Enclosure (1)
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EVALUATION OF THE IMPACT
OF A THERMAL SHIELD SUPPORT
SYSTEM FAILURE IN
THE FT. CALHOUN REACTOR

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1.0 INTRODUCTION

A ten year Inservice Inspection was conducted by Southwest Research, Inc. on the Ft. Calhoun reactor internals during late 1982 and early 1983. No unusual or unanticipated wear conditions that would raise questions concerning the structural integrity of the thermal shield and support system were reported. A ten year Inservice Inspection was also carried out in 1982 on the Maine Yankee internals. This inspection revealed that three of the thermal shield positioning pins located at the top of the thermal shield had become displaced. However, visual inspection of the thermal shield and support system revealed no indication of abnormal motion or wear of the thermal shield and support lugs. Because of the observations at Maine Yankee, an inspection was made during a refueling outage at St. Lucie 1 in early 1983. This inspection indicated extensive damage to the thermal shield and associated support system.

As a result of these inspection findings, an evaluation was carried out by Combustion Engineering, Inc. to assess two areas of potential concern:

1. If the thermal shield support system failed completely in the Fort Calhoun reactor, will the core support barrel snubbers catch and limit the downward travel of the thermal shield?

2. What will the impact of the dropped thermal shield be on the core inlet flow distribution and on the system flow rate?

This report presents the results of the evaluation in these two areas.

2.0 CONCLUSIONS

2.1 Mechanical Considerations

In the event that the thermal shield support system fails, the thermal shield will move downward less than two inches until coming into contact with the snubber spacer blocks on the core support barrel. The calculated stresses on the snubber spacer blocks resulting from a failure of the thermal shield support system are less than the normal operating allowable stresses.

After contacting the snubber spacer blocks, the thermal shield could damage the surveillance capsule assemblies. This damage can occur if the positioning pins are displaced radially outward and at least half of the thermal shield support lugs are completely removed. The core support barrel will prevent the thermal shield from contacting the reactor vessel.

2.2 Hydraulic Considerations

If the thermal shield drops to the snubber blocks in a concentric position there will be no impact on the core inlet flow distribution and system flow rate. This conclusion is based on the fact that the vertical position of the thermal shield will change by less than 2 inches, and in a concentric final position, will not affect the flow distribution entering the lower plenum and core.

In the event that the thermal shield drops to the snubber blocks in a fully eccentric position, the azimuthal variation in flow rate exiting from the thermal shield region will range from -5 percent to +13 percent about the average flow rate due to the effects of eccentricity. This variation is expected to have negligible impact on the core inlet flow distribution. Larger variations have been observed in flow model measurements taken at the bottom of the thermal shield region, for the concentric case, with no corresponding azimuthal variations being observed in the core inlet flow distribution. The impact on system flow rate will be a small increase of 0.2 percent of design flow rate (190,000 gpm). This flow increase is due to a reduced hydraulic resistance for the thermal shield region when the thermal shield is shifted to an eccentric position.

In the unlikely event that the thermal shield drops to a fully eccentric, tilted position, the azimuthal variation in the flow rate exiting from the thermal shield region will range from -31 percent to +26 percent about the average flow rate. This azimuthal variation is expected, at worst, to reduce the flow to the limiting assembly by 12 percent. The increased hydraulic resistance of the tilted thermal shield will cause a reduction in system flow rate of 2.2 percent of design flow rate.

The combined effect of the core inlet flow distribution change and the system flow rate reduction is a reduction in overpower margin by at most 14 percent. The overpower margin that is available to accommodate Anticipated Operational Occurrences (17 percent) is sufficient to offset the impact of the dropped thermal shield, and prevent the violation of the minimum DNBR limit.

3.0 COMPONENT DESCRIPTION

The major support member of the reactor internals is the core support assembly, Figure (1). This structure consists of the core support barrel, the core support plate and support columns, the core shroud, the thermal shield, the core support barrel to pressure vessel snubbers and the core support barrel to upper guide structure guide pins. The major material for the assembly is Type 304 stainless steel. Figure (2) shows the reactor internals installed in the reactor vessel.

The core support barrel and the thermal shield are the components of primary concern for this evaluation. A further description of these components is given below.

3.1 Core Support Barrel

3.1.1 Physical Description

The support barrel is a right circular cylinder with a nominal inside diameter of 120 5/8 inches and a minimum wall thickness in the weld preparation area of 1 inch. It is supported and suspended by a 4 inch thick ring flange from a ledge on the reactor pressure vessel. The core support barrel in turn supports the core support plate upon which the fuel assemblies rest. The core support plate transmits the weight of the core to the core support barrel by means of vertical columns and a beam structure. Four alignment keys, located 90 degrees apart, are press fitted into the flange of the core support barrel. The reactor vessel, closure head and upper guide structure assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

3.1.2 Mechanical Design and Functional Description

Since the core support barrel is 26 feet long and is supported only at its upper end, coolant flow induced vibrations are prevented by amplitude limiting devices, or snubbers. These snubbers, Figure (3), are installed near the bottom outside end of the core support barrel. The snubbers consist of six equally spaced lugs around the circumference and are the grooves of the "tongue-and-groove" assembly; the reactor vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. Radial and axial expansions of the core support barrel are accommodated, but lateral movements are restricted. The reactor vessel tongues have bolted and locked welded Inconel X shims and the core support barrel grooves are hardfaced with Stellite to minimize wear.

3.2 Thermal Shield

3.2.1 Physical Description

The three inch thick, Type 304 stainless steel thermal shield is a cylindrical structure with an inside diameter of 127 inches and a height of 164 inches.

The thermal shield is supported at the top by eight equally spaced support lugs welded to the outer periphery of the core support barrel. Support pins are fitted during assembly to position the thermal shield on the support lugs. These support pins are welded to the thermal shield and have a .001 to .002 inch clearance on the sides to permit relative thermal expansion of the core support barrel and thermal shield. The thermal shield is positioned radially by a total of twenty-four positioning pins, Figure (4). Eight of the pins are located approximately 25 inches below the tops of the support lugs and the remaining sixteen positioning pins are located approximately 10 inches from the bottom of the thermal shield. The positioning pins thread into the thermal shield and are preloaded against the core support barrel. Locking collars are threaded onto the positioning pins, preloaded and lock welded to both the positioning pins and thermal shield.

4.0 MECHANICAL EVALUATION OF THERMAL SHIELD SUPPORT SYSTEM FAILURE

The thermal shield is located in an annulus formed by the core support barrel on the inside and the reactor vessel on the outside. The thermal shield is supported axially near the top by support pins that rest on the support lugs attached to the core support barrel. In the event that the thermal shield support system fails, downward motion of the thermal shield is limited by the snubber spacer blocks attached to the lower section of the core support barrel. The distance from the bottom of the thermal shield to the top of the snubber spacer blocks was 1 15/16 inches when initially installed.

An evaluation has been performed to determine the consequences of the thermal shield coming free from the support lugs during plant operation. After coming free from the support lugs, the thermal shield can move downward 1 15/16 inches until contacting the snubber spacer blocks. For purposes of this evaluation, a conservative assumption was made that the thermal shield contacted only one snubber spacer block and the entire load was resisted by this one block. The dynamic forces applied to the snubber spacer block were determined by combining the mechanical and hydraulic loads forcing the thermal shield downward and multiplying these forces by a dynamic load factor. The stresses resulting from the dynamic forces were combined with the normal operating stresses and then compared to the normal operating allowable stress intensities given in the ASME Boiler and Pressure Vessel Code, 1977 Edition.

The resultant stress in the core support barrel at the interface with the weldment on which the snubber spacer blocks rest, Figure (3), is approximately 60 percent of the normal operating allowable stress intensity. The resultant stress in the snubber spacer block bolts is approximately 50 percent of the normal operating allowable stress intensity.

This evaluation considered that the eight thermal shield support lugs failed simultaneously allowing the thermal shield to move quickly downward for 1 15/16 inches until contacting the snubber spacer blocks. Inspection of the thermal shield at St. Lucie Unit One indicates that the support lugs did not fail simultaneously or become completely separated from the core support barrel or thermal shield. The thermal shield at St. Lucie has moved downward more than

one and one-half inches through wear of the contact surfaces and yielding of the material. This suggests that failure of the thermal shield support system at Fort Calhoun would likely result in the thermal shield contacting the snubber spacer blocks gradually. The blocks would not be subjected to a sudden loading by the entire thermal shield as was included in the calculation.

Impact loading due to movement of the thermal shield after it contacts the snubber spacer blocks has not been considered in this evaluation. The assumption has been made that plant personnel will detect the thermal shield motion and shutdown the plant before significant wear or fatigue stresses damage the reactor internal components.

Further evaluation was performed to determine what additional damage could result if the thermal shield came free from the support lugs and settled on the snubber spacer blocks. Since the final elevation of the thermal shield after it settles on the snubber spacer blocks is only 1 15/16 inches lower than the installed elevation, the slots in the thermal shield would still be engaged with the support lugs on the core support barrel. This engagement of the support lugs will severely limit the lateral displacement of the thermal shield. Assuming that the positioning pins are displaced radially outward and the support lugs are completely removed from at least half of the thermal shield after the shield settled on the snubber spacer blocks, the thermal shield lateral motion would be limited by the core support outer surface and the thermal shield inside diameter. This lateral motion could cause the thermal shield to contact and damage the surveillance capsule assemblies Figure (5), located in the annulus between the core support barrel and the reactor vessel. Because the thermal shield lateral motion is limited by the core support barrel, the themal shield will not contact the reactor vessel.

5.0 HYDRAULIC EVALUATION OF THERMAL SHIELD SUPPORT SYSTEM FAILURE

An evaluation was performed to determine the hydraulic consequences of the thermal shield dropping to the snubber blocks, as a result of the failure of the support system. The evaluation considered three different final resting

positions for the thermal shield.

The first resting position considered is that of the thermal shield sitting on the snubber blocks in an essentially concentric position. Since the thermal shield travel is only 1 15/16 in., the final axial position is relatively unchanged with respect to the distance that the flow must travel from the thermal shield exit to the core inlet plane. Therefore, if the thermal shield comes to rest in a concentric position, there will be no consequences on core inlet flow distribution . Further, the hydraulic resistance of the thermal shield region will be unchanged, so there will be no impact on system flow rate.

The second resting position considered is that of the thermal shield sitting on the snubber blocks in a fully eccentric position, Figure (6). To evaluate the impact of this position on the reactor hydraulics, an analytic model of the thermal shield was set up and solved by means of a computer code. The model consists of a network of parallel closed flow channels, with 12 channels being used to represent each thermal shield annulus (the inner and outer annuli). Each flow channel represents a 30° sector of the annulus as shown in Figure (6). The model starts at a station just upstream of the thermal shield and ends just downstream of the thermal shield, Figure (6). The pressure losses between these two stations for all flow channels are assumed equal.

The pressure losses in each flow channel consist of entrance, friction, and exit losses. Two sets of values for the entrance and exit loss coefficients were input to the cases. The first set, listed in Table 1, were calculated with standard methods for sudden contractions and expansions with the thermal shield treated as a single, average channel. The resulting loss coefficient values tend to be relatively small. The second set of loss coefficient values, also shown in Table 1, were obtained by taking maximum values for sudden contraction and expansion losses. The reason for using the range of entrance and exit loss coefficients is because it is difficult to pinpoint the appropriate contraction or expansion flow area ratios for the co-annular flow passages of the thermal shield region. Examining the results for the range of loss coefficient values provides a measure of the sensitivity of the results to the entrance and exit loss values.

The azimuthal variation of flow rate exiting from the thermal shield annuli is of interest because of its potential impact on the core inlet flow distribution. Also, the overall loss coefficient for the thermal shield region is important because of its impact on system flow rate.

Figures (7) and (8) show the resulting azimuthal variation of flow rate exiting from the inner, outer, and combined thermal shield annuli for the fully eccentric thermal shield position. The figures shows that while there are large azimuthal variations in flow rates for the individual annuli, there is a relatively small azimuthal variation in the flow rate for the combined annuli (the sum of the flow rates for the inner and outer annuli). The total variation for the combined annuli ranges from - 5 percent to + 13 percent about the average value for the smaller values of entrance and exit loss coefficients, Figure (7). For comparative purposes, measurements taken at the downstream end of a concentric thermal shield in a flow model of the Fort Calhoun reactor are shown in Figure (9). This figure shows that there are large azimuthal variations in the flow rates for the individual annuli, and an azimuthal variation for the combined annuli that is larger in magnitude compared to that calculated for the eccentric position of the thermal shield. Although there is the systematic variation observed in the flow rates exiting from the thermal shield region in the flow model, examination of the measured core inlet flow distribution does not show a matching azimuthal variation. The reduction in the azimuthal flow variation is attributed to the effects of the flow skirt, lower support structure bottom plate, and core support plate which lie in the flow path from the exit of the thermal shield region to the core inlet plane. Since the range in azimuthal variation of flow rate for the eccentric thermal shield position is smaller than that observed in the flow model, the impact of the eccentric thermal shield on core inlet flow distribution will be negligible.

The loss coefficient for the thermal shield region, when the thermal shield is displaced in the fully eccentric position, is slightly smaller than for the concentric position, as shown in Table 2. The reduction in loss coefficient is due to the azimuthal variation of the hydraulic diameter for the sectors of the two annuli when the thermal shield is eccentric. The hydraulic diameter

increases for those sectors (or flow channels) where the radial gaps and flow areas become larger due to the eccentricity; the hydraulic diameter decreases for the remaining sectors where the radial gaps and flow areas become smaller. As indicated in Table 2, the eccentric thermal shield position will result in a slight increase in system flow rate, about 0.2% of design flow rate (190,000 gpm).

The third resting position considered in the analysis is that of the thermal shield in a fully eccentric, tilted position, Figure (10). The upper end of the thermal shield touches the core support barrel, for example, at the 0° azimuthal position, while the bottom of the thermal shield contacts the core support barrel at 180°.

Computer cases were run for both sets of entrance and exit loss coefficients given in Table 1. The azimuthal variation of flow rate for the inner, outer, and combined annuli are shown in Figures (11) and (12). The maximum azimuthal variations for the combined annuli range from -31 to +26 percent about the average flow rate value, Figure (12). This range of variation is somewhat larger than that measured in the Fort Calhoun reactor flow model, -18 to +29 percent, Figure (9).

Results from the Ft. Calhoun reactor flow model tests also show that the lowest core inlet flow rates occur on the periphery of the core. It is believed that these low inlet flow rates to the peripheral fuel assemblies are due to the effects of the small turning radius associated with the 180° turn that these flows must make in going from the thermal shield region to the core inlet. As stated earlier, there does not appear to be a correlation between the azimuthal flow variations at the thermal shield exit and core inlet planes. Despite this lack of azimuthal correlation, in order to obtain some measure of the potential impact of the low flow rates exiting from the thermal shield region, it will be assumed that the lowest flow factors at the thermal shield exit and at the core inlet plane are directly related.

The lowest core inlet flow rate to a peripherally located fuel assembly measured in the Ft. Calhoun reactor flow model test is -12 percent relative to the average fuel assembly flow rate. From Figure (9), the lowest observed

azimuthal flow rate exiting from the thermal shield region is 18 percent below average. It may be estimated that the minimum azimuthal flow rate exiting from the fully eccentric, tilted thermal shield (31 percent below average) will produce reductions in the fuel assembly inlet flow rates of approximately twice that observed in the flow model tests, or about -24 percent relative to the average for the lowest flow assemblies. This suggests that the thermal shield dropping to a fully eccentric, tilted position will result in a flow decrease to the peripheral assemblies of 12 percent beyond that observed for those assemblies when the thermal shield is in a concentric position. It is further assumed that the limiting fuel assembly, from a thermal margin standpoint, is located at or near the periphery of the core; so the limiting fuel assembly may experience a flow reduction of up to 12 percent at the inlet.

The loss coefficient for the thermal shield region, when the thermal shield is in the fully eccentric, tilted position is larger than that for the normal, concentric position by a factor of up to 1.6, Table 2. The increase in hydraulic resistance results in a flow reduction of 2.2 percent of design flow (190,000 gpm).

To assess the combined effect of the reduction in inlet flow rate to the limiting fuel assembly (-12 percent) and in system flow rate (-2.2 percent), the conservative guideline is used that a 1 percent reduction in flow results in a 1 percent reduction in available overpower margin. Therefore, the impact of the thermal shield dropping to the eccentric, tilted position could, very conservatively, result in a total reduction in available overpower margin of about 14 percent. Since the available overpower margin to accommodate Anticipated Operational Occurrences is on the order of 17%, there is enough margin to accommodate the worst case resting position for the dropped thermal shield without violating the minimum DNBR limit. It is presumed here that the failure of the thermal shield support system and subsequent dropping of the thermal shield will be detected by the reactor operators, and that the reactor will be promptly shut down after detection.

6.0 SUMMARY

6.1 Mechanical Aspects

In the event the thermal shield support lugs and positioning pins fail to support the thermal shield, the shield will move downward until it contacts the snubber spacer blocks on the core support barrel. The distance from the bottom of the thermal shield to the top of the snubber spacer block is 1 15/16 inches. This is the maximum distance that the thermal shield can move in a downward direction. The stresses resulting from the thermal shield moving downward 1 15/16 inches and contacting snubber spacer blocks are within the normal operating allowable stresses.

The thermal shield could damage the surveillance capsule assemblies if the positioning pins in the thermal shield are displaced radially outward and the support lugs become unattached from the thermal shield. The thermal shield will not contact the reactor vessel.

Experience indicates that the thermal shield support system does not suddenly fail. The stainless steel of the support system and core support barrel exhibits a large capacity for absorbing elastic and plastic strains. Failure of the thermal shield support system, causing the shield to contact the snubber spacer blocks, will occur over a period of time. This period would allow early detection by plant personnel through the use of the loose parts monitoring system and/or neutron noise system, should the support system fail.

6.2 Hydraulic Aspects

The evaluation shows that if the thermal shield drops to the snubber blocks in an upright position, the impacts on core inlet flow distribution and on system flow rate are negligible. This result applies, regardless of the degree of eccentricity for the final resting position for the thermal shield.

If the thermal shield drops to a fully eccentric, tilted position, there may be total reductions in the inlet flow rate to the limiting fuel assembly of up to 14 percent due to impacts on the core inlet flow distribution and system flow

rate. The flow reduction, on a very conservative basis, is worth a reduction in overpower margin of 14 percent. The overpower margin available to accommodate Anticipated Operational Occurrences, 17 percent, is sufficient to offset the impact of the flow reduction arising from the thermal shield dropping to the most adverse resting position, without violating the core minimum DNBR limit.

Table 1

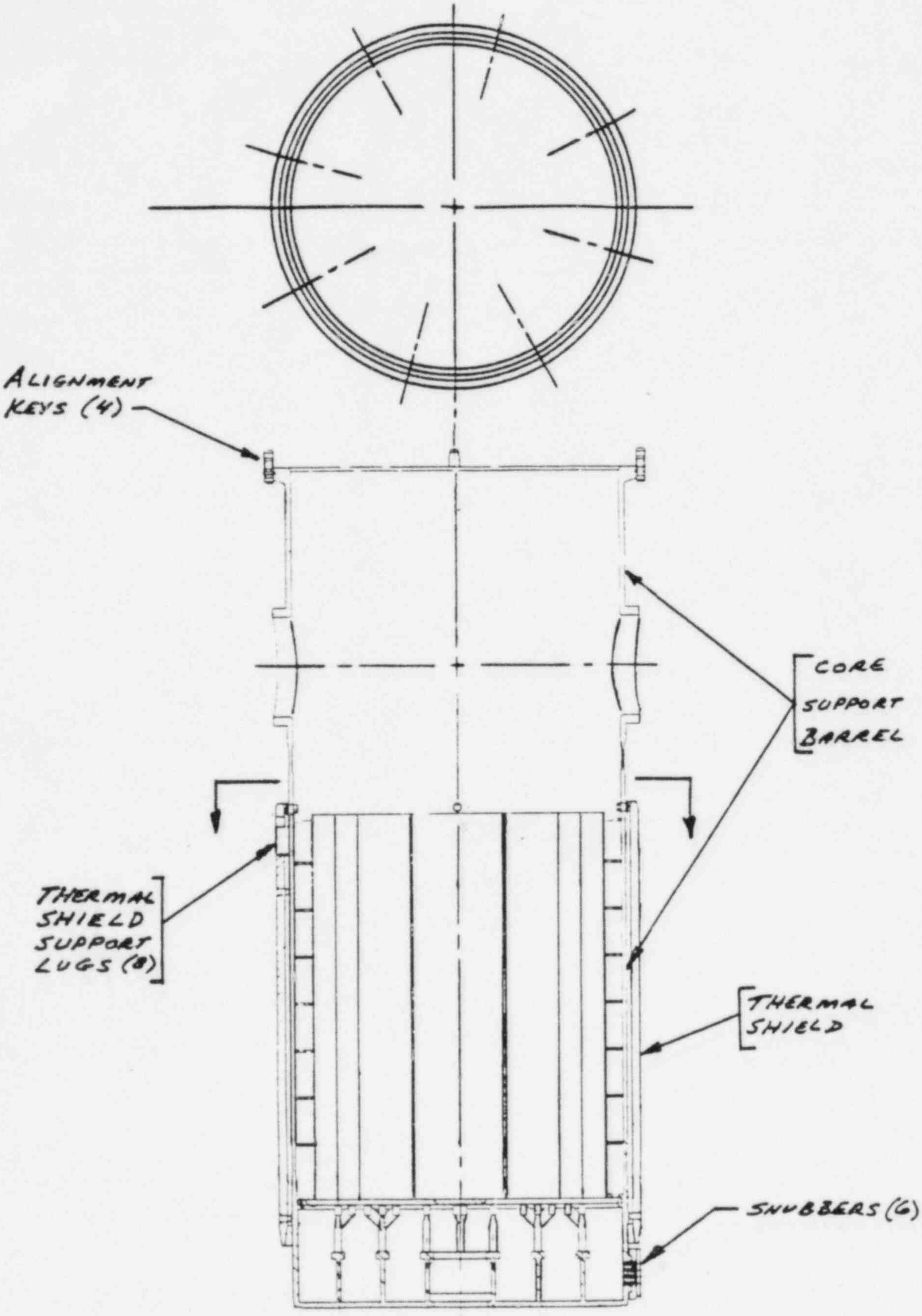
Entrance and Exit Loss Coefficients
for the Thermal Shield Region

Loss Coefficient Set	Entrance Loss Coefficient	Exit Loss Coefficient	Basis
1	0.168	0.055	Standard methods for sudden contraction and expansion losses, thermal shield region treated as a single flow channel
2	0.5	1.0	Standard methods for sudden contraction and expansion losses; maximum values used

Table 2

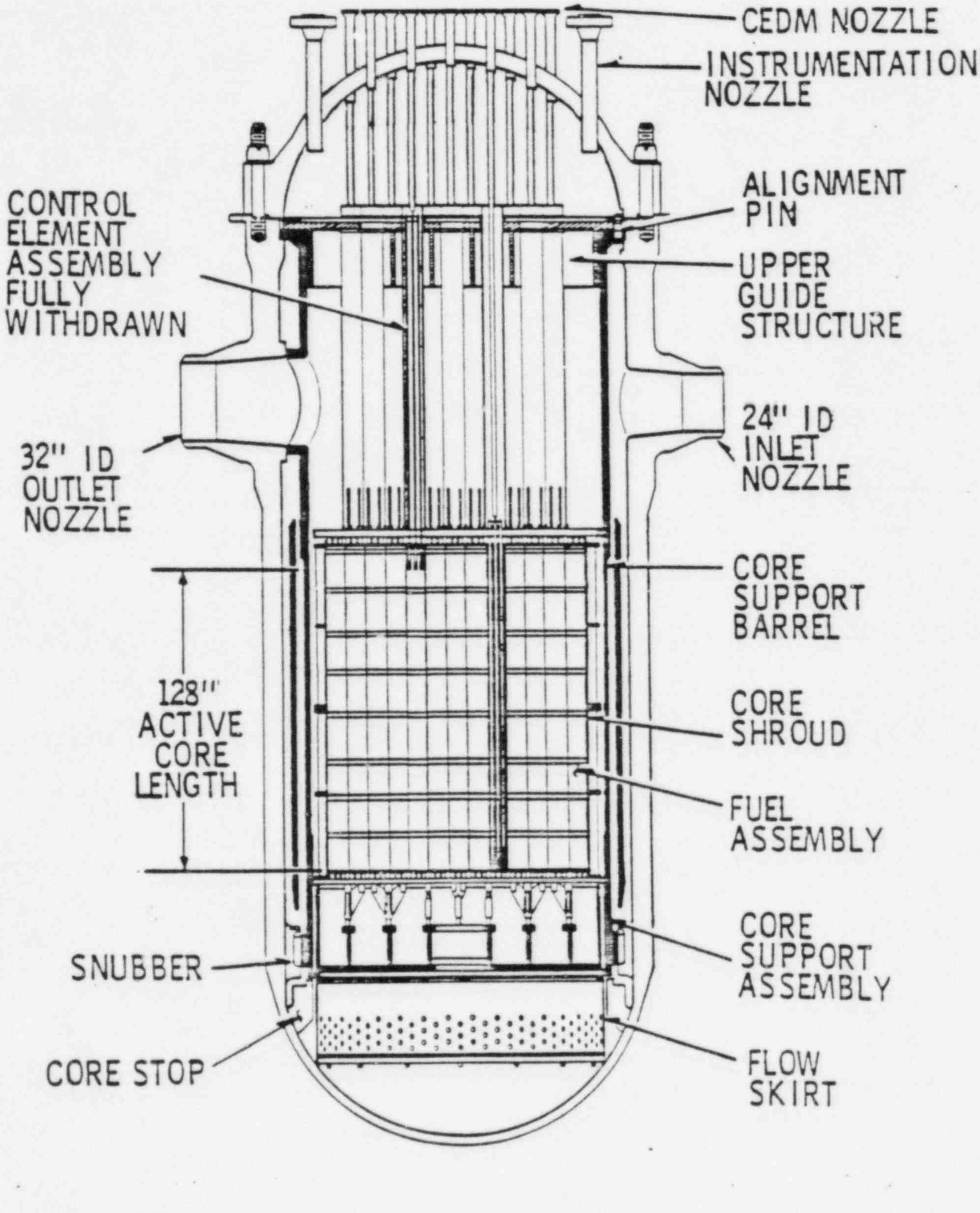
Impact of Thermal Shield
Position on System Flow Rate

Thermal Shield Position	Set of Entrance, Exit Loss Coefficients (Table 1)	Loss Coefficient for Thermal Shield Region K_{TS}	$\frac{K_T}{K_{T\ concentric}}$	Change in System Flow Rate % of Design Flow (190,000 gpm)
Concentric	1	0.64	1.00	0
Maximum Eccentricity, No Tilt	1	0.59	0.92	+0.2
Maximum Eccentricity, Maximum Tilt	1	0.87	1.36	-0.4
Concentric	2	1.92	1.00	0
Maximum Eccentricity, No Tilt	2	1.89	0.98	+0.2
Maximum Eccentricity, Maximum Tilt	2	3.06	1.59	-2.2

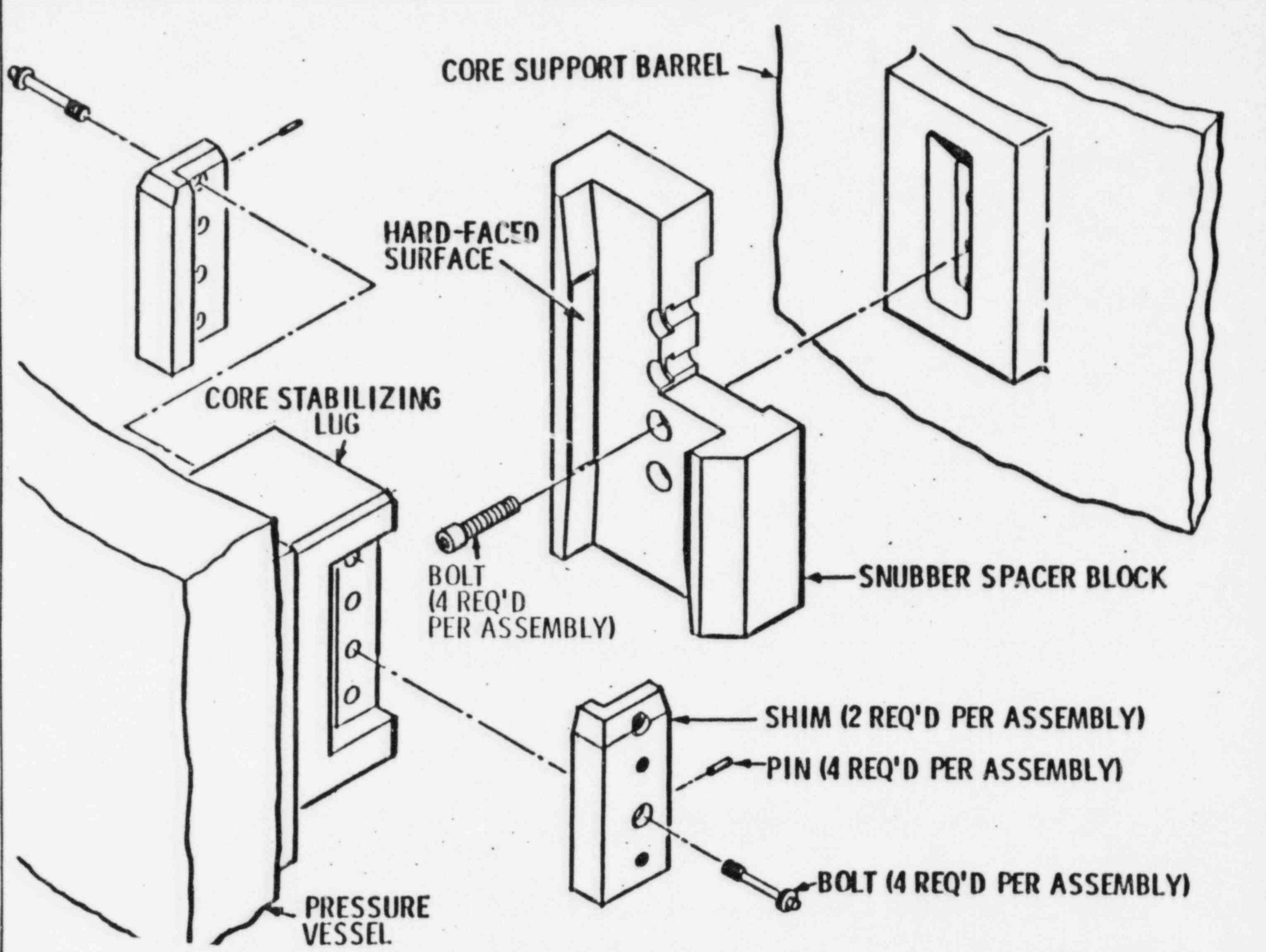


Core Support Assembly

Figure 1

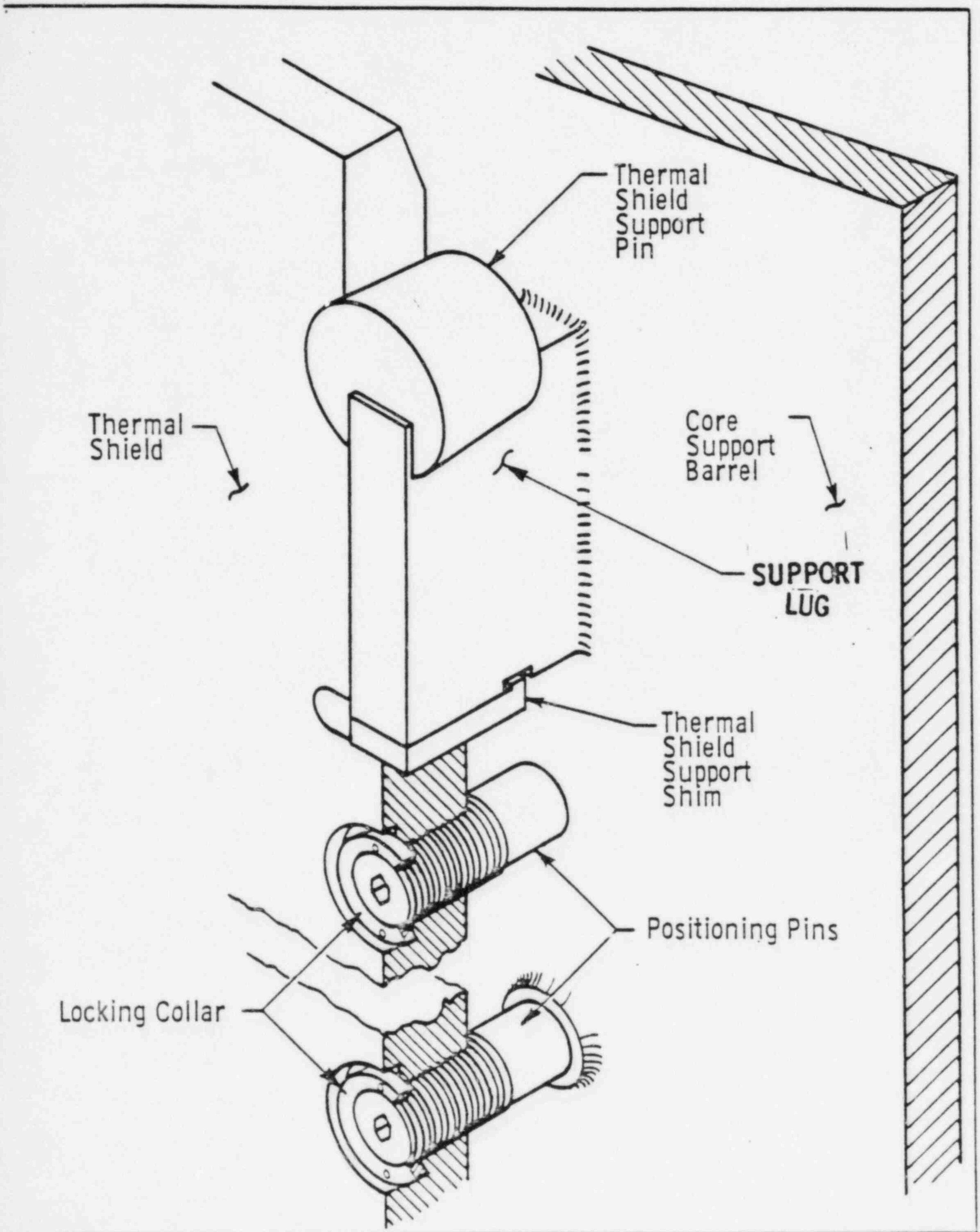


Reactor Internal Arrangement	Figure 2



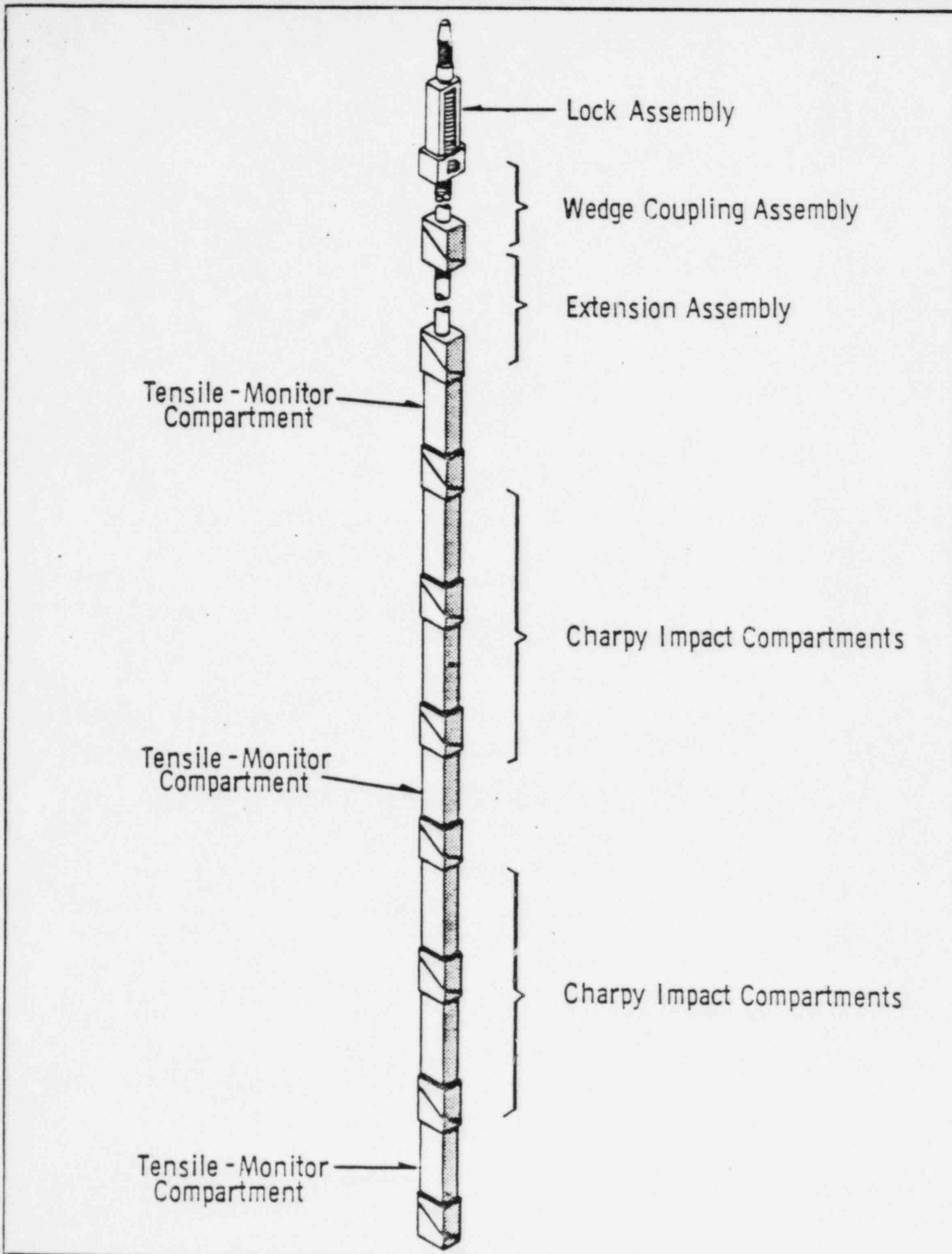
Pressure Vessel - Core Support Barrel Snubber Assembly

Figure 3



Thermal Shield Connection

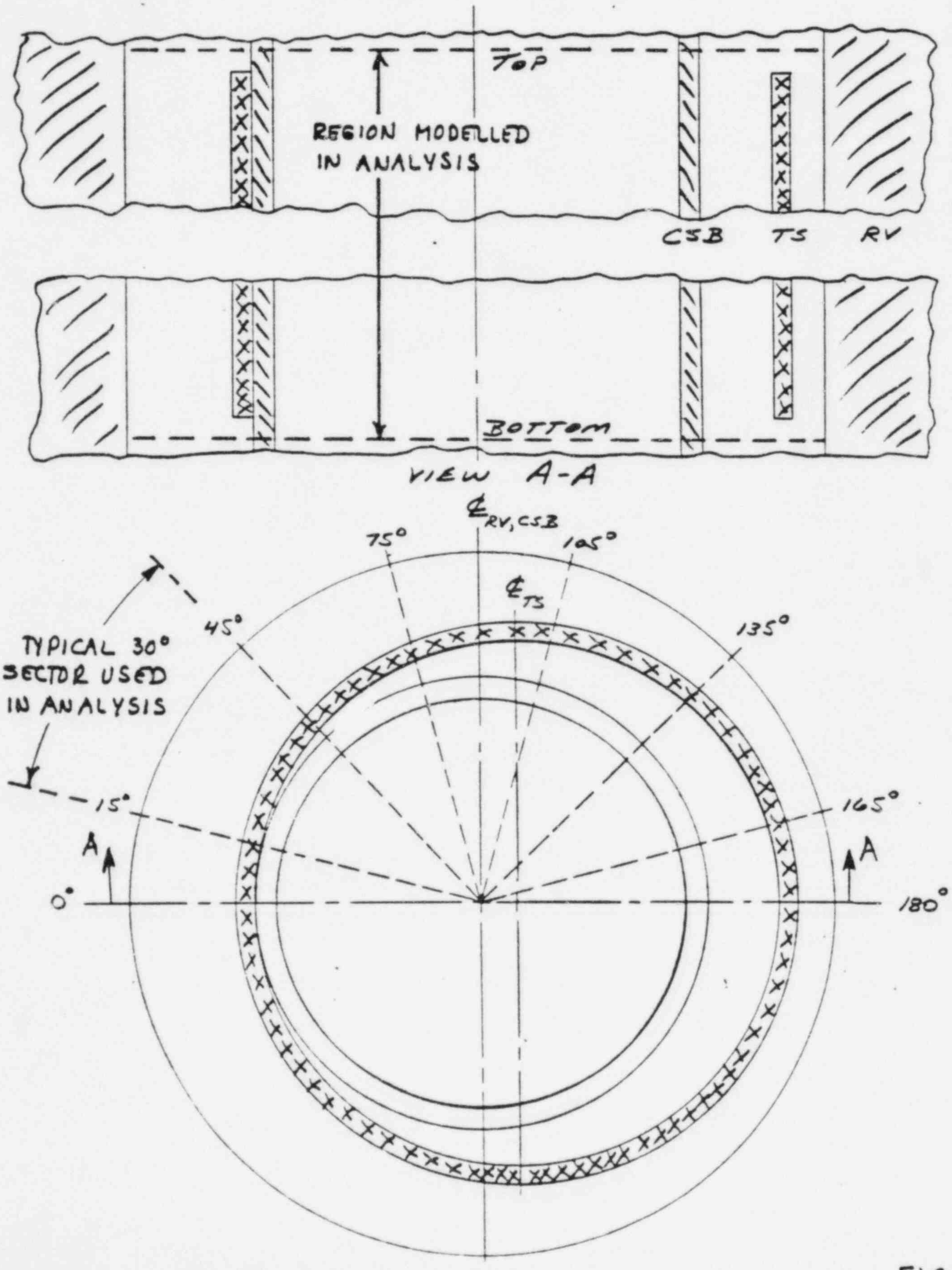
Figure 4



Typical Surveillance Capsule Assembly

Figure 5

Core Support Barrel – Thermal Shield
Configuration Model
Maximum Eccentricity, No Tilt



FIGURE

VARIATION OF LOCAL-TO-AVERAGE FLOWRATE RATIO

AT THE EXIT OF THE THERMAL SHIELD

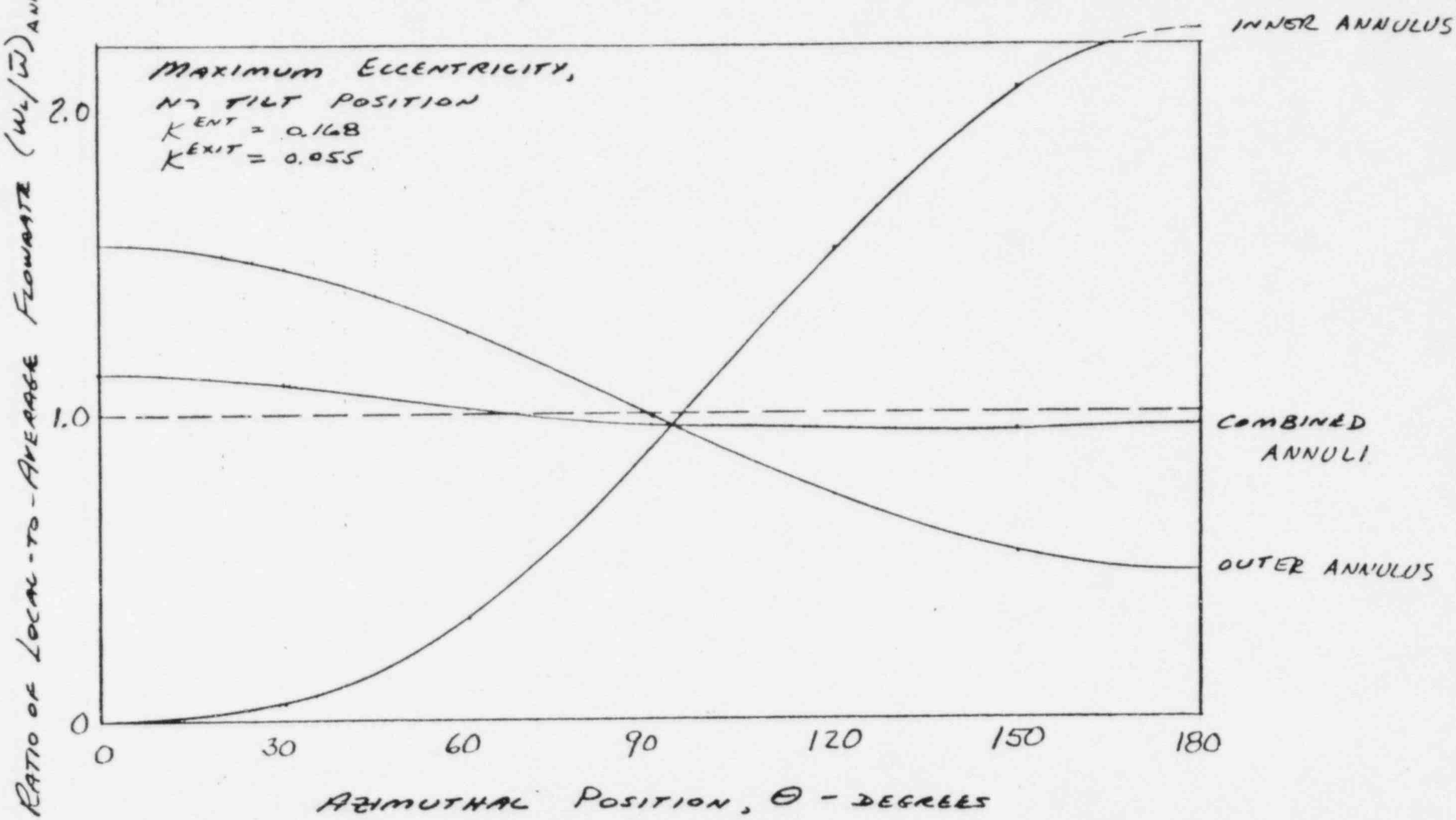
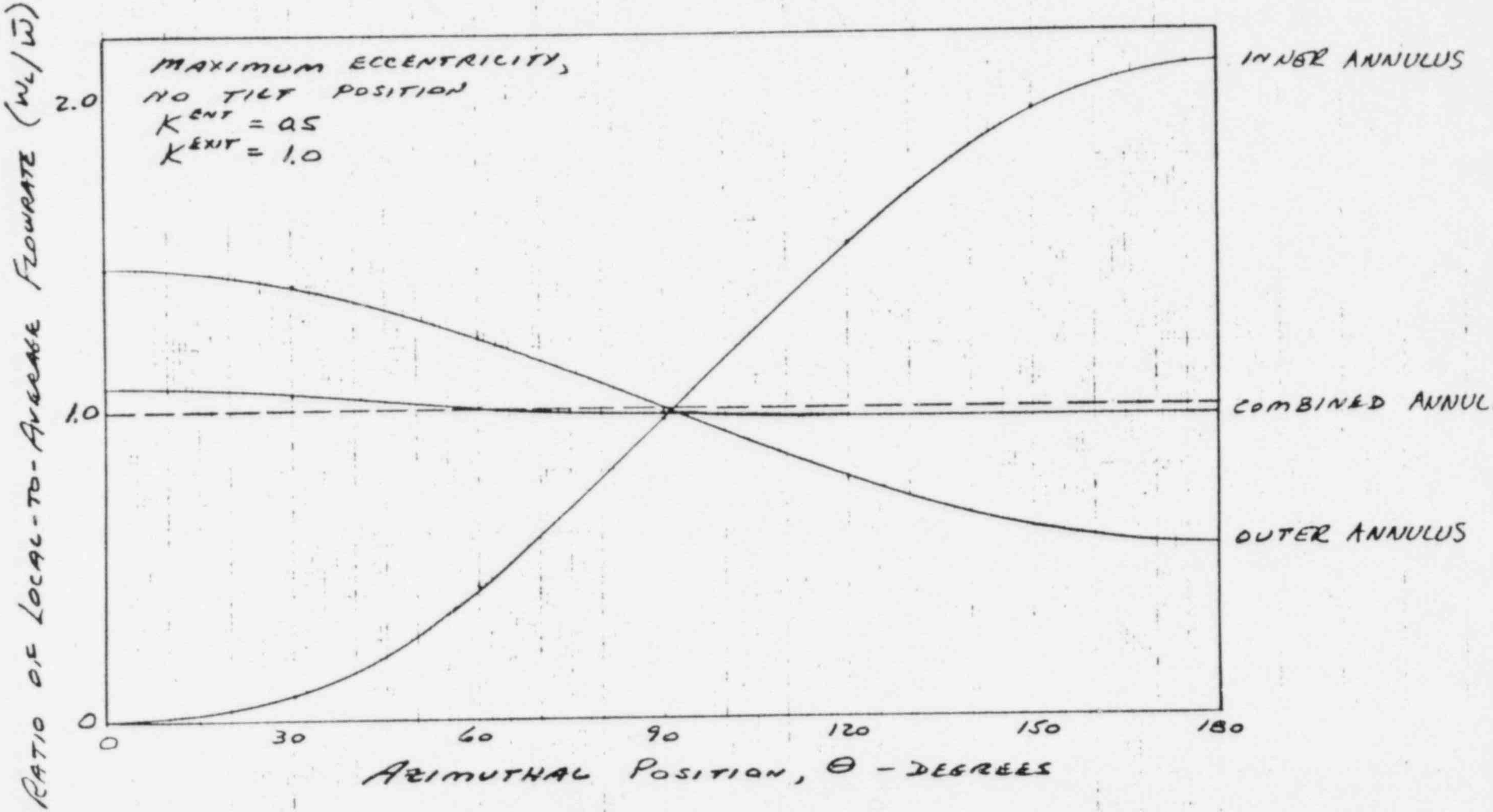


FIGURE 7

VARIATION OF LOCAL-TO-AVERAGE FLOWRATE RATIO
AT THE EXIT OF THE THERMAL SHIELD



FIGURE

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VARIATION OF LOCAL-TO-AVERAGE FLOWRATE RATIO
AT THE EXIT OF THE THERMAL SHIELD
SOURCE: OMAHA REACTOR FLOW MODEL TESTS

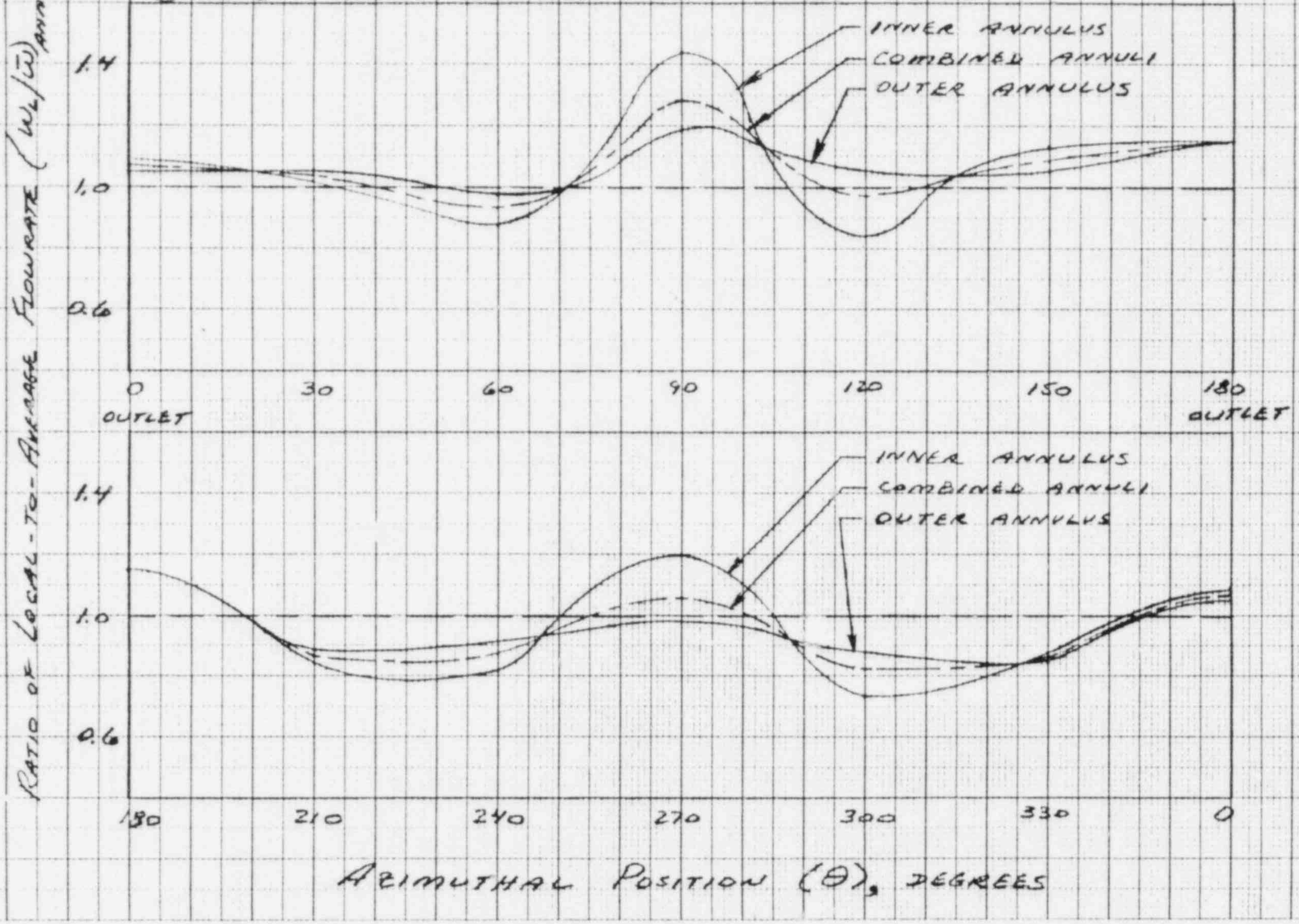
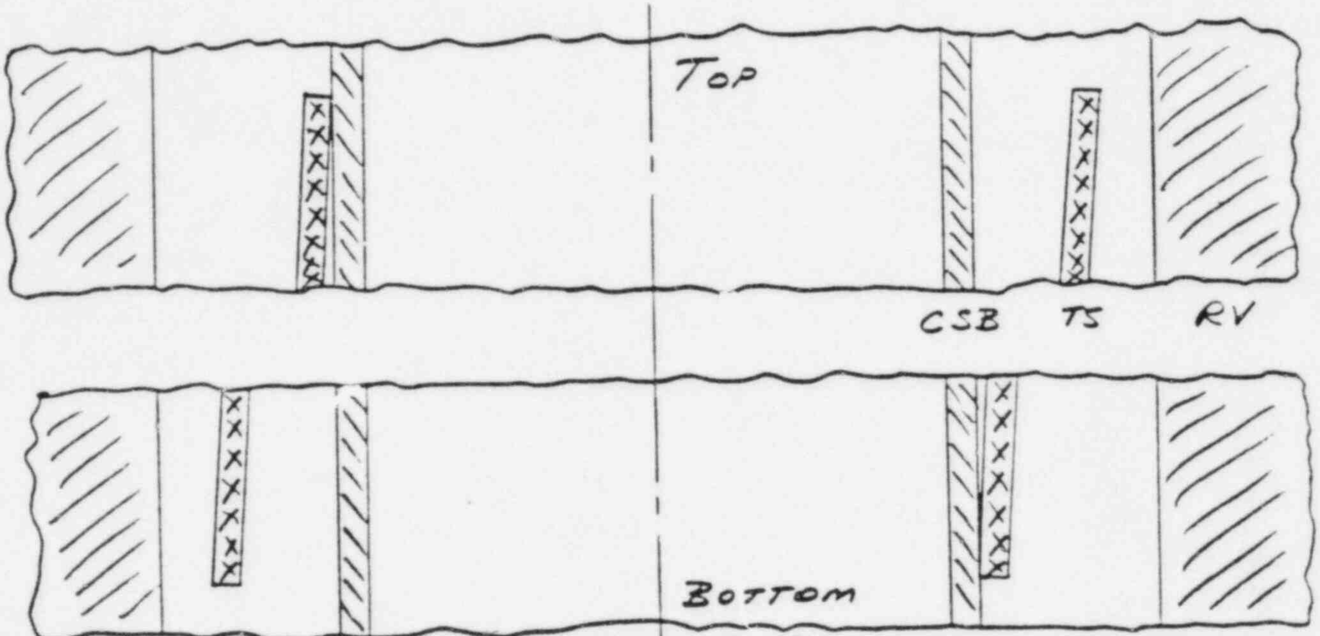


FIGURE 9

CORE SUPPORT BARREL – THERMAL SHIELD
CONFIGURATION MODEL
MAXIMUM ECCENTRICITY, MAXIMUM TILT



VIEW A-A

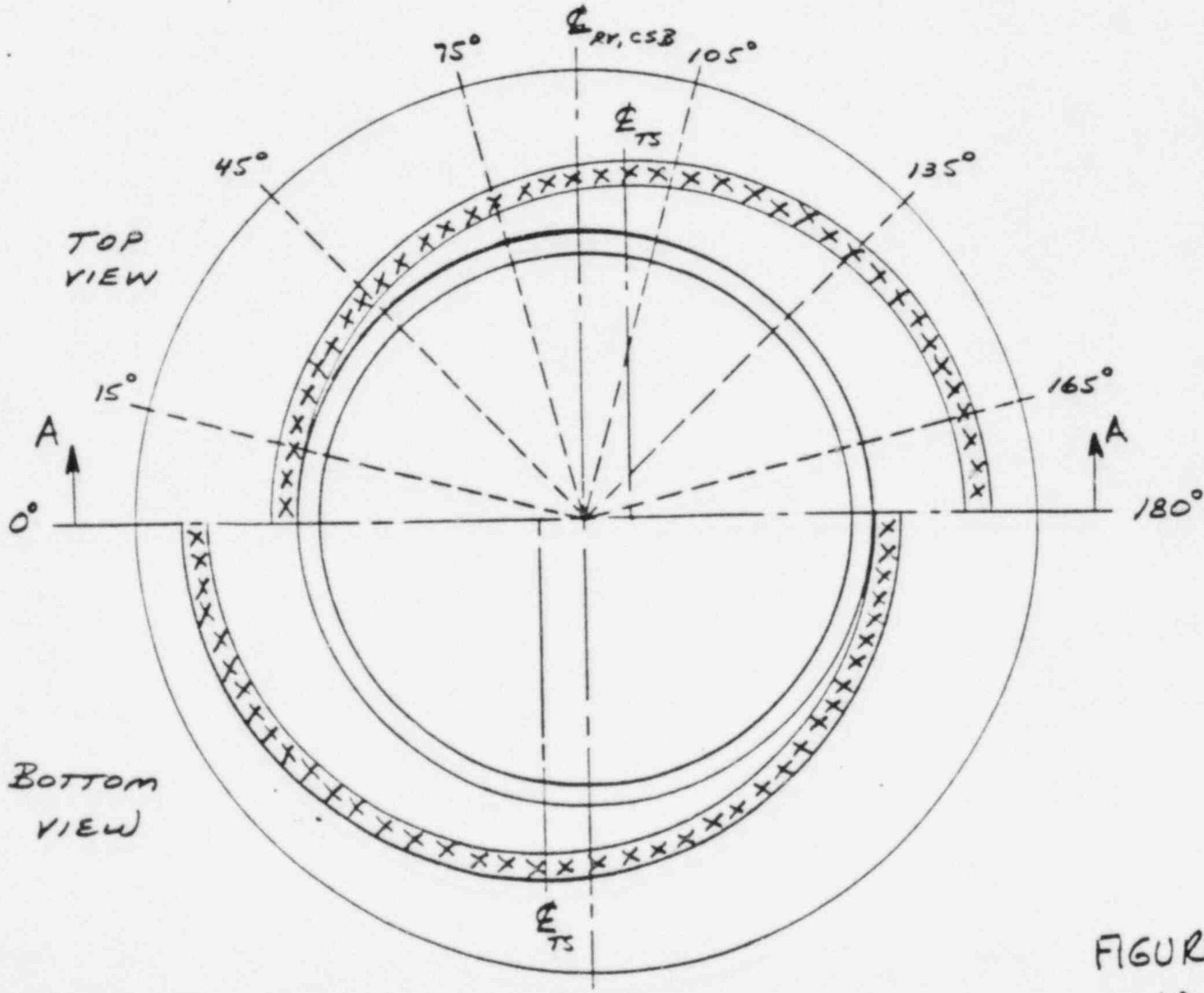
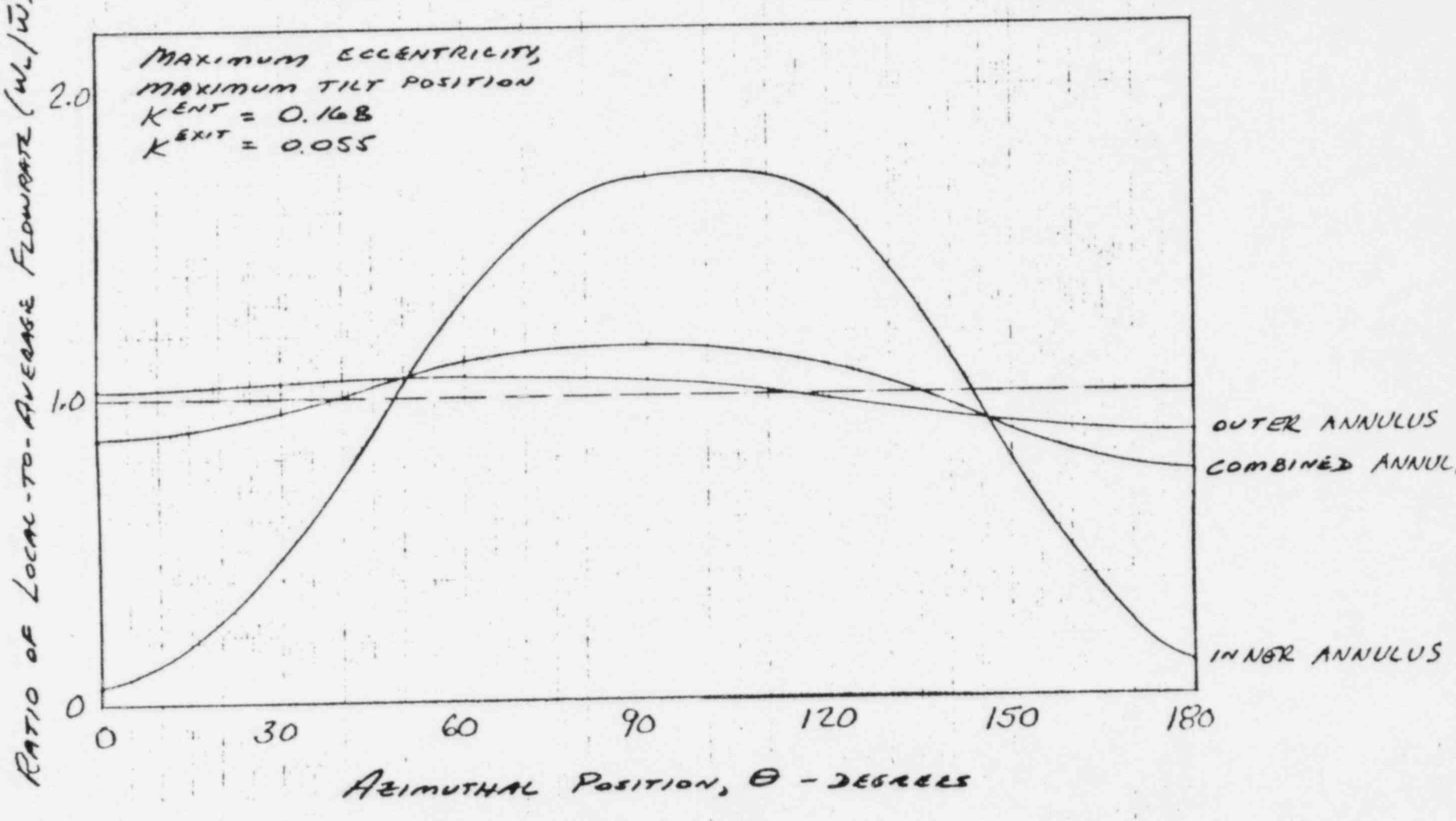


FIGURE 10

VARIATION OF LOCAL-TO-AVERAGE FLOWRATE RATIO

AT THE EXIT OF THE THERMAL SHIELD



FIGUR

VARIATION OF LOCAL-TO-AVERAGE FLOWRATE RATIO

AT THE EXIT OF THE THERMAL SHIELD

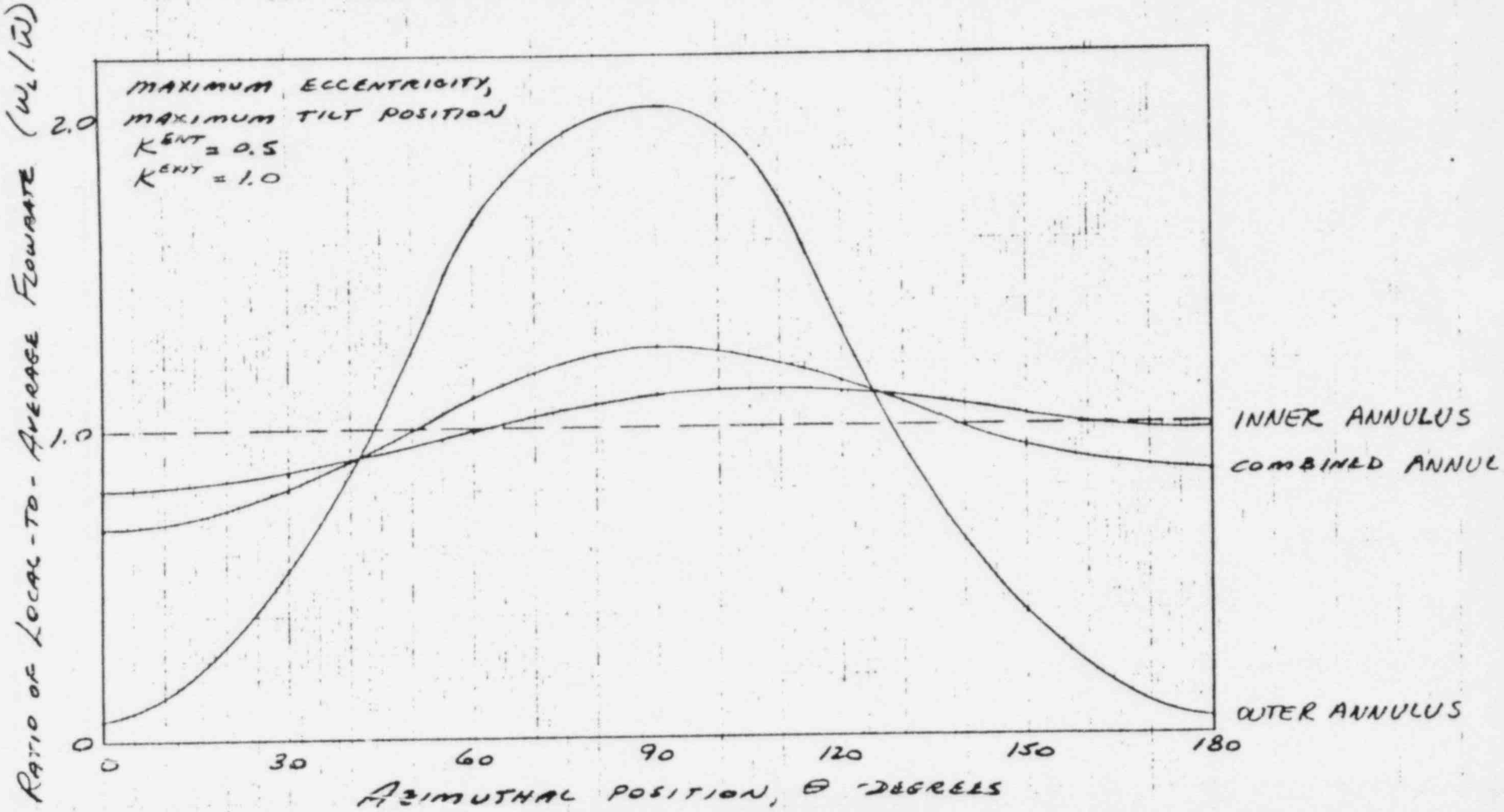


FIGURE 17