Enclosure 2

REACTOR VESSEL CORE SUPPORT CONE

STRUCTURAL INTEGRITY

Presented in Response to Questions Raised
at ACRS Meeting of August 18, 1982

Also

MEB Item 72; Letter HQ:S:82:143, J.R. Longenecker to P.S. Check, "Meeting Summary: November 22-24, 1982, Mechanical Engineering Branch/Clinch River Breeder Reactor Plant Meeting," dated December 14, 1982

ADVANCED REACTORS DIVISION
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INTRODUCTION

A meeting was held on August 18, 1982 in Washington, D.C. to present the design of system components for elevated temperature service as well as several other topics to the ACRS. The design of the reactor vessel was included in the agenda since it is exposed to elevated temperature service. During the presentation, several questions were asked concerning the core support cone to core support and vessel attachments and the consequences of attachment failure. The concern, in particular, was for gross failure of the weld joint of the cone to core support plate resulting in the core support structure and core dropping away from the control rod into the inlet plenum. The example cited was BWR experience with intergranular corrosion of the support attachments.

This paper addresses the ACRS concerns and shows that the CRBRP Reactor Vessel core support cone is structurally adequate without additional backup supports or routine in-service inspection. The paper addresses the design of the weld joints, stress levels and conservatisms in the design loadings and stress analysis.

2. Core Support Cone Thermal Environment

The core support cone is in a benign thermal region of the reactor vessel located between the inlet plenum and the reactor vessel/core barrel annulus. In this region, the design temperature during full power steady state operation is 775 °F and the design thermal transients are relatively benign. There is conservatism in these design values because this steady state temperature exceeds expected temperatures by as much as 75°F and the design transients are correspondingly more severe.

Transient temperatures in the inlet plenum were calculated using results from an empirical equation for inlet plenum mixing. The mixing constants at different locations within the plenum were determined from two loop and three loop mixing tests performed at HEDL in the one quarter scale Inlet Plenum Feature Model.[1]

The plant duty cycle event transients were analyzed and three events were selected as umbrella transients for the core support cone design. These three events are as follows:

1. N-4A (up) - Plant loading from 40% to 100% power.
2. U-2e - Plant loading at maximum rod withdrawal rate - from 40% to 100% power.
3. F-4A - Saturated steam line rupture.

The first two of these events are symmetric and the third is asymmetric. The transient boundary temperatures at the bottom of the support cone are illustrated in Figures 3 through 5 for the three umbrella events. The location on the support cone where these temperatures apply is shown on Figures 6 and 7. The transients that occur in the reactor vessel core barrel annulus are similar to the inlet plenum transients but they are mitigated as a result of flowing through the core support plate. Thus, the transients above the support cone are even less severe than those below and the temperature difference across the support cone is small.

Reference 1) HEDL-TME 76-33, PM McConnell, et. al., "Inlet Plenum Feature Model Flow Tests of the Clinch River Breeder Reactor: Addendum V Results," March 1976.

3. Core Support Cone Materials Considerations

The operational environment of the weld between the core support structure (CSS) and the reactor vessel cone (RVC) is benign and thus minimizes concern for in-service materials related problems.

The design temperature of 775°F is sufficiently low that metal loss due to sodium corrosion will be virtually non-existent. Furthermore, at the very low flow rates involved, no allowances are necessary for erosion losses. The only sodium-related change in the material will be some slight surface carburization, possibly to a carbon level of about 2000 ppm. However, due to the low carbon diffusion coefficient at 775°F, the carburized layer is unlikely to exceed 3 mils in thickness and thus constitutes a very small percentage of one total cross-section thickness.

In addition to effects due to sodium exposure, consideration was also given to the possibility of neutron-radiation-induced embrittlement. The low temperature involved (775°F) assures that no helium embrittlement will be experienced and radiation effects will be restricted to displacement damage. The support cone displacements per atom (dpa) are 0.0001 maximum. Generally, the displacement threshold at which radiation effects such as ductility loss and fracture toughness begin to be observed in the austenitic stainless steels and their weldments is considered to be 0.5-1.0, dpa. Below this value, radiation effects may be ignored. Since the predicted displacement level is nearly four orders of magnitude below this threshold value, no radiation damage is foreseen for the CSS/RVC weldment.

Consideration has also been given to the possibility of material property changes arising from prolonged operation at 775°F. Thermal aging is known to be detrimental to the austenitic stainless steels and their weldments, particularly when a sigma phase is produced and severe embrittlement results. However, the operating temperature of the CSS/RVC weldment area (775°F) is much lower than that required for sigma formation ($\sim$1000°F mininum) and no embrittlement is expected to result from this reaction. The other effect of thermal aging is to induce carbide precipitation, leading to increases in tensile and yield strengths, and associated losses of ductility. Again, although some carbide formation is likely at the CSS/RVC weldment, the low operating temperature will ensure that this reaction will remain very localized at the surface and thus will not produce any measurable changes in mechanical properties over the plant lifetime.

4. Reactor Vessel/Core Support Cone Stress Assessment

This section presents a summary of the stress levels and fatigue damage at the upper and lower weld joint areas of the core support cone (Fig. 8, Sections B-B and A-A, respectively).

The summary of stress analysis results on the core support cone are shown in Tables 1 and 2. Section A-A has the minimum factor of safety [allowable stress/calculated stress] of 1.186 for an upset load combination of OBE plus thermal. This factor is a comparison of the primary plus secondary stress intensity range with the appropriate criteria for this stress intensity. Section B-B has a minimum factor of safety of 1.238 for a faulted load combination of SSE plus pressure. This factor is a comparison of membrane plus bending stress intensity with appropriate stress intensity limits from Code Case 1592. Section A-A is the limiting section for primary plus secondary stress intensity limits and fatigue damage, while Section B-B has the lower margins for primary stress intensity limits.

Analyses of the core support cone were performed for both mechanical and thermal loading conditions. The conservatisms in the analyses for each type of loading will be addressed in the following paragraphs.

The components were analyzed to the ASME Code for Nuclear Components which contains margins of safety, such as using minimum properties for the physical properties of materials, combined with a factor of 3 on tensile strength, and a limit of 2/3 on yield strength for primary stress intensity, and for fatigue limits, a factor of 20 on cycles and 2 on stress. Additional checks are required which include insuring "shake down" and preventing "thermal ratcheting". The applicable ASME Code Edition, Code Cases and RDT Standards are listed in the PSAR.

PRIMARY STRESSES

The mechanical loadings consist of differential pressure, dead weight and seismic. The following is a list of the conservative approaches used in the evaluation of the primary stresses due to the mechanical loadings. These approaches apply to both Sections A-A and B-B unless noted otherwise. Other areas were analyzed using standard techniques defined in the structural criteria documents.

1. The stresses due to differential pressure (Δp) were calculated using 170 psi (design Δp), while the maximum operating pressure is 139 psi.

2. The dead weight stresses are in the opposite direction to the Δp stresses. In the evaluation they were either ignored or the absolute value was added to the pressure stresses. However, this is a small effect because the dead weight (DW) stresses are small compared to the Δp stresses. The largest ratios of DW/Δp stresses are obtained for the membrane stresses. Tables 1 and 2 show the relative magnitude of the stresses produced by these two loading conditions.

3. The total seismic stresses were calculated for Section A-A by direct summation of the stress intensities due to the North-South (N-S), East-West (E-W) and Vertical seismic loadings. There are two conservative aspects to this approach. On an axisymmetric structure, the combination of the N-S and E-W directions by the appropriate square root of the sum of the squares (SRSS) method implies that the combined stress cannot exceed the maximum stress from either of the two seismic directions. In addition, the direct summation of the vertical component with either of the horizontal componants is conservative because the combination should be done by SRSS method. Additional conservatism arises from working with stress intensities instead of stress components.

4. The total primary stress intensities were obtained by adding the stress intensities of the different mechanical loadings in absolute fashion. A less conservative, but acceptable approach would be to perform the combination (with appropriate signs) on the stress component basis.

5. The stress allowables at Section A-A and B-B are based on the High Temperature Code Case 1592 at 850°F, while the maximum operating temperature is 750°F and the design temperature is 775°F. The reasons for this conservatism is that the structure will be subjected to temperatures above 800°F (≈ 835°F for A-A and 810°F for B-B) for a limited time (about 10 hours in the reactor life) during some thermal transients.

6. The Reactor Vessel was designed and analyzed to a conservative seismic spectra. The seismic spectra which are present in the RV specification were later superceded by spectra which reduced g-levels by approximately 25%. This reduction would result in a proportionate reduction in seismic stresses at the core support cone Section B-B.

7. The structural evaluation of Section B-B was based on a 4 1/2" thick core support cone. The actual minimum thickness of plate used was 4.97". Additional analyses showed that, in general, the primary stresses are less than those used in the results for the 4 1/2" thickness and the stresses due to thermal transients remained approximately the same in both cases.

SECONDARY MEMBRANE PLUS BENDING AND FATIGUE STRESSES

These criteria are generally dominant for cyclic thermal loadings. In the analyses, the evaluated primary plus secondary stress intensity ranges ($P_L + P_b + Q$) satisfy the $3S_m$ stress limit at Sections A-A and B-B in the Core Support Cone. The minimum margin due to this stress category is 1.186 at Section A-A. Fatigue damage on the core support cone has been considered and is maximum at Section A-A. The total fatigue damage is 0.06 based on the ASME Code criteria. Fatigue damage at Section B-B is <0.1 and is not controlling on the cone.

In summary, the demonstrated factors of safety on primary and primary plus secondary stresses are satisfactory and meet the ASME high temperature Code criteria but could be increased significantly by reducing the conservatisms included in the analysis, such as using the SRSS combination for seismic loads, using actual cone thickness, using stress limits for 750°F, using the operating instead of design differential pressure, using new seismic spectra and adding stresses component by component before calculating total stress intensities.

TABLE 1

SUMMARY OF CSS/CONE PRIMARY STRESS RESULTS

(Section A-A)

Loadings or Criteria	Load or Stress Category	Primary Stress Intensities (psi)		Primary plus Secondary Stress Intensities (psi)
		Membrane	Membrane + Bending	
Loadings	Pressure	3054	5602	5602
	Dead Weight (1)	390	351	351
	OBE (2)	3189	3757	3757
	SSE (2)	4502	5293	N/A
	Thermal Stress Range (Linear)	--	--	37731
Normal + Upset	P_m (3)	6243	--	--
	$P_1 + P_b$ (3)	--	9359	--
	$(P_1+P_b+Q)_r$	--	--	41488 (6)
	Allowable (4)	S_{mt} = 14800	K_tS_t = 21230	$3S_m$ = 49185
	FS (5)	2.37	2.27	1.186
Faulted (SSE + ΔP)	P_m	7558	--	--
	$P_1 + P_b$	--	10895	--
	Allowable	$1.2S_t$ = 23160	$1.2K_tS_t$ = 25500	--
	FS	3.06	2.3	--
Fatigue	Σ n/n	0.06	< 0.9 allowable per T-1435 of CC1592	

(1) Dead weight has opposite sign to other stresses. It is conservatively excluded from the evaluations.

(2) The seismic loads are the summation of N-S, E-W and Vertical.

(3) Normal + upset (OBE + ΔP) are evaluated to Code Case 1592 allowable.

(4) Allowables at 850°F. This is conservative since the temperatures are below 800°F at this region.

(5) Factor of safety = $\frac{\sigma \text{ allowable}}{\sigma \text{ calculated}}$

(6) $(P1 + Pb + Q)_{range}$ = OBE + Q_{range} thermal (pressure decays before transient is severe)

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TABLE 2

SUMMARY OF RV/CONE PRIMARY STRESS RESULTS

(Section B-B)

Loading or Criteria	Load or Stress Category	Primary Stress Intensities (psi): Membrane	Primary Stress Intensities (psi): Membrane + Bending	Primary plus Secondary Stress Intensities (psi)
Loadings	Pressure	3800	11100	11100
	Dead Weight (1)	1100	1100	1100
	OBE (2)	2600	4200	4200
	SSE (2)	5100	8400	N/A
	Thermal Stress Range (Linear)	--	--	(6)
Normal + Upset (OBE + ΔP)	P_m	6400	--	--
	$P_1 + P_b$	--	16400	--
	$(P_1+P_b+Q)_r$	--	--	$<3S_m$ (6)
	Allowable (3)	$S_{mt} = 14800$	$K_tS_t = 21230$	$3\bar{S}_m = 49185$
	FS (4)	2.31	1.295	>1.0 (6)
Faulted (SSE + ΔP)	P_m	8900	--	--
	$P_1 + P_b$	--	20600	--
	Allowable	$1.2\ S_t = 23160$	$1.2K_tS_t = 25500$	--
	FS	2.6	1.238	--
Fatigue	Σ n/n	< .1 (6)	< 0.9 allowable per T-1435 of CC1592	

(1) Dead weight has opposite sign to other stresses. It is conservatively excluded from the evaluations.

(2) Seismic loads are the combination of N-S, E-W and Vertical.

(3) Allowables at 850°F per CC1592 as in Table 1 (for conservatism).

(4) Factor of safety $= \frac{\sigma \text{ allowable}}{\sigma \text{ calculated}}$

(5) $(P_1 + P_b + Q) = \text{OBE} + Q_{range}$ thermal (pressure decays before transient is severe)

(6) This result is based on a scoping evaluation using current design transients, which are less severe than those used at the time the component design analysis was performed. The scoping study established the magnitude of the factor of safety and fatigue damage summation but precise values are not available.

SUMMARY

1. The reactor internals are supported by the reactor vessel core support forging which is attached to a 5 inch thick cone which is in turn welded to the core support structure. Welding was performed in accordance with ASME Code requirements as supplemented by RDT standards. The welds were accepted based on meeting radiographic and liquid penetrant examination requirements contained in the ASME Code and applicable RDT Standards.

2. The pressure loading (normal operating condition) is upward resulting in compressive loads on the cone which tends to avoid creation of potential flaws.

3. The weld joint designs are such as to position the weld joints away from the transition region between the cone and the 24 inch thick core plate and the vessel core support forging and the cone.

4. The thermal environment in the vicinity of the reactor vessel core support cone is benign.

5. The material of construction is austenitic stainless steel which has excellent retention of ductility throughout the 30 year life of the reactor.

6. The neutron radiation induced embrittlement is neglegible as are other environmental effects on the material.

7. The stresses in the vicinity of the welds are low and substantial margins are available based on conservative analyses. The primary plus secondary stresses for normal plus upset loads have been shown to be less than $3S_m$. The fatigue damage summations are < 0.1.

8. Six locations are provided through the Horizontal Baffle Assembly baseplate which provide access to the annulus between the reactor vessel and the core barrel including the upper surface of the cone, if a future need for such access is identified.

CONCLUSION

Based on the information presented, it is concluded that the core support cone has been designed and fabricated in a manner that precludes the need for routine in-service inspection. However, capability for future inspection of the core support cone is provided and can be utilized to the extent practicable by the development of under sodium in-service inspection techniques.

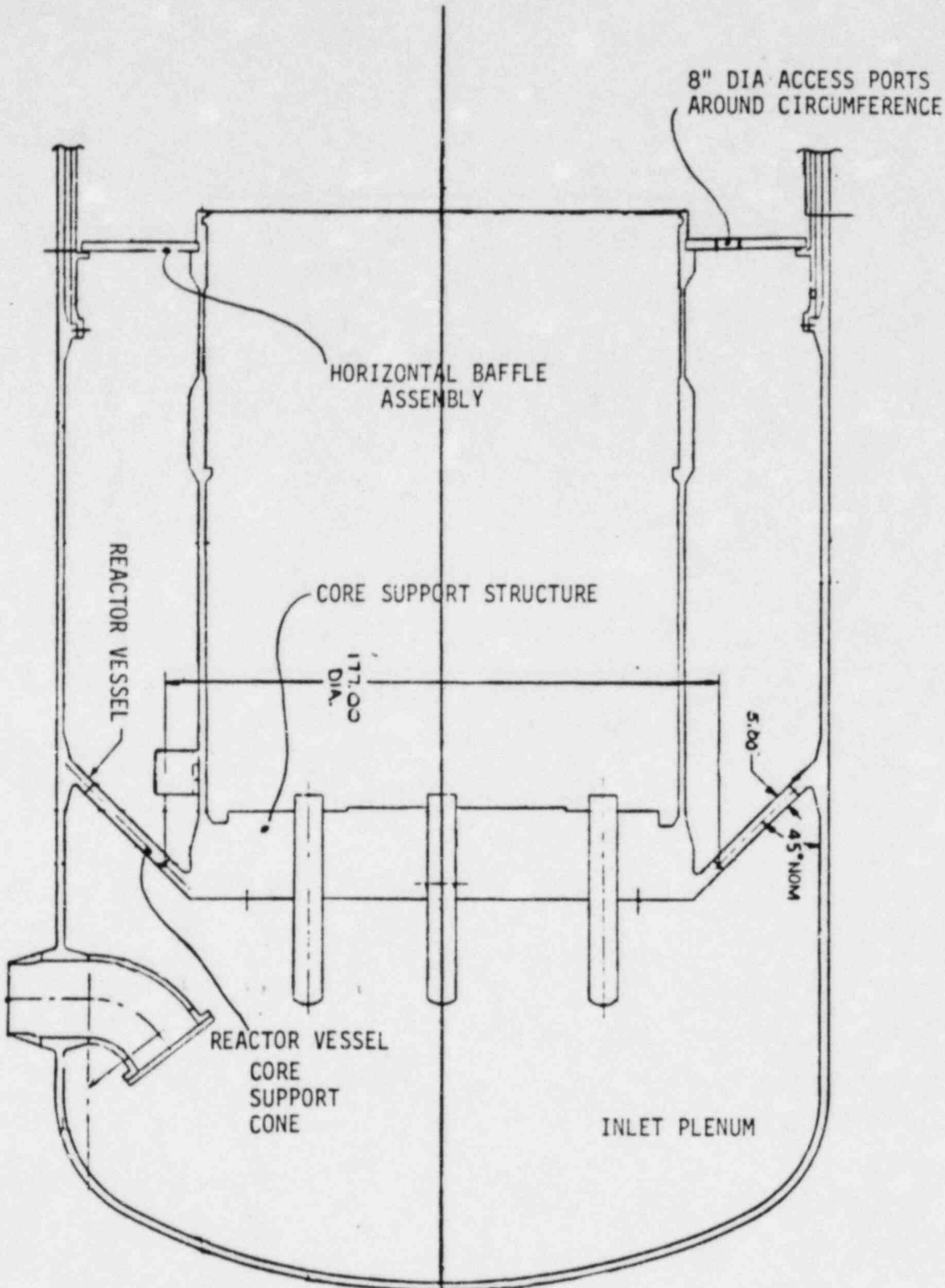


Figure 1

REACTOR VESSEL/CORE SUPPORT STRUCTURE ASSEMBLY

SKETCH SHEET
WESTINGHOUSE FORM 2444D

WESTINGHOUSE ELECTRIC CORPORATION

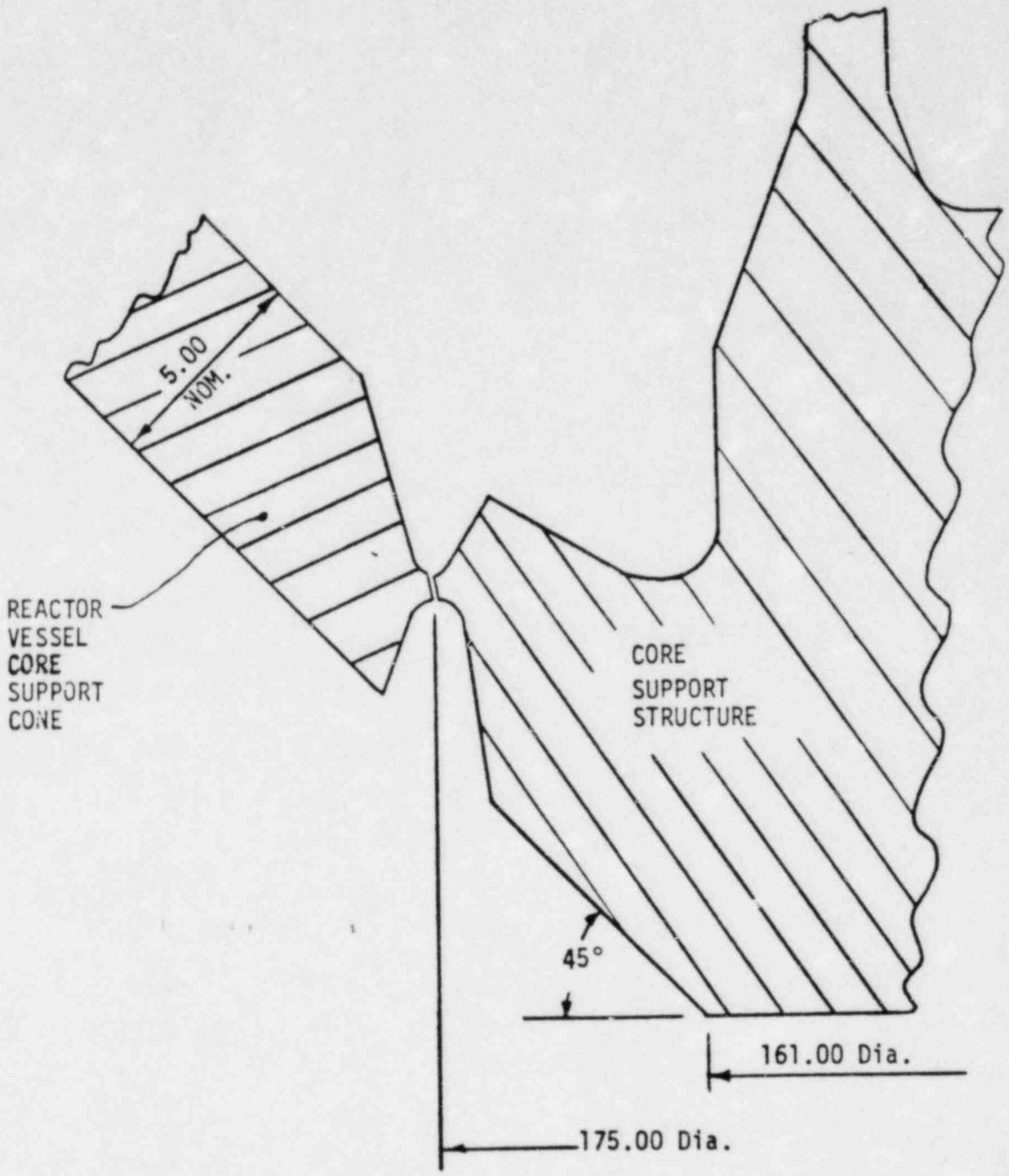


Figure 2

REACTOR VESSEL CONE CORE SUPPORT STRUCTURE WELD PREPARATION DETAIL

INLET PLENUM SODIUM TEMPERATURE RESPONSE
DURING A N-4A PLANT LOADING BETWEEN 40% AND 100% POWER

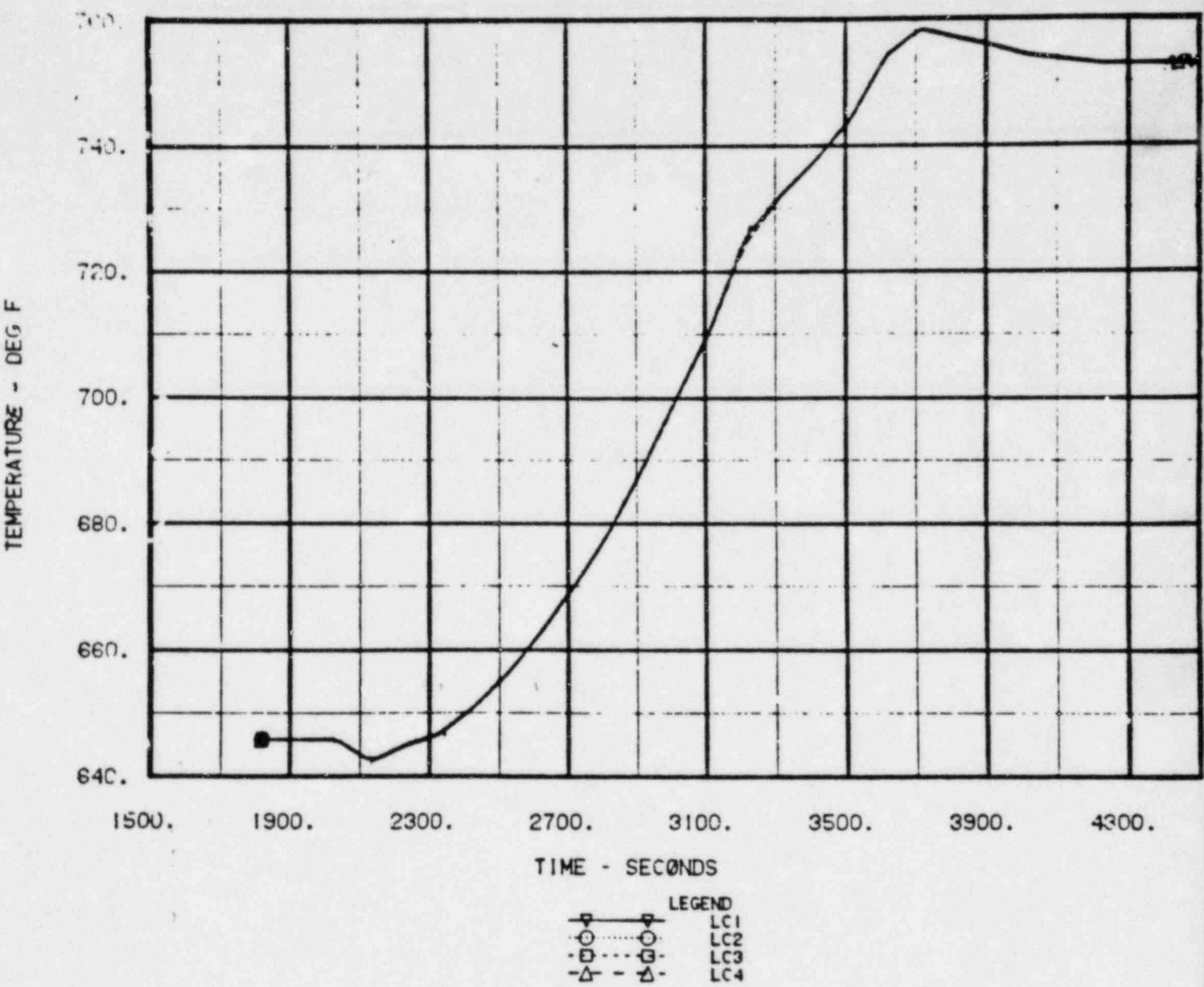


Figure 3

INLET PLENUM SODIUM TEMPERATURE RESPONSE

DURING A U-2E PLANT LOADING AT MAXIMUM ROD WITHDRAWAL RATE
BETWEEN 40% AND 100% POWER

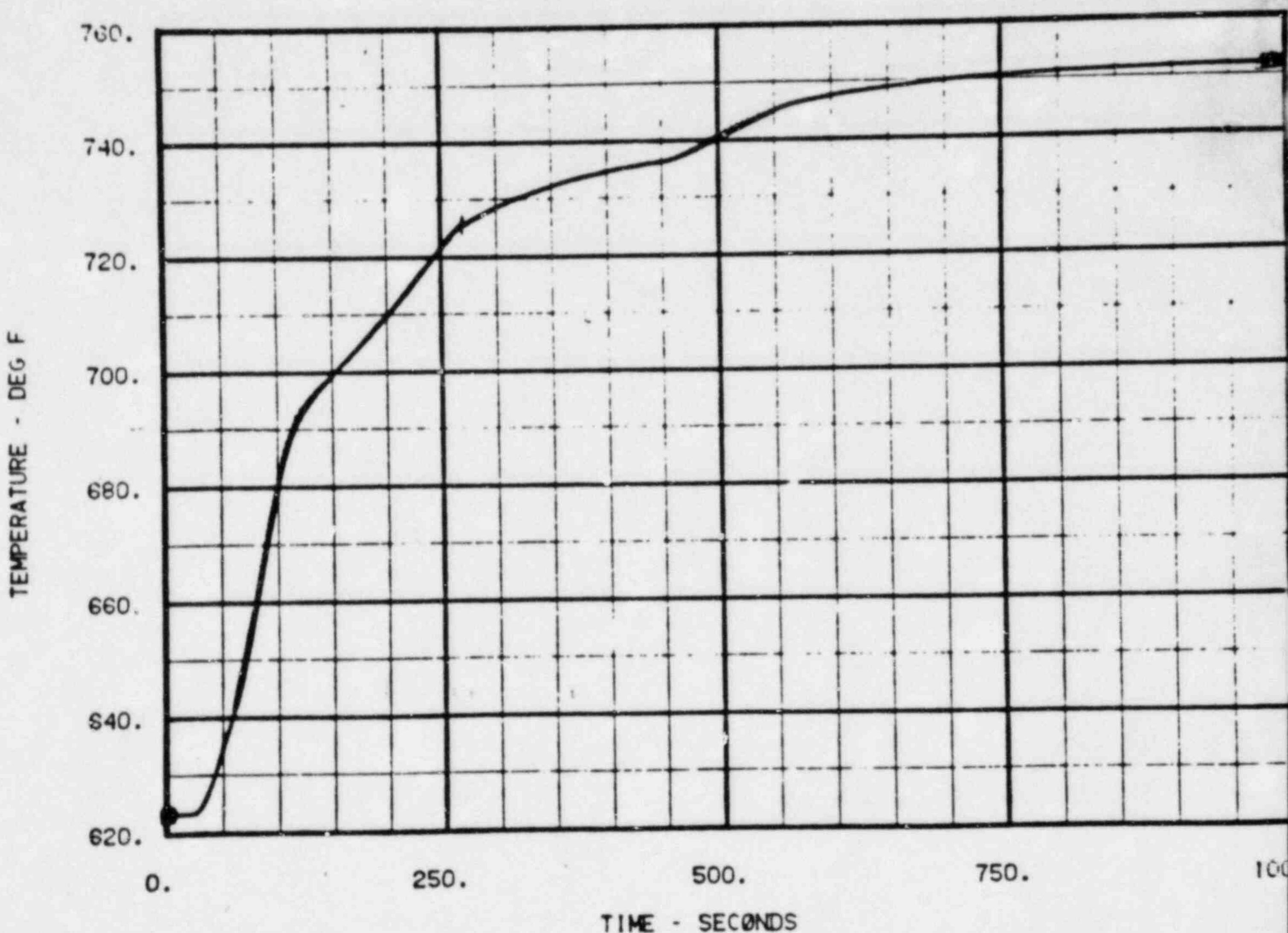
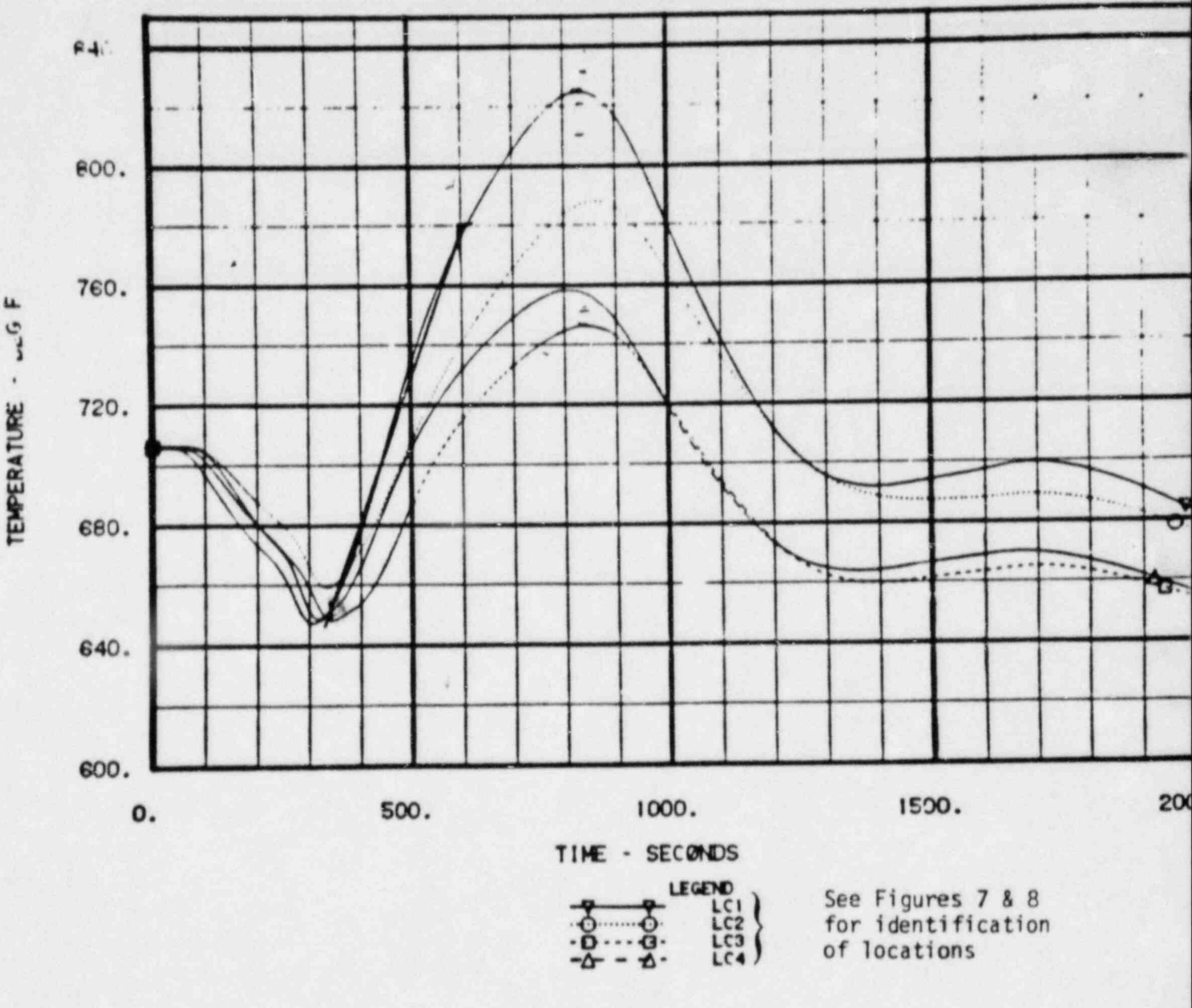


FIGURE 4

INLET PLENUM SODIUM TEMPERATURE RESPONSE
DURING A F-4A SATURATED STEAMLINE RUPTURE



See Figures 7 & 8 for identification of locations

Figure 5

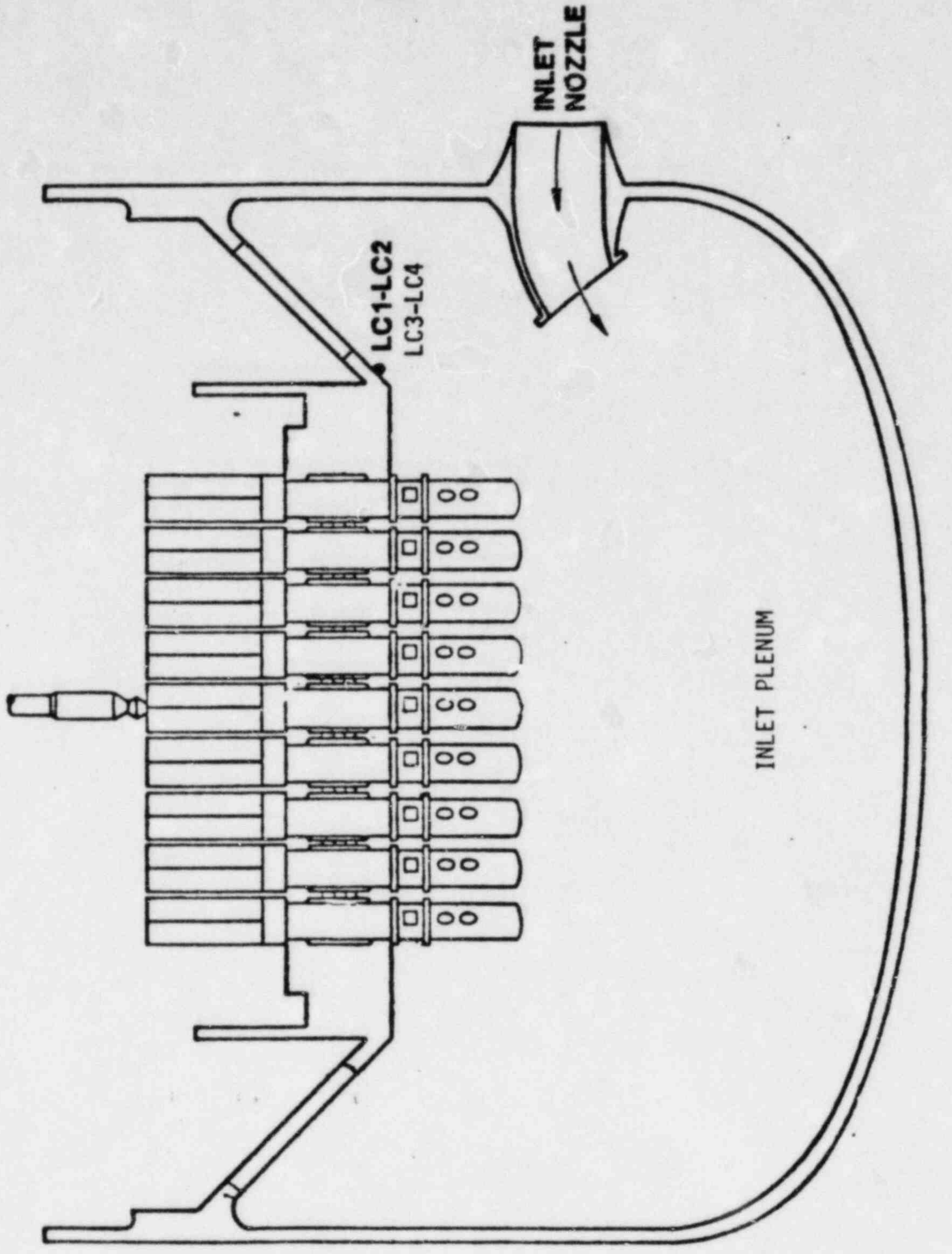


FIGURE 6 Measurement Locations in the CRBR Inlet Plenum - Elevation View