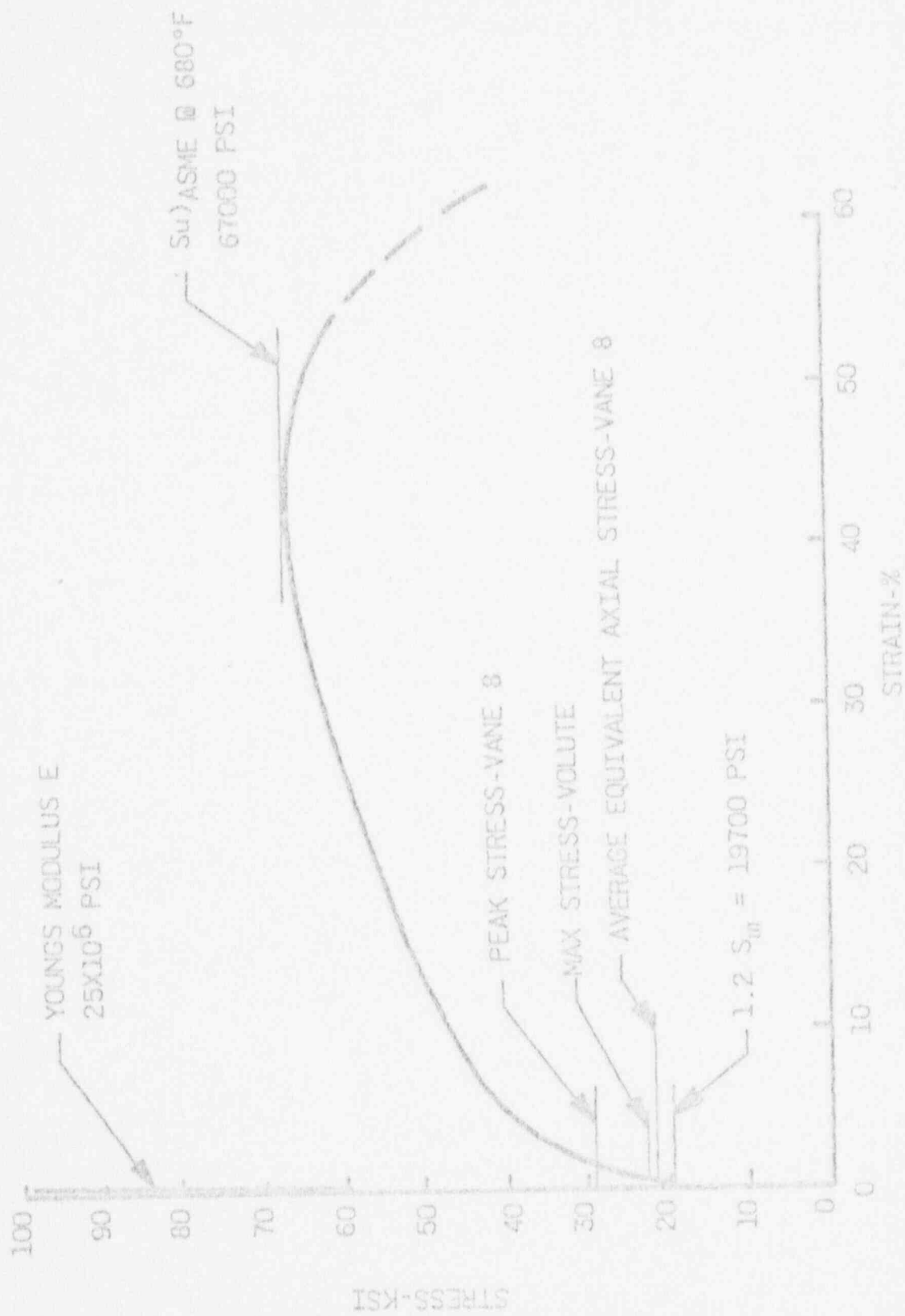


FIGURE 14 STRESS COMPARED TO STRESS-STRAIN CURVE