CONF-8707106--4-Draft

DE88 004421

CONF-8707106--4-Draft

INTOR DRAFT REPORT FOR CONFIGURATION AND MAINTENANCE

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*Research sponsored by the Office of Fusion Energy, U.S. Department of Energy, under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Incorporated.

2. VERTICAL AND HORIZONTAL ACCESS CONFIGURATIONS

A number of configuration features and maintenance operations are influenced by the choice of whether a design is based on vertical or horizontal access for replacing reactor components. For the purpose of this discussion, "vertical" means vertically or obliquely removing first wall/blanket components and includes some limited horizontal access for peripheral reactor equipment. "Horizontal" means only horizontally replacing first wall/blanket components without any vertical access, as in the INTOR reference design whereby complete torus sectors are removed. The features which are impacted most include the first wall/blanket segmentation, the poloidal field (PF) coil locations, the toroidal field (TF) coil number and size, access port size for in-vessel components, and facilities.

2.1 Configuration Features

2.1.1 Torus Segmentation and Passive Shell Structures

The horizontal access configuration allows a design with the least number of torus segments. The simplest torus segmentation for this approach is illustrated in Fig. VII-1 and is the basis for an earlier INTOR configuration [3]. The torus is divided into a number equal to the number of TF coils, and the size of the coils is chosen to permit straight radial extraction of torus sectors. In the (ref. 3) INTOR configuration, the window opening (or port) is approximately 4 X 7 m, and the TF coils are larger than the requirement for plasma edge ripple permitting the first wall/blanket to be designed as one segment. (The INTOR Phase Two A, Part 2 configuration [4] has coils which are sized to meet the ripple requirement and their smaller size requires two sectors for each TF coil.)

The vertical access configuration requires three to six segments depending on several factors. Three segments meet the geometric requirements if only outboard blanket coverage is needed for both vertical and oblique removal. Six segments are required for full coverage. Figure VII-2 shows an arrangement of three inboard and three outboard blanket segments in an oblique access configuration based on a 16 TF coil NET design [5]. Removal of the key plug allows disassembly of either the inboard or outboard center segment followed by the side segments. Figure VII-3 is a plan view of an oblique access arrangement with only outboard blanket segments [6]. Removal of the center segment allows removal of the side segments. Figure VII-4 shows an arrangement of blanket and shield segments for a vertical access design [7]. This design requires initial removal of the blanket/shield assembly numbered 5a in order to gain access to the remaining blanket segments, b through d.

The presence of the blankets close to the plasma can be used for passive stabilization of the plasma. The horizontal access approach having the fewest number of segments provides the best stabilization; the vertical approach with higher segmentation of structures facing the plasma may be detrimental. To overcome this, it may be necessary to install highly conducting structures behind the first wall. This solution is being considered for NET [5]. Another possible solution is to provide conducting joints between segments. This approach may require the use of

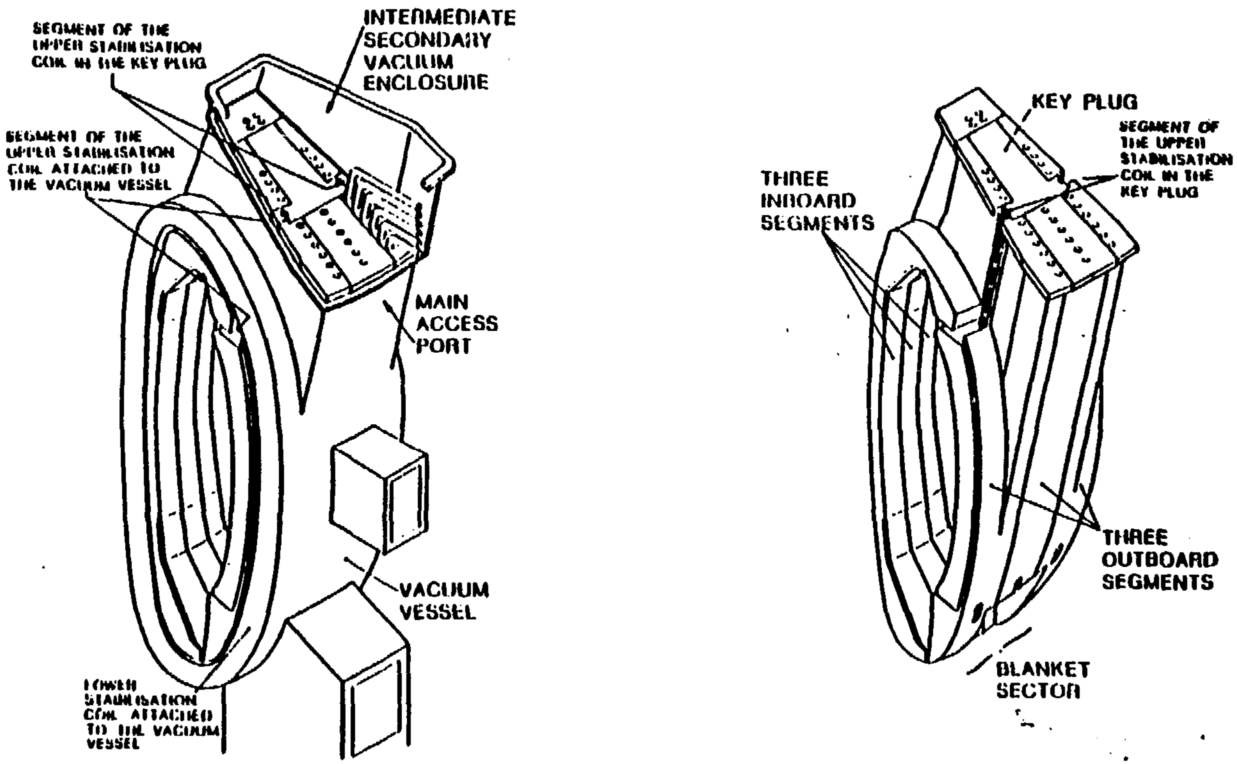


Fig. VII-2. Oblique access approach based on full blanket coverage.

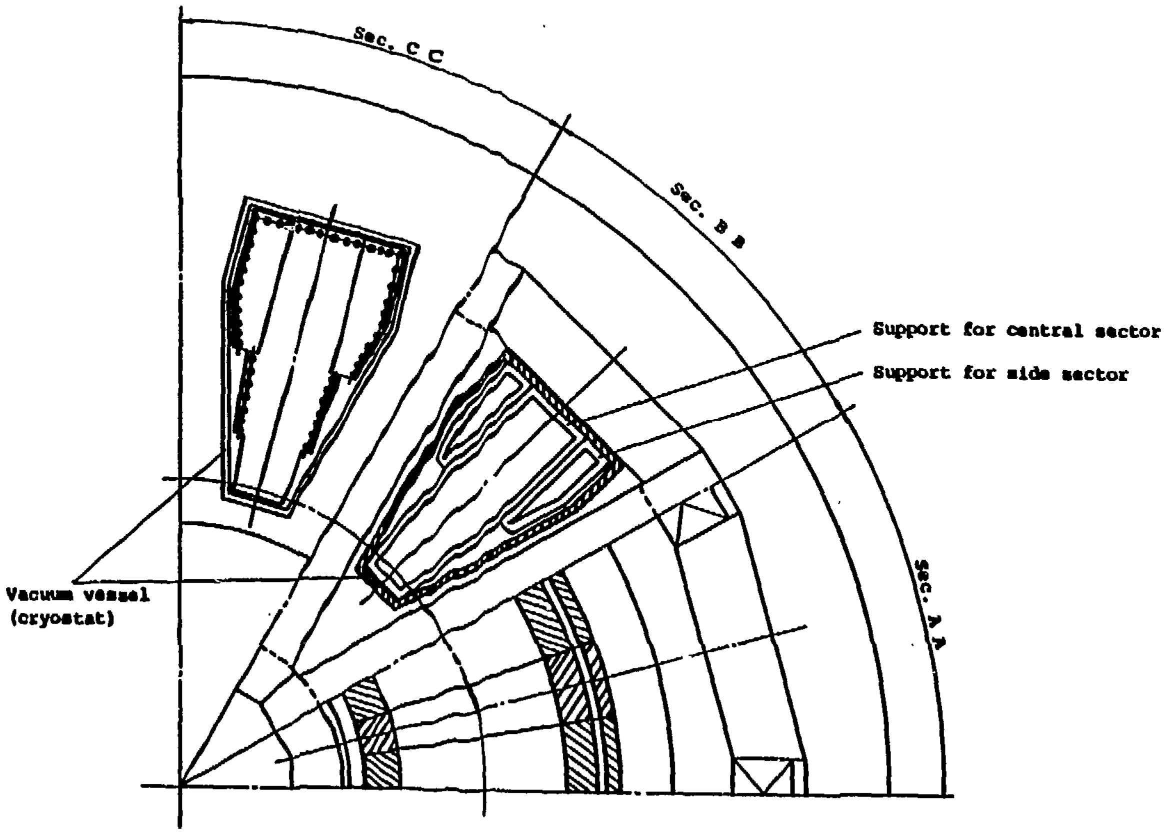


Fig. VII-3. Oblique access approach based on outboard blanket coverage.

an in-vessel manipulator for connecting and disconnecting these joints.

2.1.2 TF Coil Bore and TF Coil Structure

The TF coil bore for the Phase Two A, Part 1 INTOR configuration [3] was sized to permit horizontal removal of one sector for each TF coil. This resulted in a coil size which yielded a plasma edge ripple of 0.9%. The ref. 4 configuration reduced the coil horizontal bore by 0.3 m to 9.3 X 6.3 m to meet the ripple requirement of 1.2% while maintaining horizontal sector removal and, consequently, required two torus sectors for each TF coil. In order to reduce the coil bore further, it becomes necessary to increase the number of coils to meet the ripple limit, resulting in reduced access for horizontal sector removal and a reduced port opening for vertical segment removal. In the former case, additional segmentation of the sectors is necessary which is contrary to the simplicity of the horizontal access design. In the latter case, the vertical port opening may be oriented obliquely to increase the toroidal dimension as shown in Fig. VII-2.

The TF coils require substantial structure to react the out-of-plane forces caused by vertical magnetic fields. The ref. 3 and 4 INTOR configurations were limited to positioning these shear/compression structures at the top and bottom of the TF coils because of the large clear opening required for sector removal. In addition, internal bending structure is required for the unsupported outboard leg caused by the large window opening. The vertical access configuration permits a distribution of the

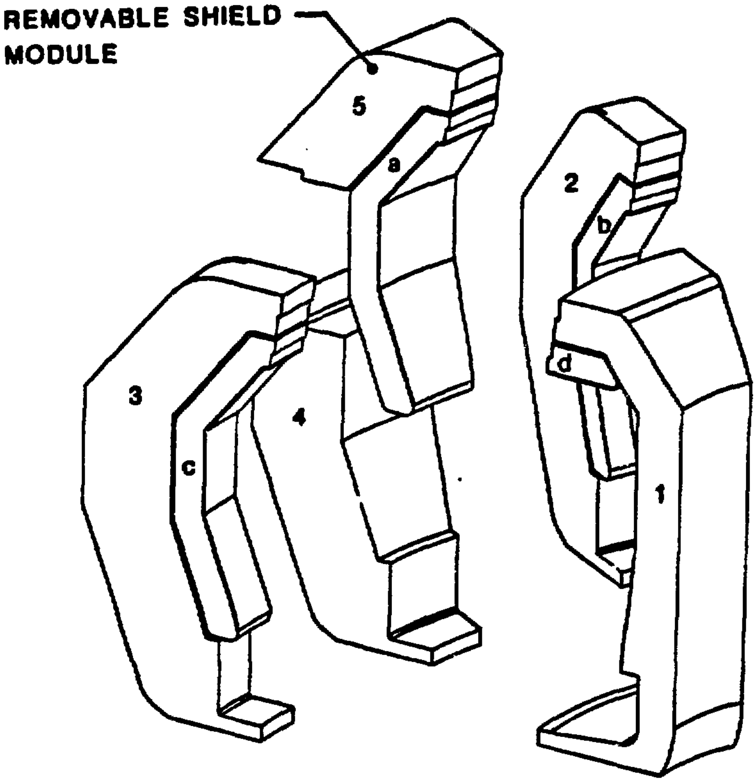


Fig. VII-4. Vertical access approach based on outboard blanket coverage.

shear/compression structure along the outer leg except in the window opening required for peripheral reactor components. Consequently, there is reduced bending in the unsupported leg with a reduction in internal structure.

2.1.3 Access Port Size

The horizontal access port size (window) for the ref. 4 configuration is 3.7 m X 8 m and is measured by the boundary of adjacent TF coils and the upper and lower outboard PF coils. This opening is sized to permit removal of one torus sector, as shown in Fig. VII-1.

The vertical access port in Fig. VII-2 is bounded by the TF coils and the upper inboard and outboard PF coils. This reduced trapezoidal opening requires that in-vessel components which are accessed must be segmented into pieces which can be vertically (or obliquely) lifted through the port. Those which lie under the TF coils must be translated first.

The vertical port, in addition to being small, accommodates pairs of coolant lines for each first wall and blanket segment and the semi-permanent shield modules. Fig. VII-5 is a plan view of the port which provides access to the segments in Fig. VII-4. It contains 13 sets of lines ranging in size from 4 to 20 cm diameter with each line having two demountable couplings. Table VII-1 is a detailed listing of the line sizes including the main headers. The overall size of the port is 4 X 4 X 3 X 1 m and has approximately one-third of the area of the full sector opening.

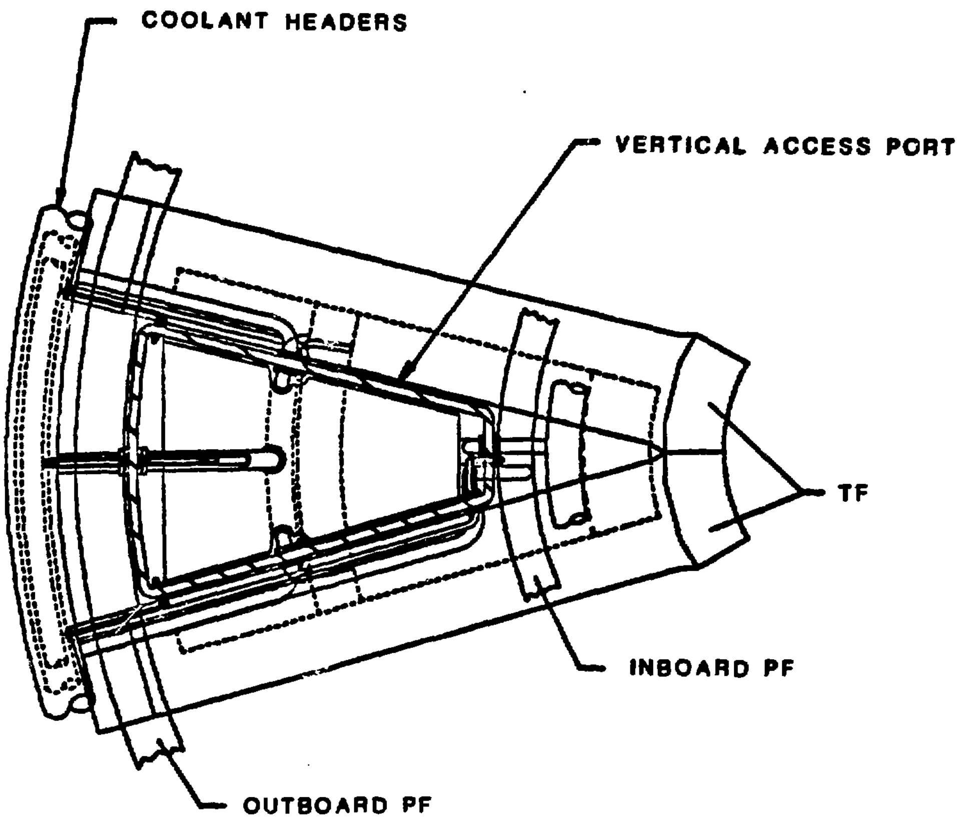


Fig. VII-5. Plan view of a vertical access port bounded by TF coils and upper inboard and outboard PF coils.

Table VII-1. Coolant requirements for a vertical access, single-null configuration

Component, segment no.	Main header (diameter, cm)	Feedline to segment (diameter, cm)
First wall, a	37	9
b		7
c		7
d		6
Inboard shield, 1	39	15*
Outboard shield, 5	24	5*
2		4*
3		4*
4		8*
Blanket, a	58	20
b		15
c		15
d		15

*The shield segment lines may be routed through the lower TF intercoil structure.

2.1.4 Active Control Coil Arrangements

The horizontal access configuration allows active control coils to be arranged in virtually any poloidal position around the plasma except in the area of the vacuum pumping duct and the divertor. Fig. VII-6 [6] shows an upper and lower coil located within the bore of the TF coils, above and below the removable torus sector. In this arrangement, the coil leads penetrate the cryostat through a vacuum-tight conduit above and below the window opening. This design is not amenable to replacing a coil if a failure occurs because these poloidal coils are trapped in the bore of the TF coils. If it is more advantageous to position control coils closer to the midplane, the saddle coil arrangement

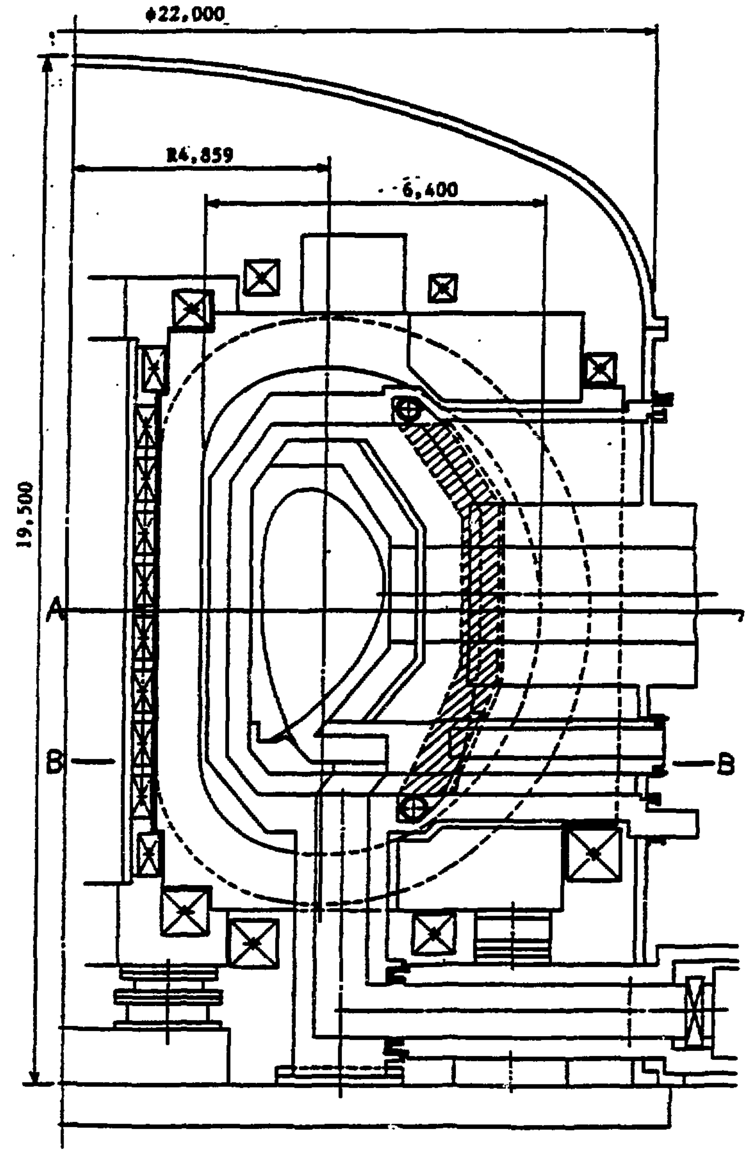


Fig. VII-6. Active control coils located outside the torus sector in the horizontal access configuration.

shown in Fig. VII-7 allows the coil to be located in each torus sector [4] where its replacement is made possible by removing the sectors.

The vertical access configuration requires a different solution for locating control coils. The large access ports located at the top of the reactor make up an area where continuous coils cannot be located, because this region is dedicated for removal of blanket segments. Hence, a coil can only be located at the upper or lower edge of the port. Fig. VII-8 [5] shows an upper coil arrangement at the lower edge of the port opening. This may present a problem for designing a coil set with some degree of symmetry since the port area is an exclusion zone. The use of saddle coils does not appear reasonable here because each segment would require a saddle with a pair of leads; therefore, each port would have six leads connected to relatively small coils if only outboard blankets are assumed.

The lower coil in the same figure is positioned above the divertor and is segmented so that coil replacement is more easily achievable. In this arrangement, a pair of coil leads is located between each pair of TF coils.

2.1.5 <u>Bellows</u>

The thermally induced movement of the port openings, relative to a stationary support point in the reactor, will be different for the horizontal and vertical access configurations. If the stationary point (C) is taken to be the centerline of the machine at the reactor hall ground level, and the movable points

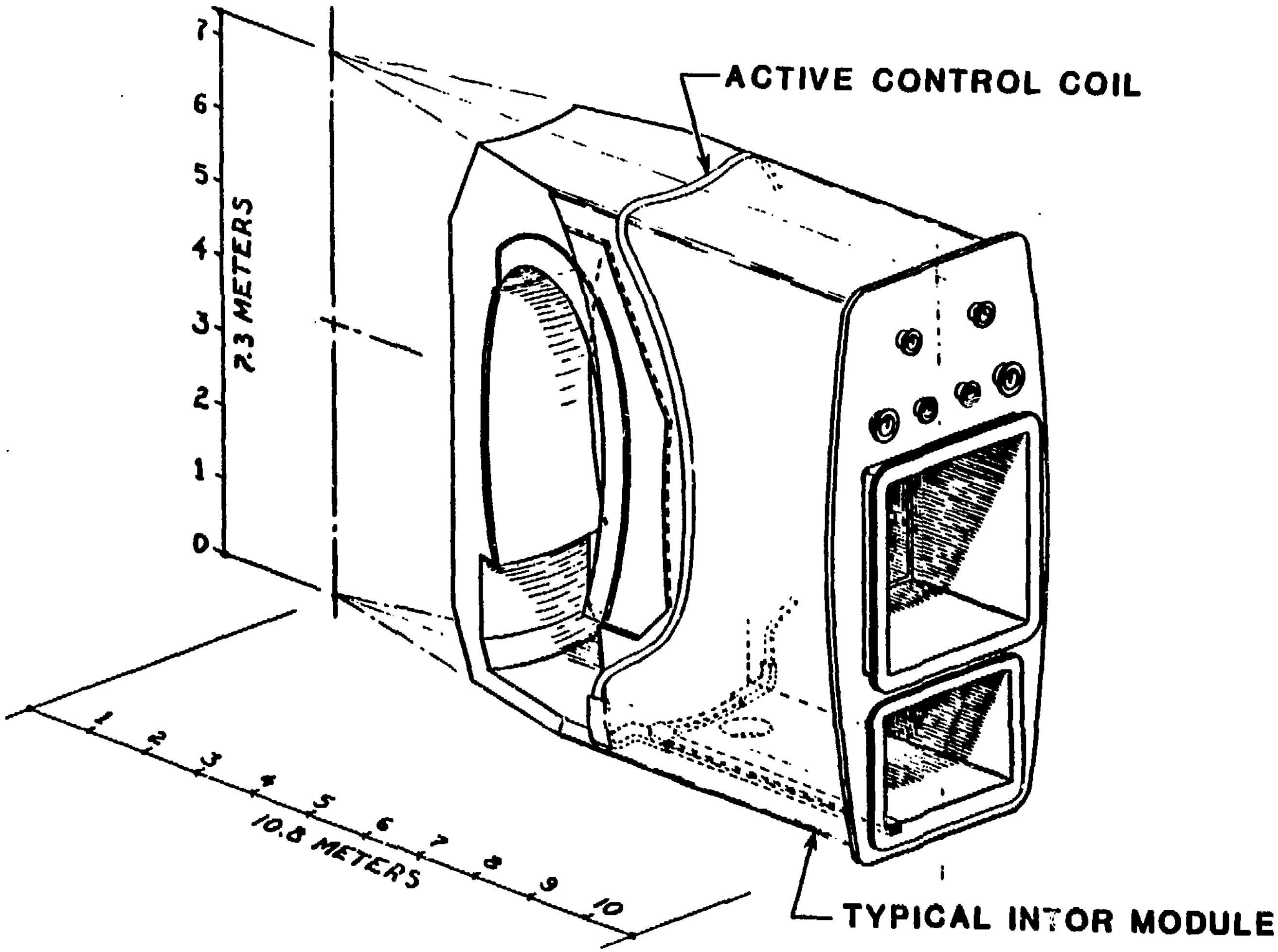


Fig. VII-7. An active, saddle control coil located in the removable torus sector.

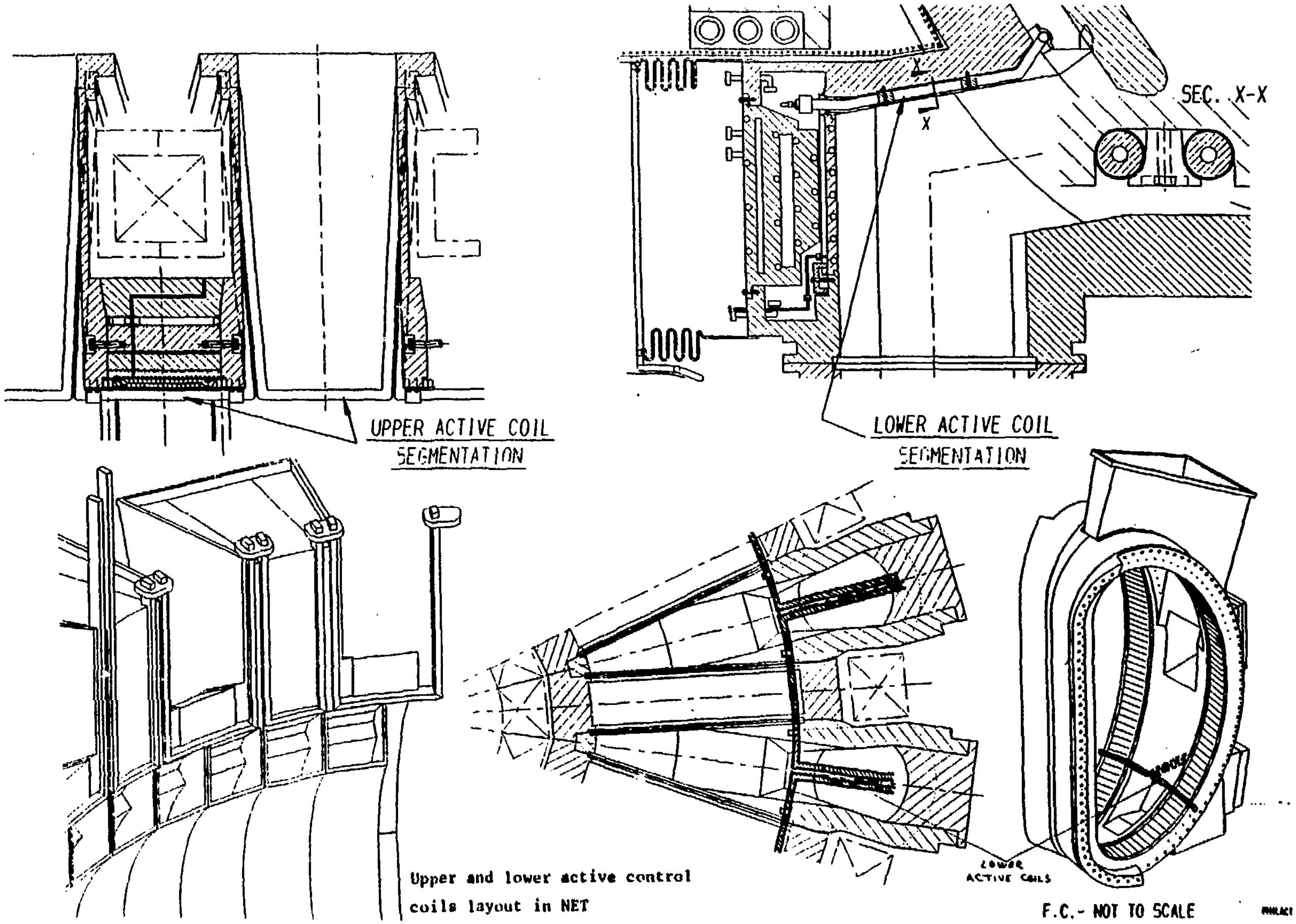


Fig. VII-8. Upper and lower control coils for a vertical access configuration.

(A and B) the upper and lower port edges, a measurable difference results. Figures VII-9 and VII-10 show the geometrical arrangements for the horizontal and vertical access configurations, respectively. An analysis assuming a 150 degree C bakeout temperature was done to quantify the relative movement between the two approaches [6]. Table VII-2 summarizes the dimensional changes of points A and B relative to point C.

Table VII-2. Comparison of horizontal and vertical port movements caused by bakeout at 150 degrees C.

	Horizontal movement (mm)	Vertical movement (mm)
Horizontal Port		
A	22.9	17.0
B	22.9	0
Vertical Port		
A′	9.6	34.9
B′	18.3	34.9

Note that for the horizontal port configuration, the horizontal movement is in the direction of the bellows axis, and the vertical movement is perpendicular to that axis. The vertical port configuration has a bellows axis which is 76 degrees above the horizontal surface. The resultant motion of A′ is 75 degrees, which is nearly parallel to the axis, and the motion of B′is 62 degrees, a 14 degree difference.

It is desirable to have relative motions between structures taken up along the axis of a bellows in order to avoid excessive

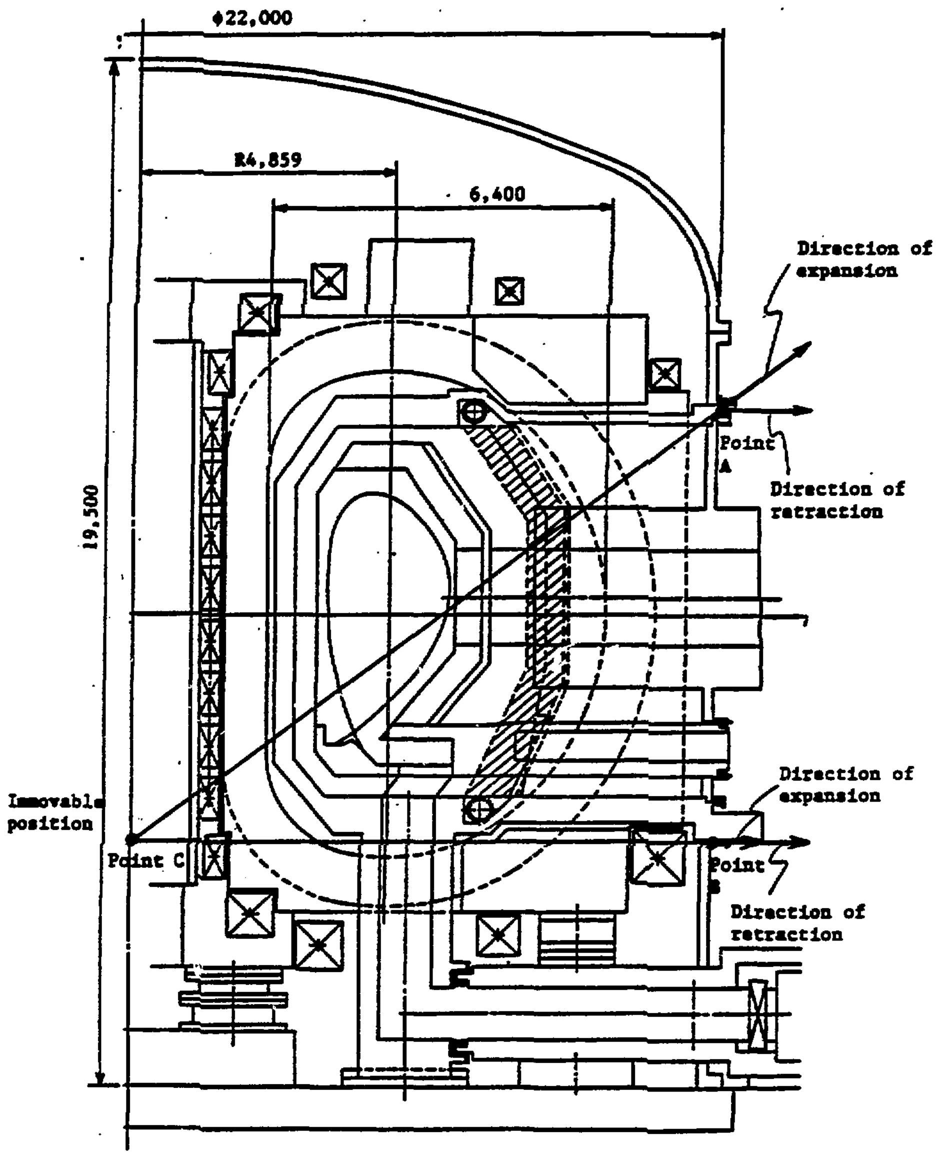


Fig. VII-9. Thermal deflections for the horizontal port design.

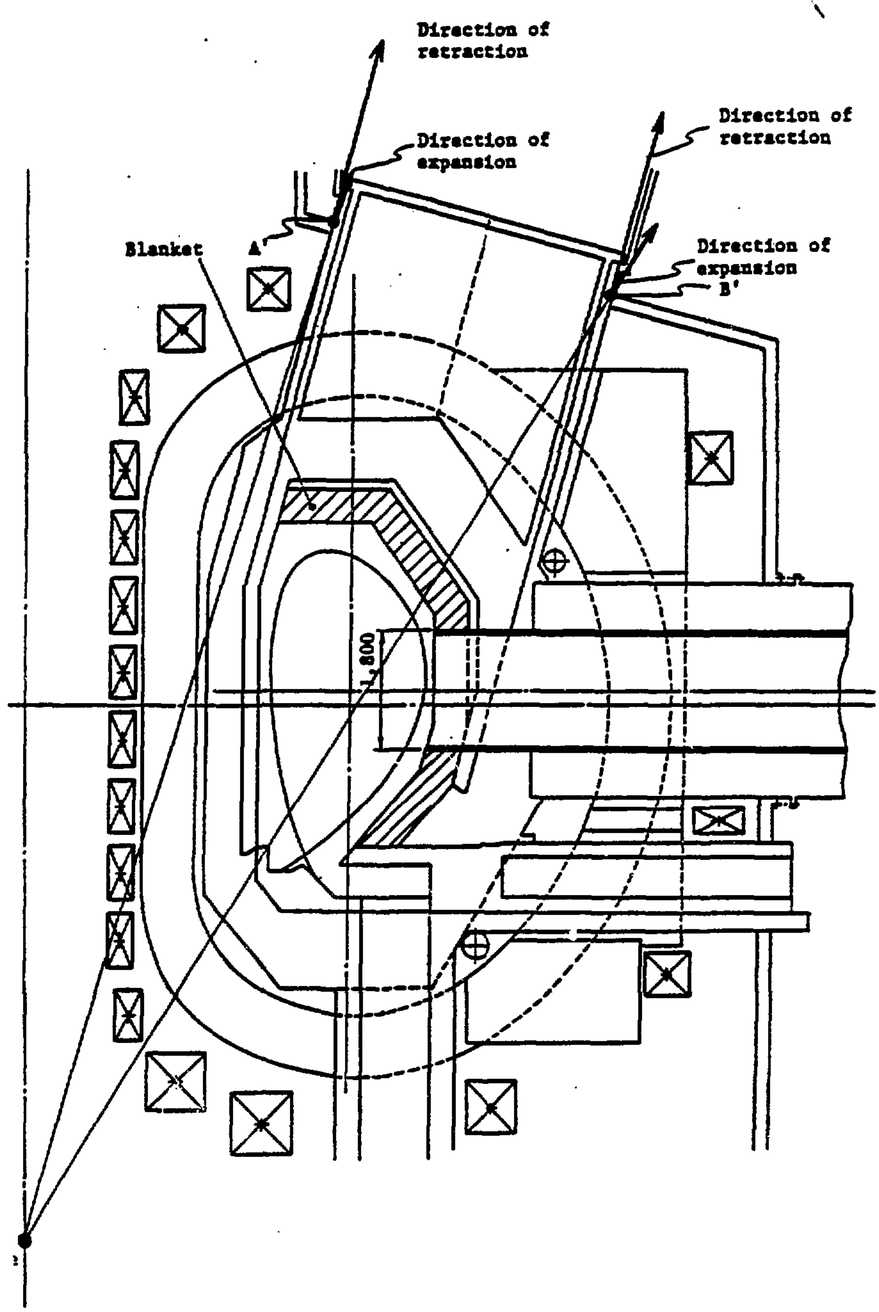


Fig. VII-10. Thermal deflections for the vertical port design.

stress concentrations (in the bellows). On this basis, the vertical port arrangement is closest to being parallel to the direction of motion. By adjusting the orientation of this port, it is possible to compromise the relative angles of points A and B and reduce the 14 degree difference at point B. The reorientation of the port is limited, however, by the removal of first wall/blanket components.

Lateral movement, or differential thermal expansion of structures may be accommodated by installing bellows in pairs. This approach may be required around the horizontal port, particularly if bakeout temperatures higher than 150 degrees are required. The double bellows has the effect of moving the port flange interface radially outward, which is not a configuration constraint.

Although the 17-mm vertical displacement does not appear excessive considering the size of the structures involved, another solution is possible. This requires preforming the upper portion of the bellows to match the predicted angle of movement. In this case, point A moves up 37 degrees. Therefore, if that part of the bellows was manufactured to that angle, local movement would be parallel to the bellows. Clearly, for structures the size of INTOR, it is reasonable to assume that some amount of development would be required for manufacturing unsymmetrical bellows.

2.1.6 Facilities

The discussion on facilities for the vertical and horizontal configurations is based primarily on work done for the Next

European Torus (NET) and the Fusion Engineering Reactor (FER). Some earlier facilities work was also reported in references 4 and 10. For a conventional reactor hall which is sized to accommodate the peripheral reactor equipment, it was found that the enclosed volume is essentially the same, 2.5E5 m3 [6] assuming a circular building for the vertical design and a rectangular building for the horizontal design. However, large differences exist for the span of the bridge cranes.

The circular building has a diameter of 72 m based on providing PF coil laydown area in an eccentric arrangement where the center of the reactor is approximately 5 m from the center of the building. The rectangular building has a span of 58 m and a length of 83 m and is able to accommodate the PF coil laydown area in one or more of the building corners. Either crane system is suitable for the vertical or horizontal access reactors, and the cost difference between the bridge spans is not significant compared to the reactor cost.

There is potentially a cost difference between the two building approaches considering the roof structures. Clearly it is more costly to construct a freestanding 72-m circular roof compared to a freestanding 58-m rectangular roof. Since this level of detail was outside the scope of this study, the cost difference cannot be quantified. However, the depth of the polar crane bridge will be larger than the rectilinear bridge by a factor of the building span squared, thereby also increasing the circular reactor hall height by that amount.

Figure VII-11 is a perspective cutaway of the FER

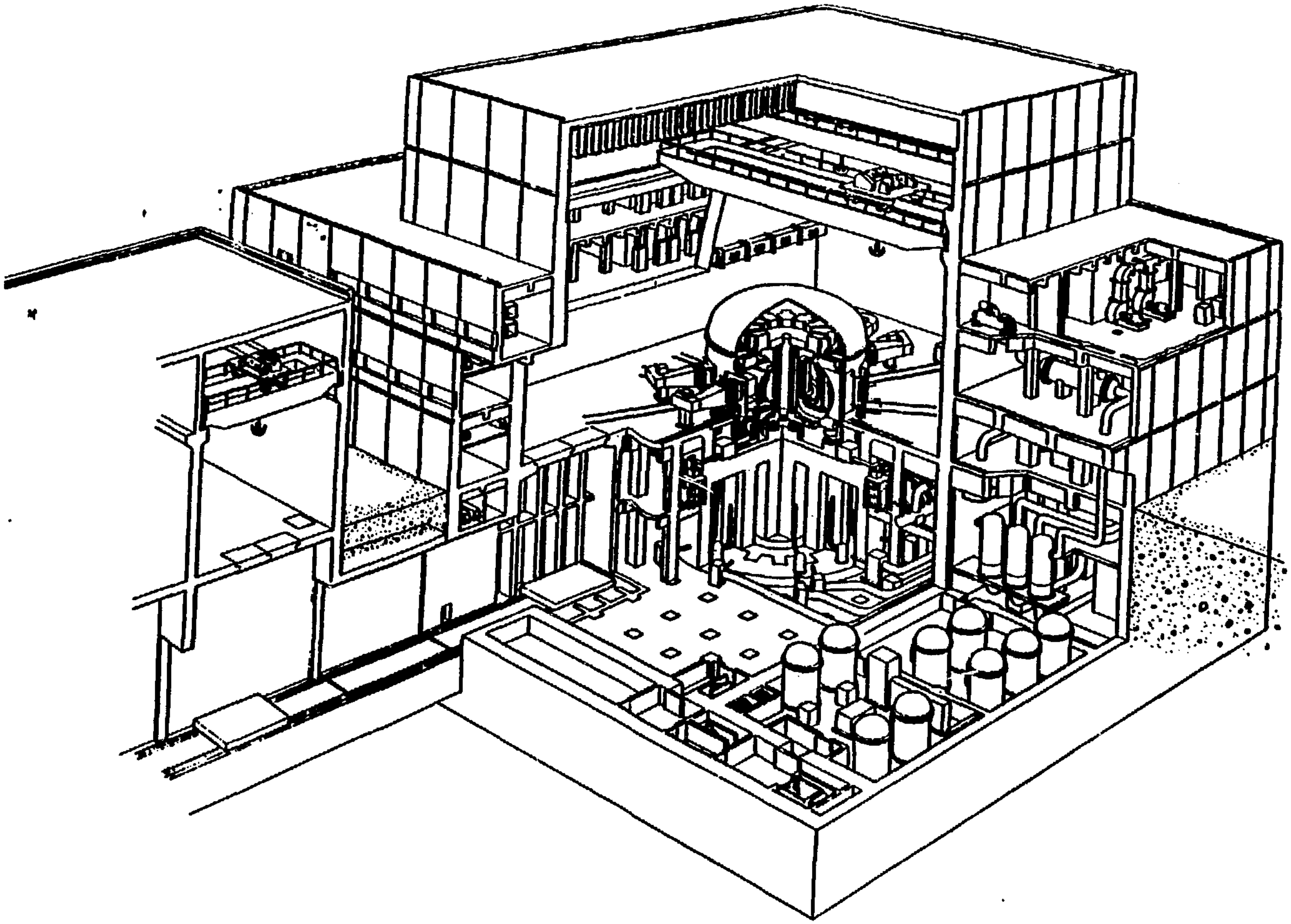


Fig. VII-11. Perspective cutaway of the FER conventional facility which is sized for reactor peripheral equipment.

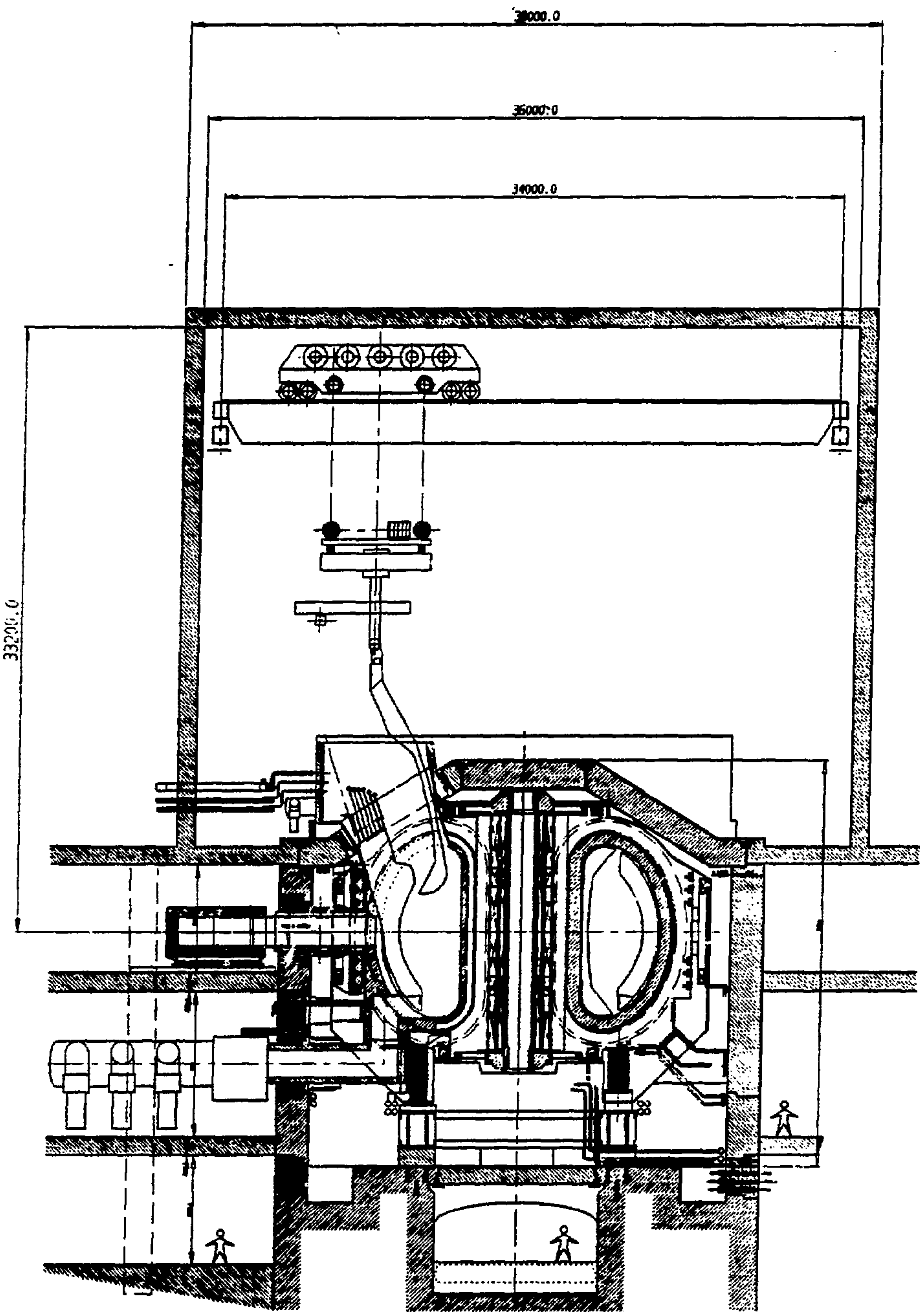
conventional facility showing a rectangular reactor hall. This design is representative of the INTOR facility design.

A nonconventional facility design is shown in Fig. VII-12 and is based on one of the NET arrangements. In this configuration, it is possible to have a span of 34 m because the building is sized for the reactor and not the peripheral equipment. This arrangement is more suitable for a vertical access reactor configuration and has the smallest reactor building span. A circular building with PF coil laydown area would have a larger diameter.

2.2 Impact From The PF Coil System And Plasma Engineering

The INTOR Phase Two A, Part 2, and Part 3 configurations were based on an arrangement of PF coils which provide a large window for horizontal sector removal while meeting the requirements for plasma shaping (elongation = 1.6 and triangularity = 0.25). No consideration was given to optimizing the location of the coils for the purpose of reducing their cost. During this time, the concept for a less costly PF coil arrangement was introduced as a result of studies undertaken by the NET Team. Their work, although based on different reactor parameters, created a spirited interest to reevaluate the INTOR PF system.

The impact to the INTOR design is to relax the horizontal window constraint by allowing PF coils to be located closer to the midplane and to change the access and disassembly approach for replacing first wall/blanket components. An inspection of Figs. VII-9 and -10 shows the major differences between the two



NET maintenance equipment for substitution of the internal segments inside an upper Tight Intermediate Containment (T.I.C.)

Fig. VII-12. Elevation of a NET reactor hall arrangement sized for the reactor only.

approaches. In the first figure, the only constraint is to exclude PF coils from the outboard region so that complete torus sectors may be radially extracted. No consideration was given to vertical access; therefore, the upper outboard PF coils are spaced close together. In the second figure, the horizontal constraint is only to provide access to midplane equipment, but there is an exclusion area above the reactor for a vertical port.

Figure VII-13 was less of a departure from the INTOR baseline, but nevertheless incorporates a PF system which permits vertical removal of in-vessel components. In this arrangement, a 3-m horizontal access window was deemed necessary for peripheral equipment such as test modules and heating modules, and the locations of upper PF coils were constrained by a port for vertical (not oblique) disassembly. It was found that a modest reduction in PF cost (7%, measured by MA-meters) [7] was possible according to systems code comparisons. This result was not initially duplicated by all of the delegations because of different configuration assumptions, but it is generally agreed that for low elongation plasmas, e.g., less than K=2.0, PF cost reductions of at least a few percent are possible. For higher elongations, e.g., 2.2%, locating the PF coils closer to the midplane of the machine can significantly reduce the PF system cost. Reductions of approximately 20% and 40% were calculated for stored energy and MA-meters, respectively [8].

The optimization of the PF coil locations coupled with key plasma engineering parameters is very much a function of configuration constraints. Therefore, it is not possible to

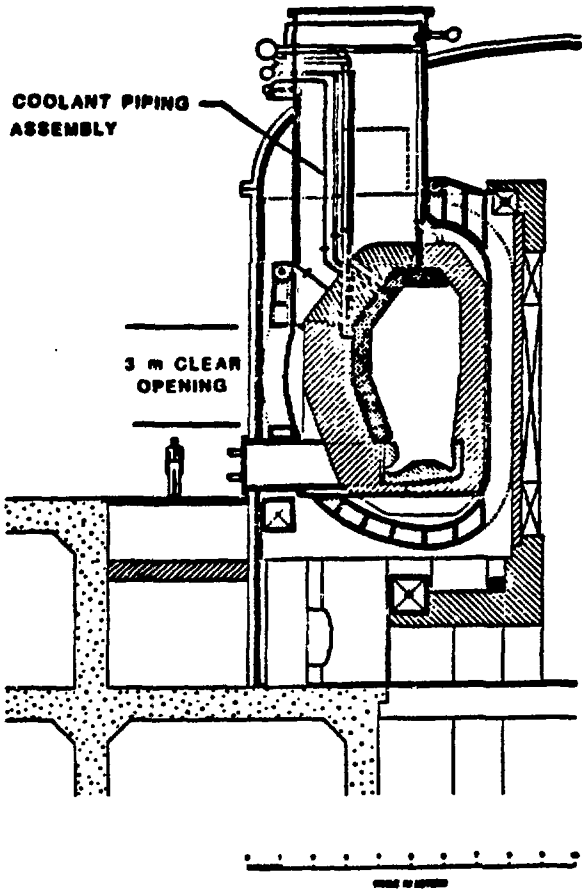


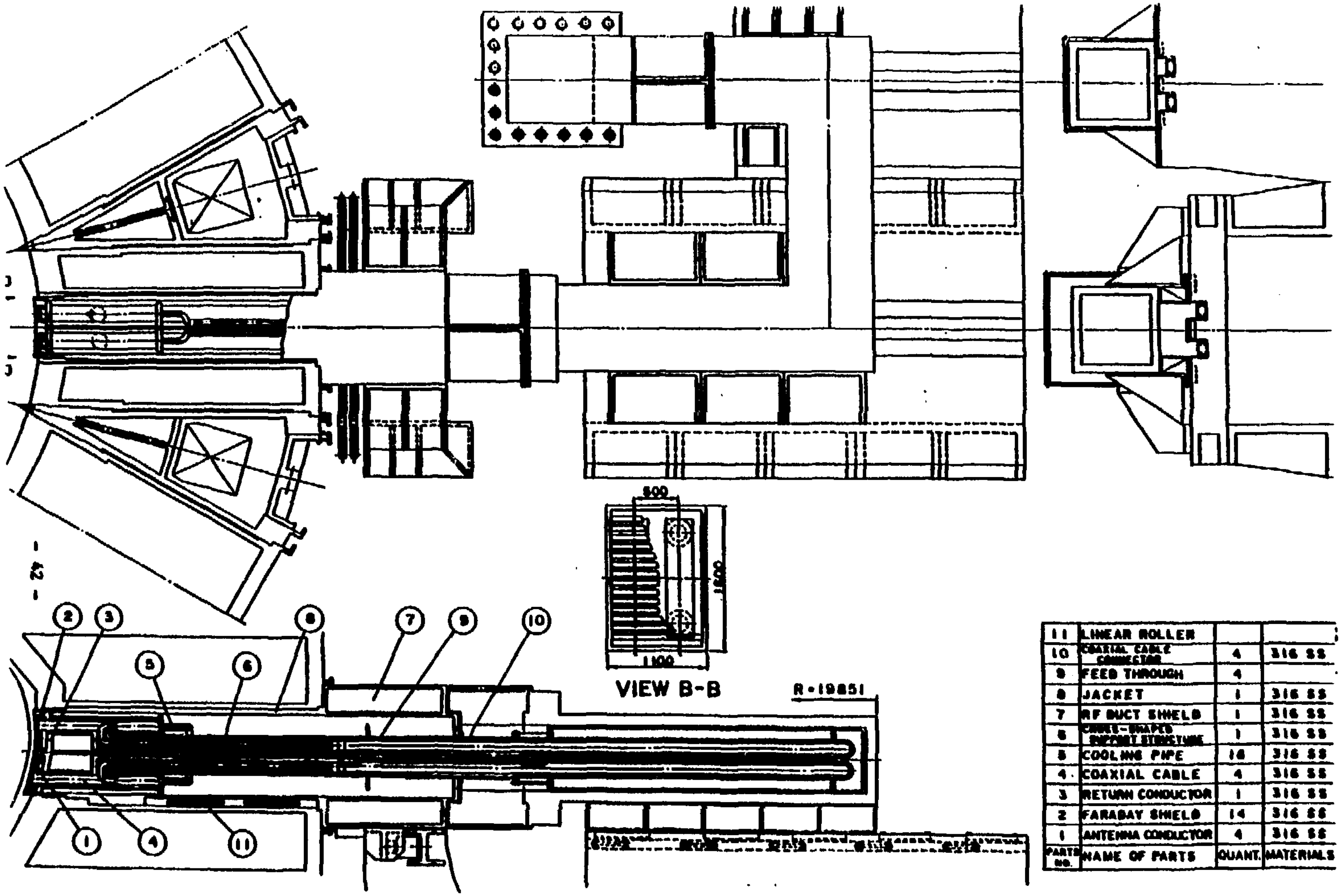
Fig. VII-13. A modified INTOR configuration for vertical access to the in-vessel components.

optimize except around a given set of constraints, which may be defined by exclusion zones for installing peripheral reactor equipment, external support structure, access for replacing components, and even the plasma major radius. It is, in fact, constraining the plasma radius which drives the configuration to oblique access. It is possible, however, to quantify for a given triangularity that increasing elongation reduces the amount of PF coil conductor, stored energy, etc..., and moving coils closer to the midplane has the same effect. This applies to both single- and double-null divertor configurations. A detailed discussion of the PF system from the point of view of coil locations and plasma engineering parameters is presented in Chapter VI.

2.3 Major Reactor Interfaces

2.3.1 RF System

The space required for installing the rf system is one of the constraints for locating PF coils around the midplane. In Fig. VII-14, it can be seen that a vertical clearance at the midplane of approximately 2 m is required for the ICRH module [6]. Allowing for the rf duct shield, an exclusion zone of approximately 3 m is the constraint for locating PF coils. In the horizontal direction, the only constraint is to provide several meters between TF coils, because the module is installed perpendicular to the plasma. Fig. VII-15 is a perspective view of an ICRH module capable of delivering 2 MW of energy [7]. Because of its compact design, modules can be arranged to



PARTS NO.	NAME OF PARTS	QUANT.	MATERIALS
11	LINEAR ROLLER		
10	COAXIAL CABLE CONNECTOR	4	316 SS
9	FEED THROUGH	4	
8	JACKET	1	316 SS
7	RF DUCT SHIELD	1	316 SS
6	CROSS-SHAPED SUPPORT STRUCTURE	1	316 SS
5	COOLING PIPE	16	316 SS
4	COAXIAL CABLE	4	316 SS
3	RETURN CONDUCTOR	1	316 SS
2	FARADAY SHIELD	14	316 SS
1	ANTENNA CONDUCTOR	4	316 SS

Fig. VII-14. The ICRH module requires a 3-meter vertical exclusion zone at the midplane.

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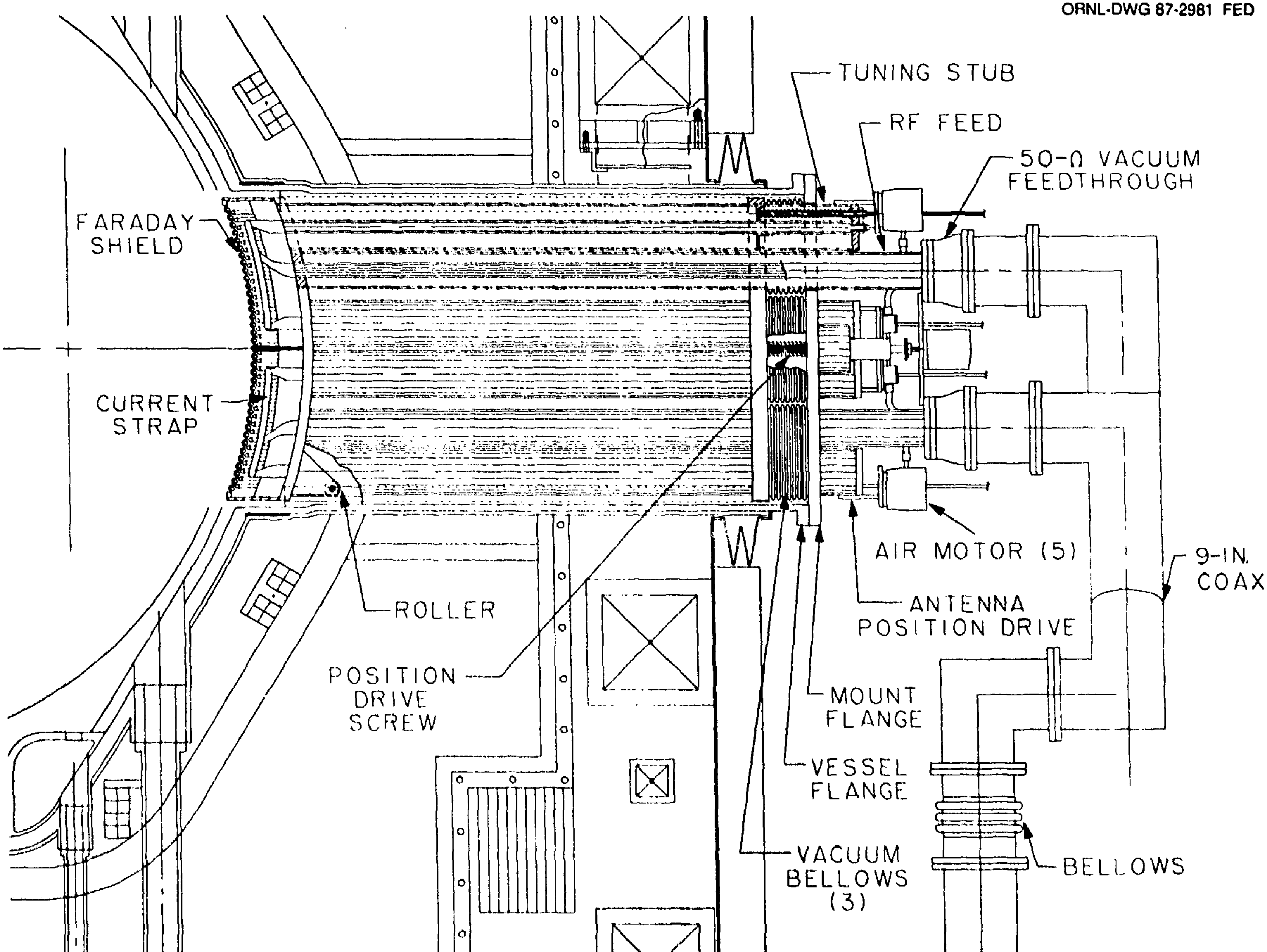


minimize the access constraints between PF and TF coils.

The space requirements for the LHRH module are approximately the same as those above. Figure VII-16 shows that 2 m is needed for the vertical clearance of the module [5], plus the addition of radiation shielding. Since this is also a perpendicular installation there is no constraint to the TF coils.

Configurations with obliquely installed heating modules were also investigated and found to have difficult interfaces. In a vertical access configuration, they compete for space with the coolant lines from the first wall/blanket segments.

2.3.2 <u>NBI System</u>

The neutral beam injection system installed perpendicular to the plasma has space constraints which are less than the rf injection systems above. Tangential injection, on the other hand, does impact the space between TF coils but not the vertical clearance between PF coils. Figures VII-17 and -18 show two arrangements with tangential injection. The first shows injection to a location less than the plasma major radius, the second at the major radius. It is clear, from a configuration point of view, that injection systems should be located horizontally at the midplane; therefore, vertical access or other constraints are not considered for this system.

2.4 Maintenance Procedures and Equipment

The maintenance procedures and equipment that are of interest deal with the replacement of first wall/blanket components and divertors. It was shown previously that the vertical access configuration has a minimum of three segments for

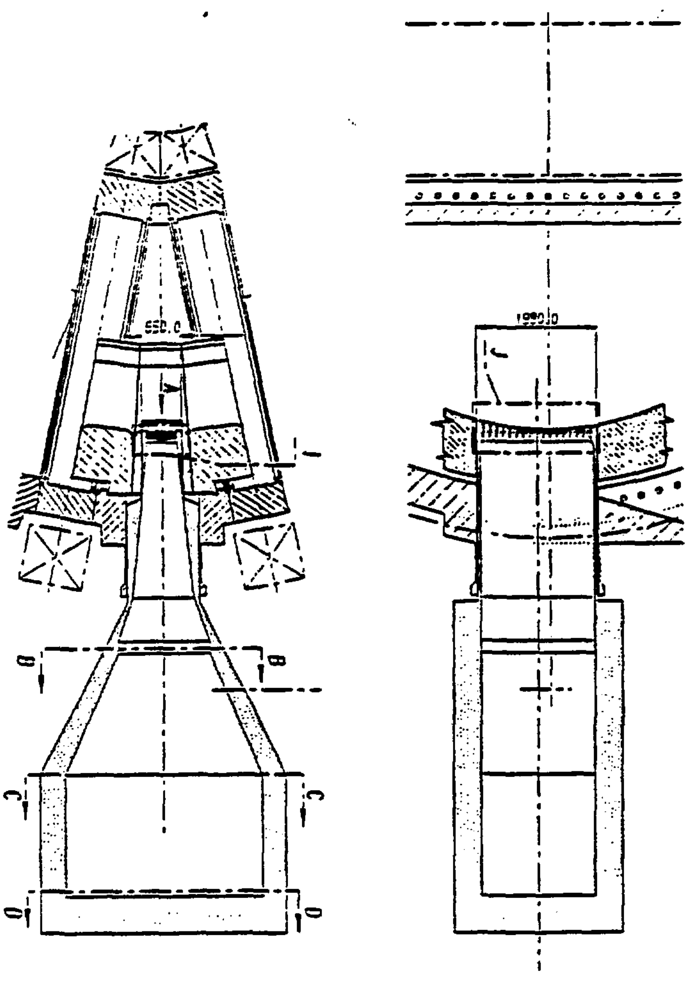


Fig. VII-16. LHRH module has space requirements similar to ICRH at the reactor interface.

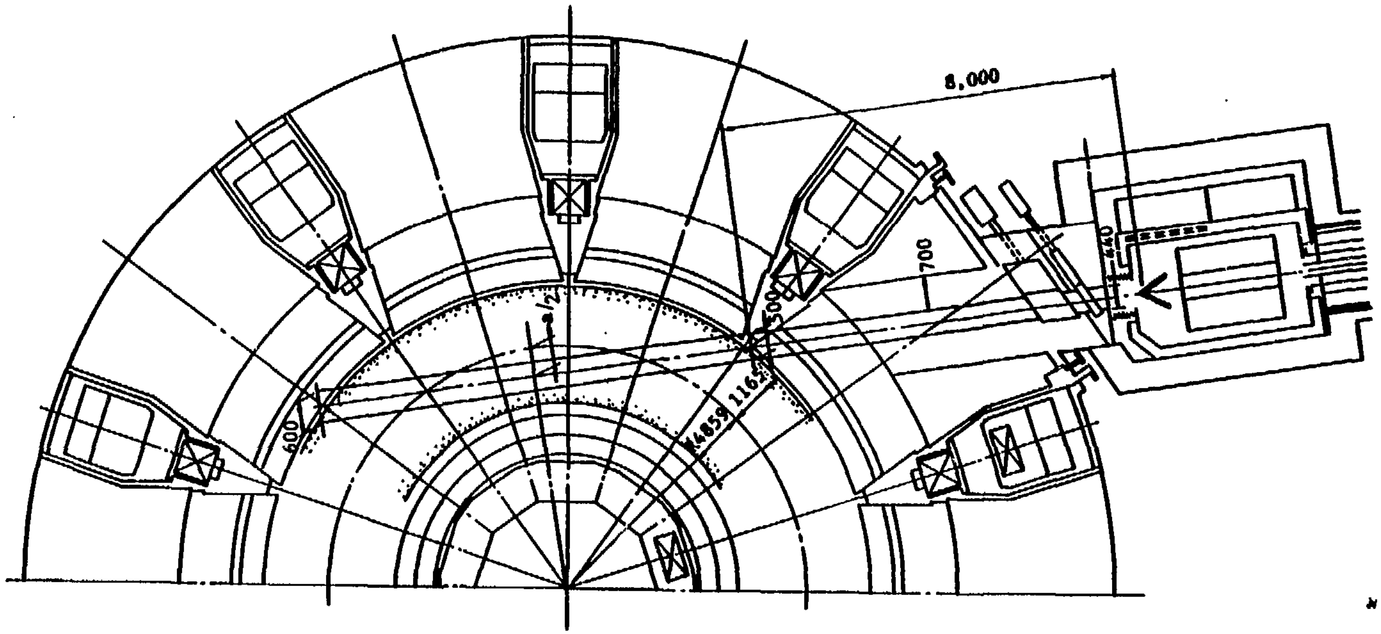


Fig. VII-17. Tangential injection with a negative-ion source; target location at R - a/2.

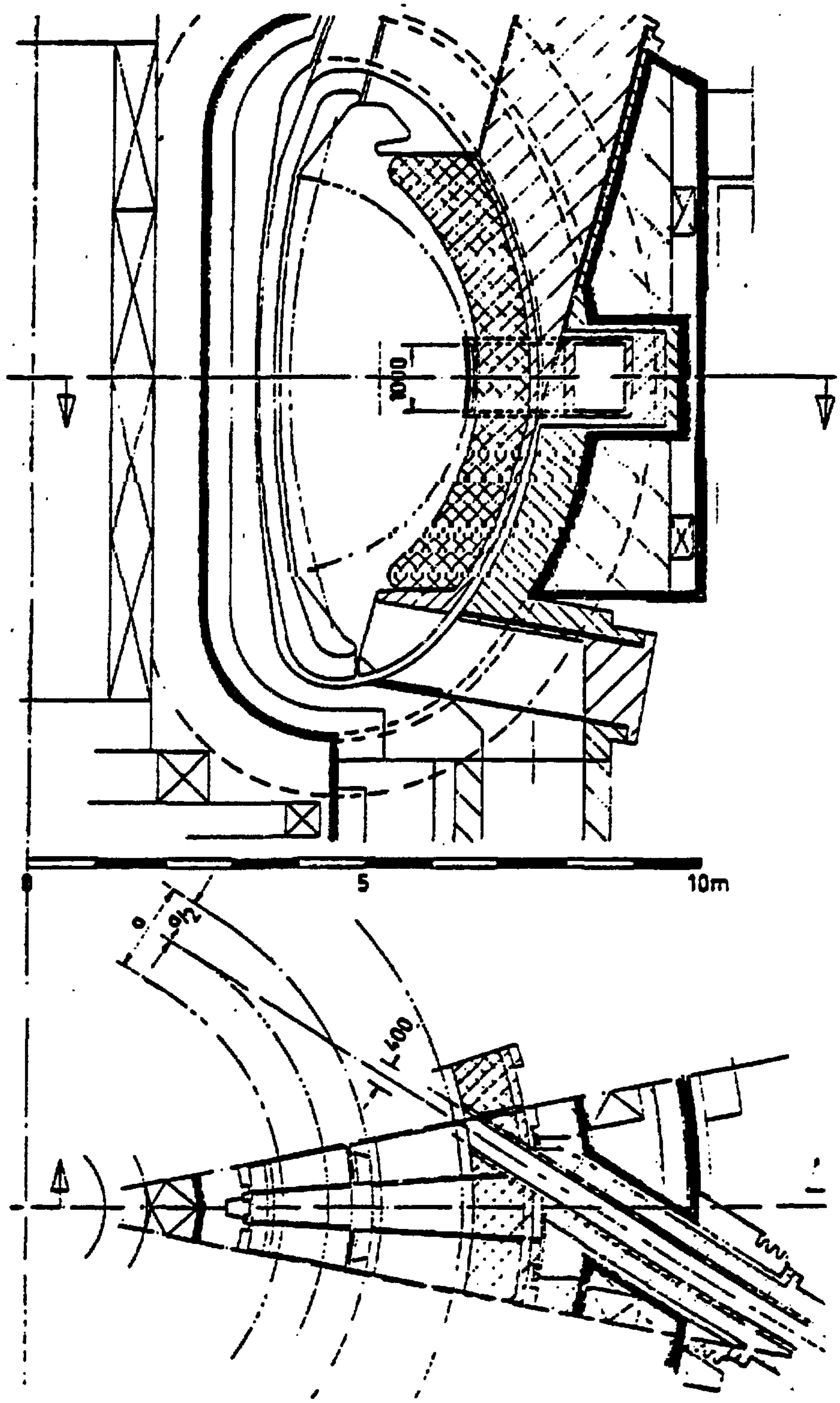


Fig. VII-18. Tangential injection at the plasma major radius.

each TF coil, in contrast to the horizontal access configuration which has one removable sector. Hence, it is not surprising that the downtime for replacing the in-vessel components is consistently higher for the former case, since there at least three times as many coolant lines, mechanical interfaces, and handling procedures. Table VII-3 is a summary of the downtimes which were independently estimated by the delegations [5,6,7]. It should be noted that the downtime consists mostly of time for pre- and post-bakeout of the plasma chamber and ranges from 50-80%.

Table VII-3. Comparison of downtime for replacing first wall/blanket components for vertical and horizontal configurations.

		Horizontal configuration	Vertical configuration
FW/B Replacement	(EC)	360	440
	(J)	515	570
	(USA)	335	374

A number of different concepts for replacing in-vessel components are possible and are illustrated in the following figures. Figure VII-19 is an approach whereby the entire torus sector is removed by means of a transporter and is applicable for the INTOR configuration shown in Fig. VII-1. Figure VII-20 is a similar approach except in this case the divertor module is removed first, as shown in Fig. VII-21, in order to engage the sector transporter.

The ability to independently remove the divertor modules is

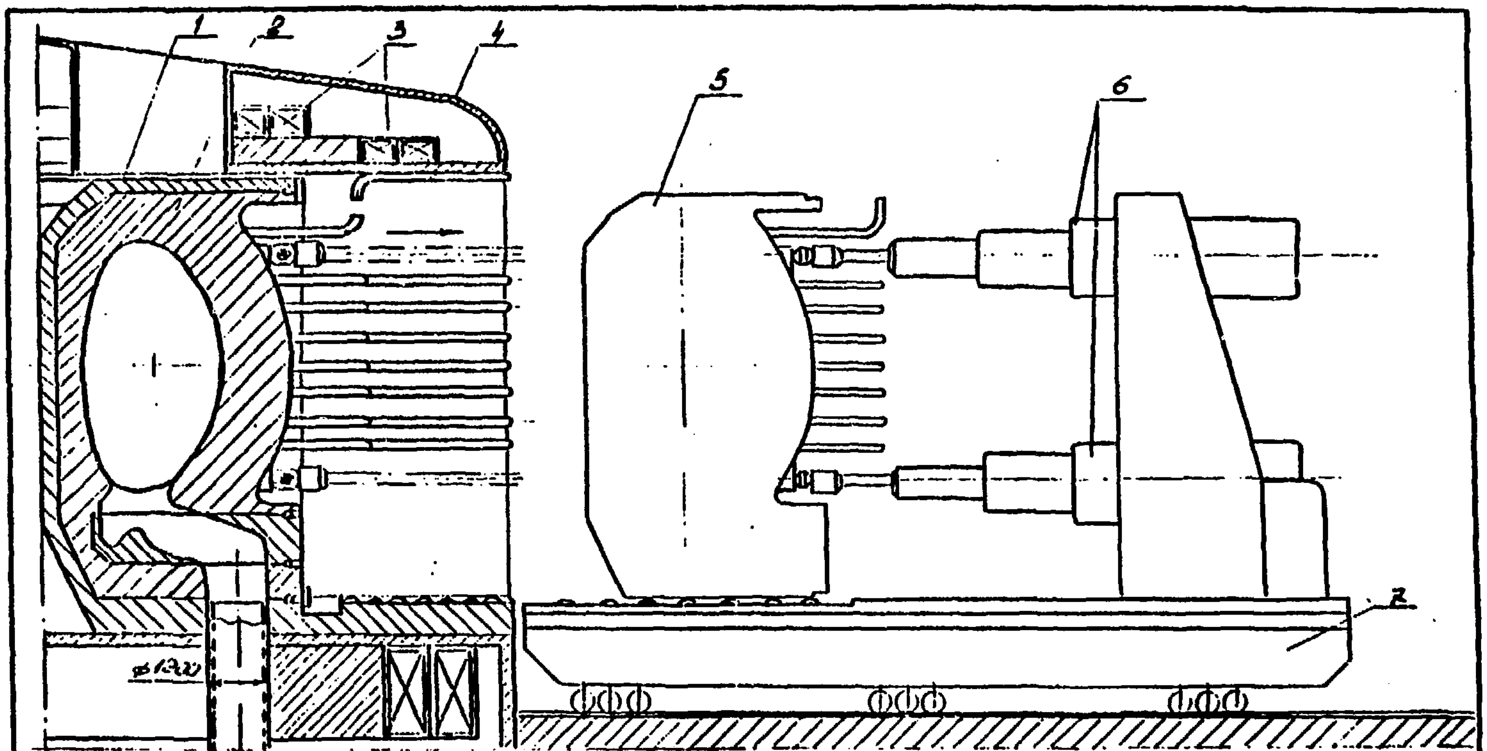


Рис 1 Горизонтальная схема обслуживания термоядерного реактора (ТЯР): 1-стационарная часть бланкета; 2-выемная часть (секция) бланкета; 3-катушки полоидального поля; 4-криостат; 5-положение секции бланкета на платформе; 6-манипулятор для извлечения секций бланкета; 7-самоходная платформа.

Fig. VII-19. Horizontal transporter for sector removal.

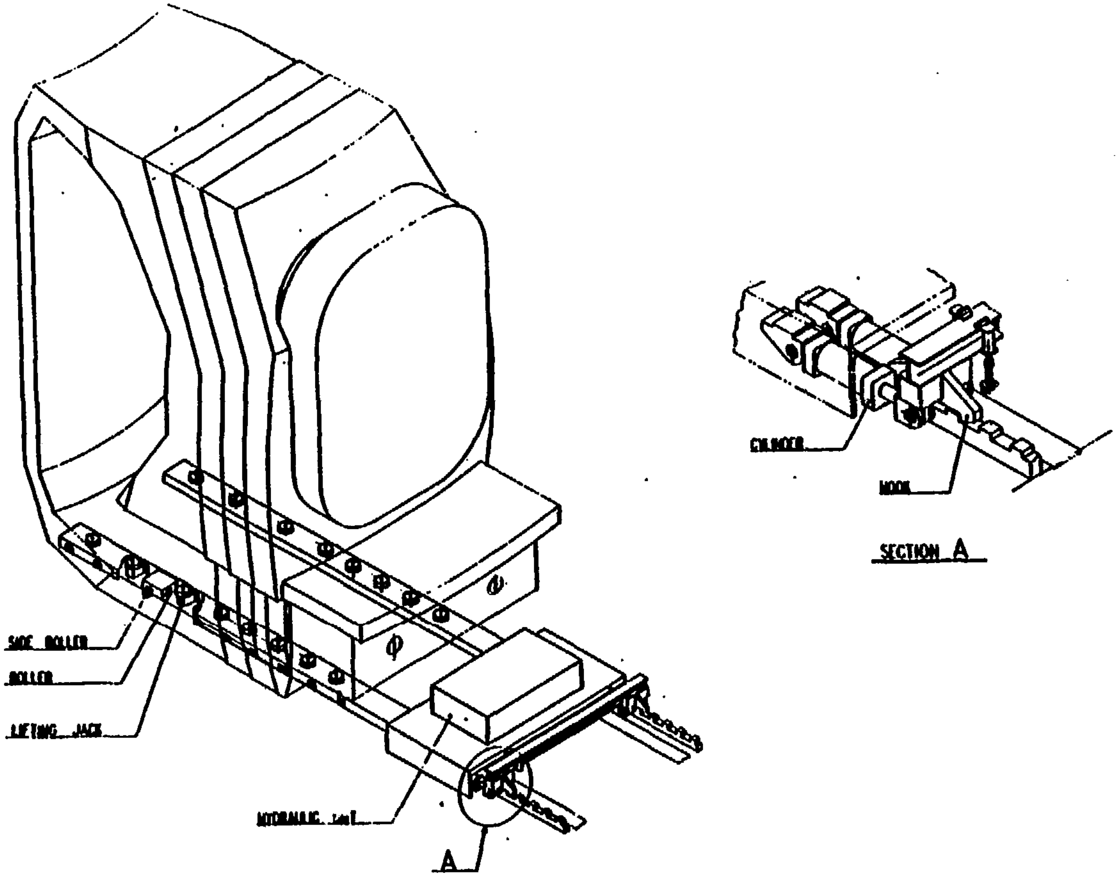


Fig. VII-20. Horizontal sector removal utilizing the divertor port.

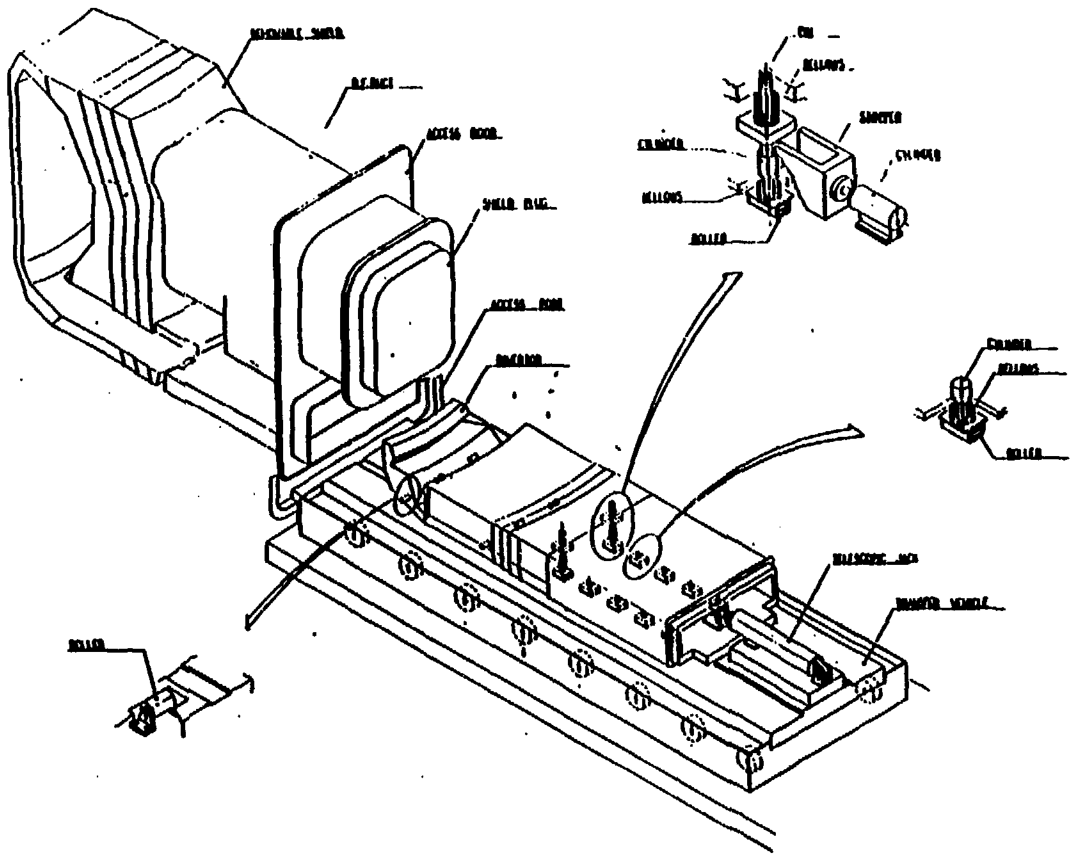


Fig. VII-21. Removal of the divertor module independent from sector removal.

significant since these components are expected to require frequent replacement. With this in mind, a concept was developed for replacing divertors within a containment structure as a means of reducing downtime. Figure VII-22 was taken from reference 4 to illustrate this concept.

Removal of first wall/blanket segments for the vertical access configuration requires overhead maintenance handling equipment. Figure VII-12 shows an example of crane mounted equipment used to remove blanket segments. In this example, the equipment must be capable of providing for rotation as well as lifting of components. Figure VII-23 is a schematic representation of a remotely controlled mechanical arm operating on a first wall segment.

In Fig. VII-24, the reactor configuration has been rearranged so that first wall/blanket components may be removed using pure vertical motion. This approach is similar to that shown in Fig. VII-13.

2.5 Data Base

The data base for fusion reactor maintenance equipment is based on remotely operated machines used in fission plant refueling, nuclear fuel reprocessing, the Joint European Torus (JET) project, the Tokamak Fusion Test Facility (TFTR), various mockup demonstrations at both NET and JAERI and, most recently, the Compact Ignition Tokamak (CIT) project. Reference 6 contains a discussion of most of these data base listings.

During the Phase Two A, Part 2, Workshop, the concept of using an in-vessel manipulator system for maintaining first wall

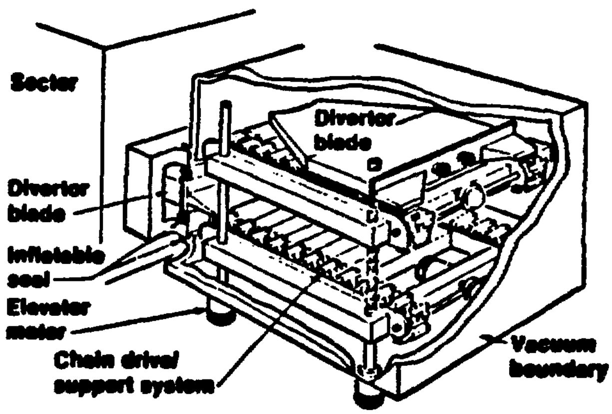


Fig. VII-22. Divertor replacement in a containment structure to reduce downtime.

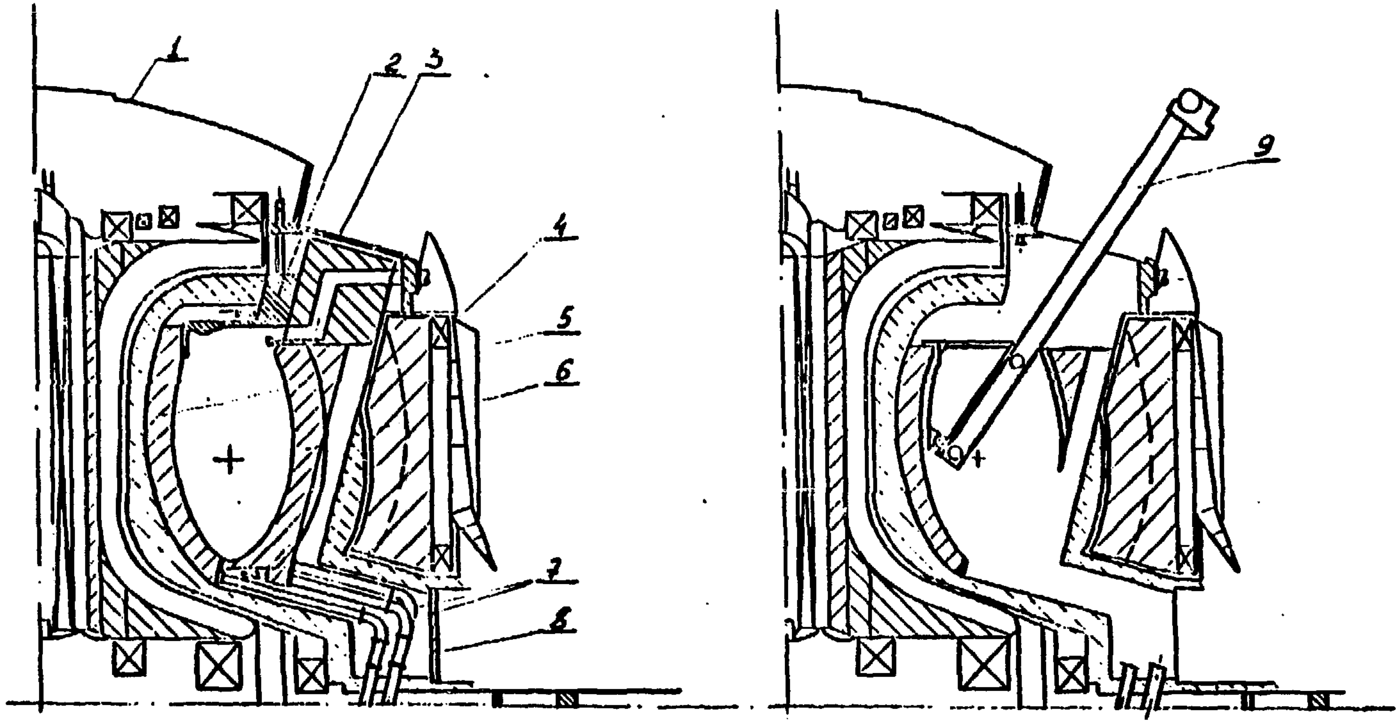


Рис 2. Вертикальная схема обслуживания ТЯР:
1- криостат; 2- дивертор; 3- крышка загрузочного люка; 4- катушка полоидального поля; 5- задний модуль секции бланкета; 6- передний модуль секции бланкета; 7- технологические трубопроводы; 8- крышка герметичная; 9- манипулятор для извлечения секции бланкета.

Fig. VII-23. Remotely controlled mechanical arm operating on a first wall segment.

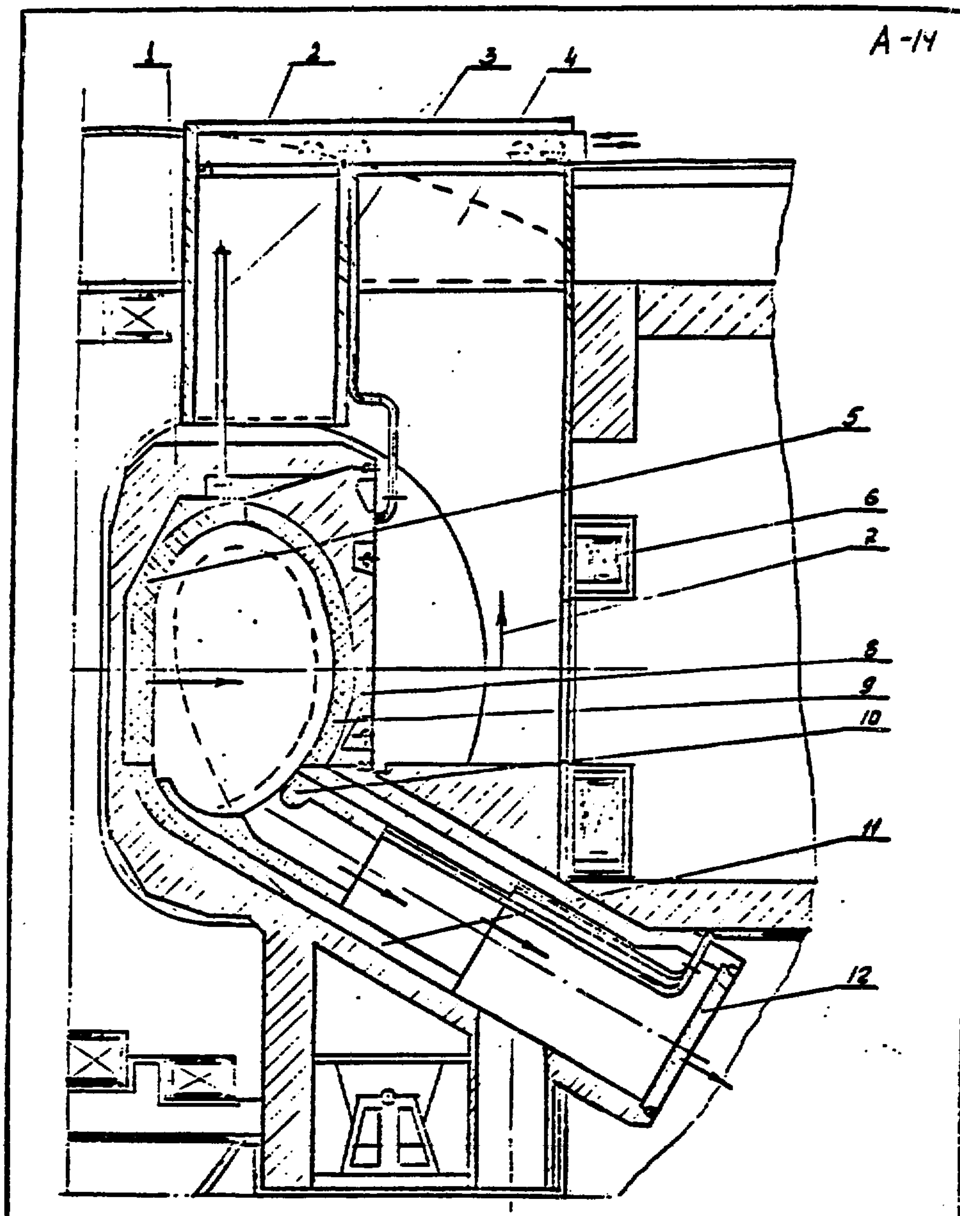


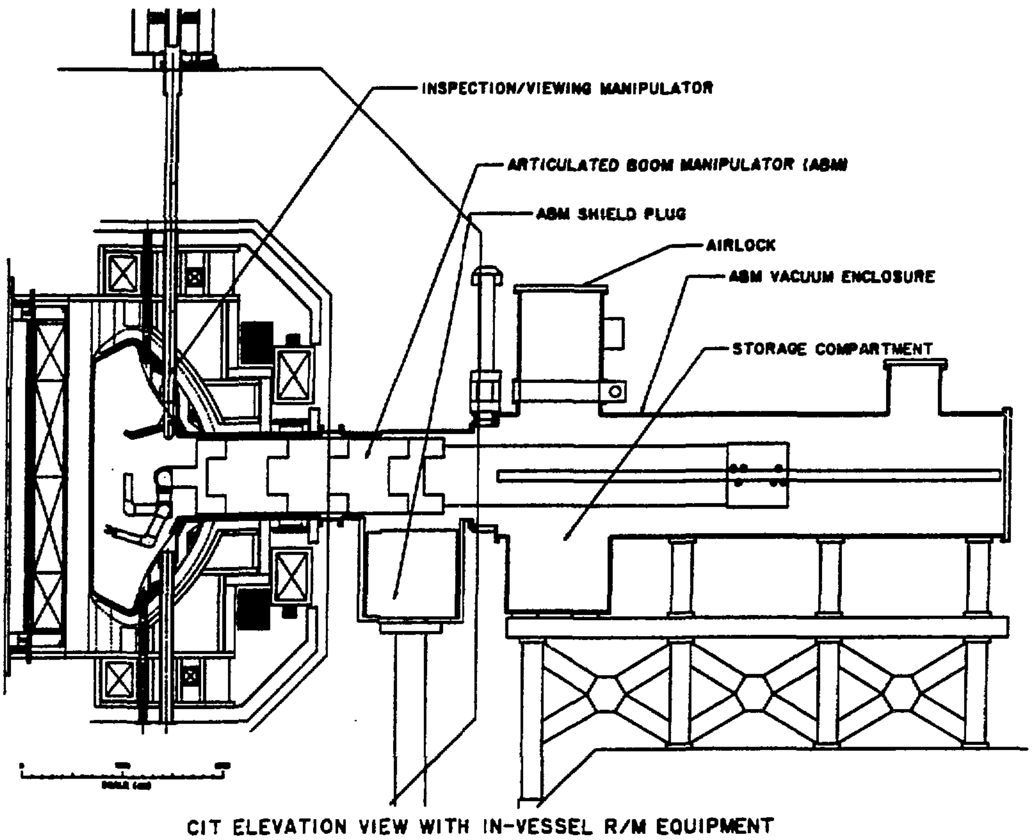
Рис 4. Комбинированная схема обслуживания (вариант 2):
1-Стационарная часть бланкета; 2-защитная плита; 3-технологические трубопроводы; 4-загрузочный люк; 5-задний модуль секции бланкета; 6-катушка полоидального поля; 7-направление извлечения модулей секции; 8-блок защиты переднего модуля секции; 9-передний модуль секции, 10-дивертор; 11-нижний модуль секции; 12-крышка.

Fig. VII-24. Configuration arrangement for vertical removal of in-vessel components.

components was introduced [10]. More recently, it was discussed during this phase of INTOR, although not specifically proposed. The potential advantages of this approach are: minimum impact to downtime if the system is vacuum rated or operable in inert atmosphere, allows maintenance without disturbing peripheral components, and is suitable for unscheduled operations. Figure VII-25 is the concept being developed for the CIT and is a smaller version of a similar concept for TFTR. The articulated boom and the manipulator are contained in an ante-chamber behind a moveable shield plug. The basis for this design and that of TFTR is the successful implementation of this approach for JET.

2.6 Discussion Of Vertical and Horizontal Access Configurations

The INTOR configuration is based on the horizontal removal of torus sectors, including the biological shield, for the replacement of first wall and blanket components. The comparative study between this approach and the vertical approach is based on the configuration developed for the Next European Torus (NET), whereby first wall-blanket components are removed in an oblique (almost vertical) fashion, as shown in Fig. VII-26. The primary differences between these configurations is the location of the PF coils and the emphasis of the maintenance philosophy. The INTOR design was initially developed with simplified maintenance as the primary objective, i.e., straight, radial extraction of complete torus sectors. This led to a configuration where the PF coils were positioned to provide a large window opening for the sector without considering the impact to the cost of the PF system. The vertical access design,



CIT ELEVATION VIEW WITH IN-VESSEL R/M EQUIPMENT

Fig. VII-25. In-vessel manipulator system for maintaining first wall components.

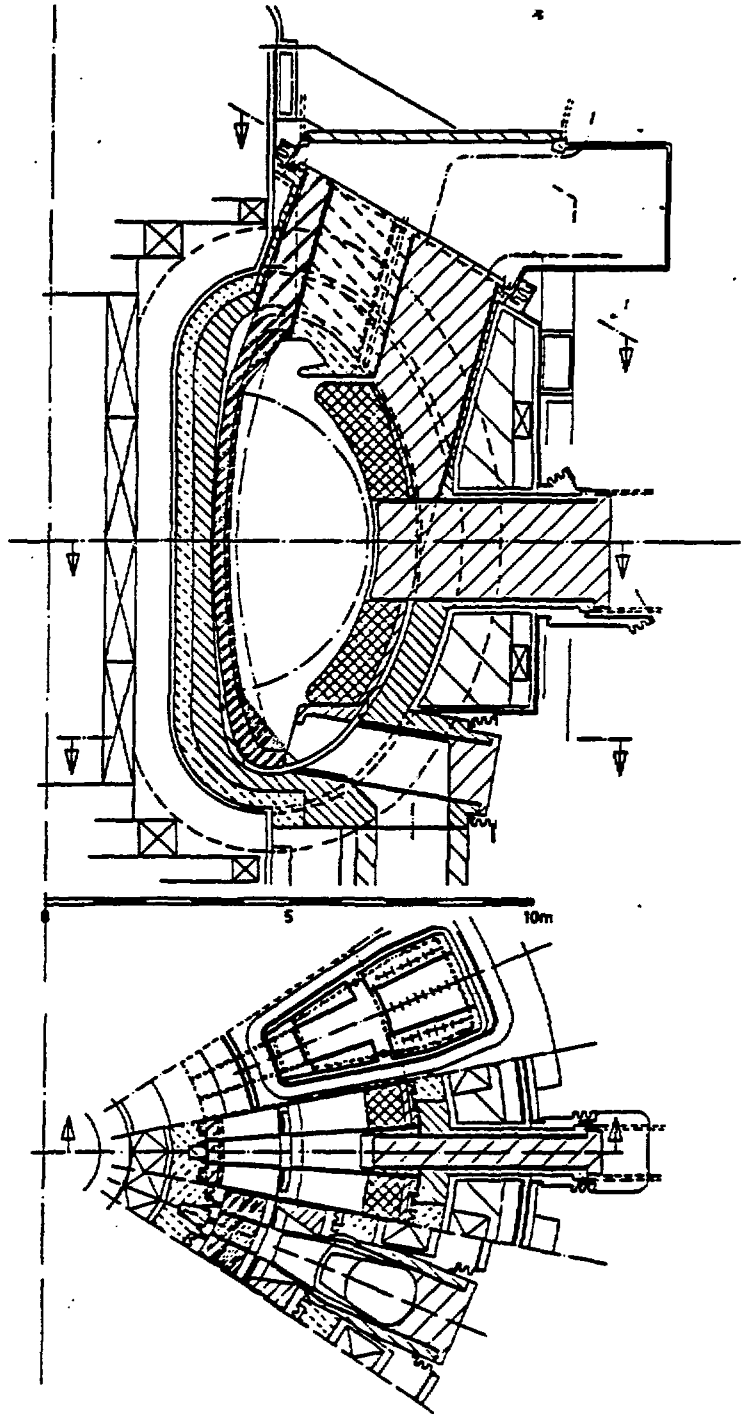


Fig. VII-26. Vertical access configuration based on NET: 3 inboard and 3 outboard blanket segments per sector.

shown in Fig. VII-27 as a modification to the INTOR baseline, has PF coils located to provide a small horizontal window for heating and test modules and a vertical access port for removal of first wall-blanket components.

A comparison of these approaches showed that the latter design had a 25% reduction in the cost of the PF coils; however, most of that reduction was the result of reducing the diameter of the lower outboard coil. The cost reduction contribution from relocating the other coils was approximately 7%.

This modest reduction is the result of a low plasma elongation, k = 1.6. Further study showed that for elongations greater than or equal to 2.0, PF cost reductions are substantial.

Time and motion studies for replacing first wall-blanket components did not reveal any substantial difference in the maintenance time required. Both approaches were within approximately 10% of each other for downtime. Also, it does appear possible to vertically remove internal reactor components without disturbing peripheral equipment such as heating modules and test modules. While this is clearly an advantage, vertical removal requires a greater number of first wall-blanket segments and complex handling equipment. The configuration shown in Fig. VII-27 has 48 outboard blanket segments corresponding to 12 torus sectors. This greater number of segments requires a more complex arrangement of cooling pipes, and the greater number of surface gaps and mechanical connections will reduce the effective blanket surface available in the torus.

Based on the level of design detail to date, it appears that

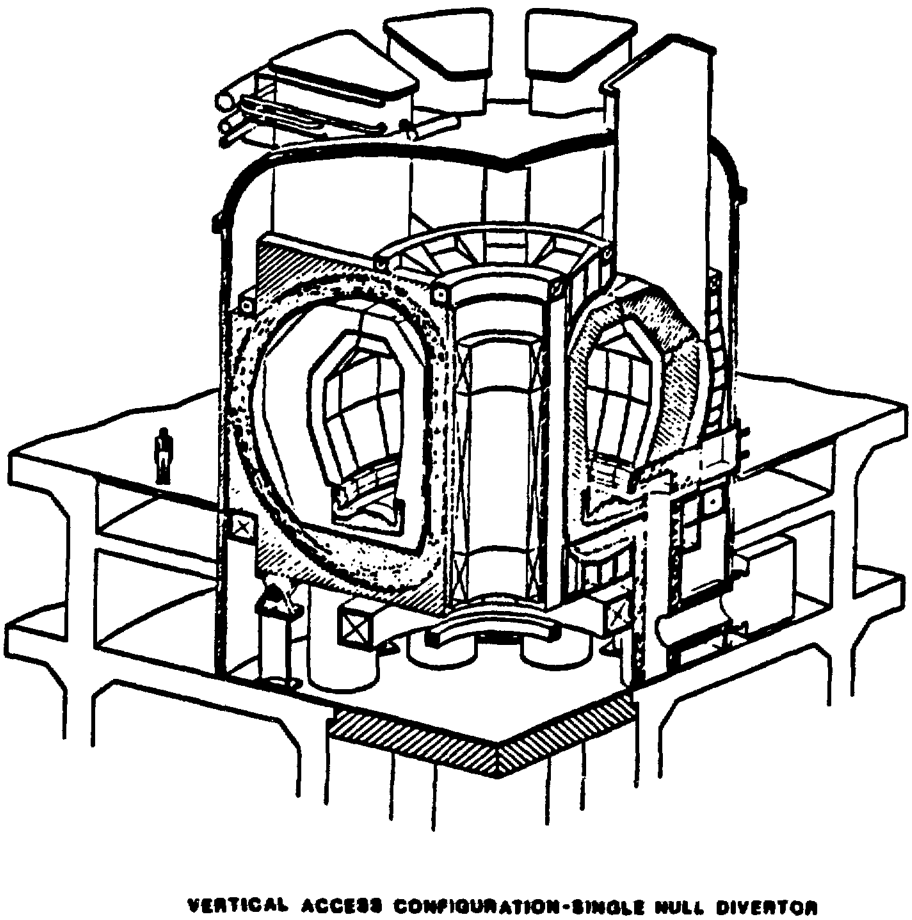


Fig. VII-27. Modified INTOR configuration based on purely vertical access.

both approaches are feasible. For higher elongation plasmas, a vertical access approach with "optimized" PF coil locations should be pursued in conjunction with developing feasible segmented blanket designs. In addition, other variations to the configuration arrangement should be considered in sufficient detail so that their relative merits can be quantified. Figure VII-28 is a configuration based on an "optimized" PF system which combines pure vertical and horizontal access with a canted divertor. A similar arrangement for a double-null divertor design is shown in Fig. VII-29 where inboard blanket segments are removed with the divertors and outboard segments by means of horizontal access.

2.7 Conclusions

A configuration is established as being "vertical" or "horizontal" on the basis of the location of the PF coils. However, there is no exclusive vertical or horizontal arrangement since our comparisons were actually a mix of both approaches. The coils for INTOR were arranged to provide a large window for radially removing torus sectors; the coils for NET (as an example) were arranged near their ideal magnetic locations resulting in a mix of horizontal and vertical access. Since either can be made to work, the choice between the two is not clear cut because both have certain advantages. It is apparent that there are large cost benefits in the PF coil system for ideal coil locations for high elongation plasmas (k greater than or equal to 2.0) and marginal savings for the INTOR case, k = 1.6.

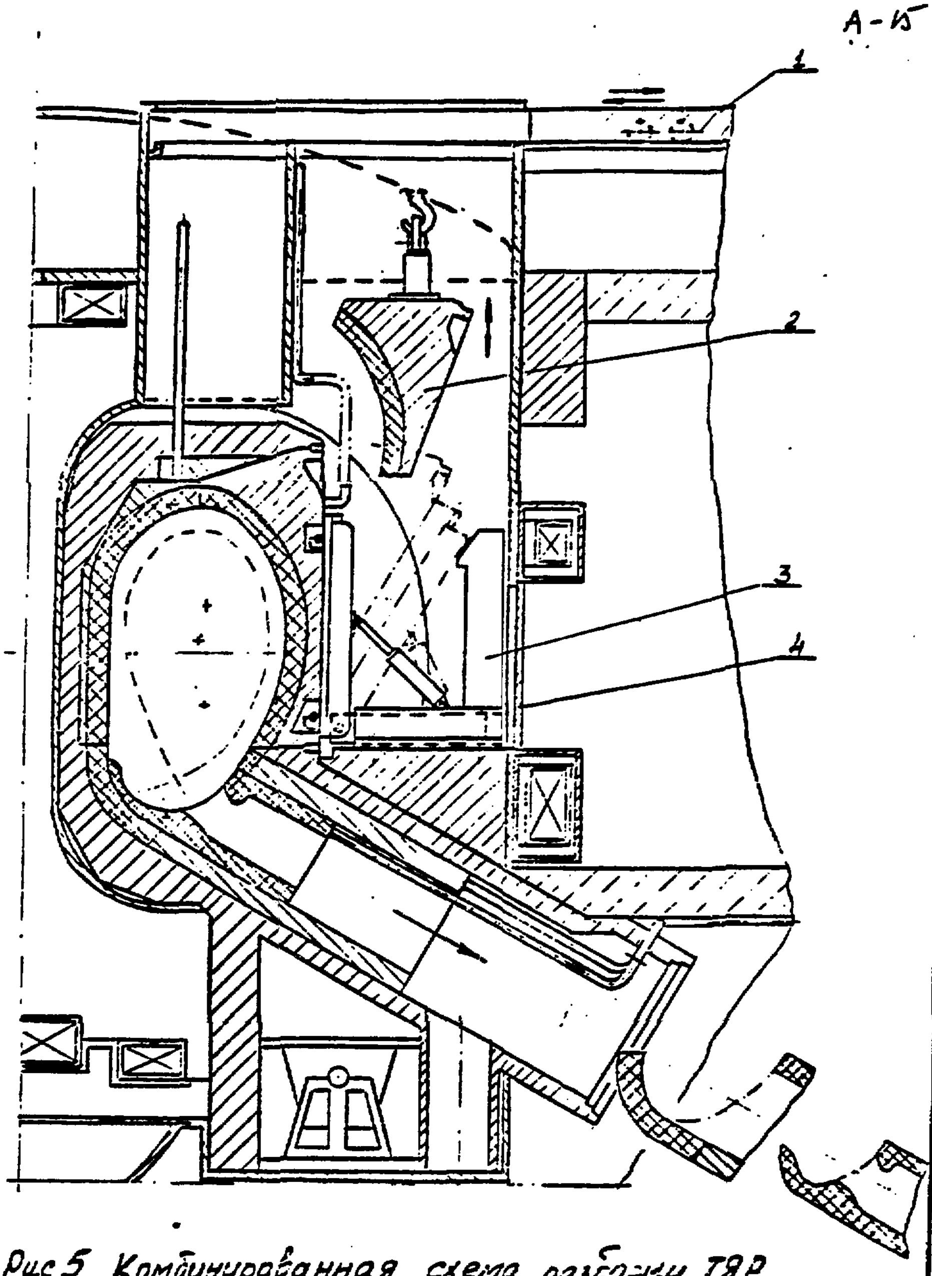


Рис 5 Комбинированная схема разборки ТЯР
1-защитная плита; 2 - извлекаемый передний модуль секции бланкета, 3-манипулятор для извлечения модулей секции бланкета; 4-крышка вертикальная.

Fig. VII-28. Single-null configuration incorporating vertical and horizontal access.

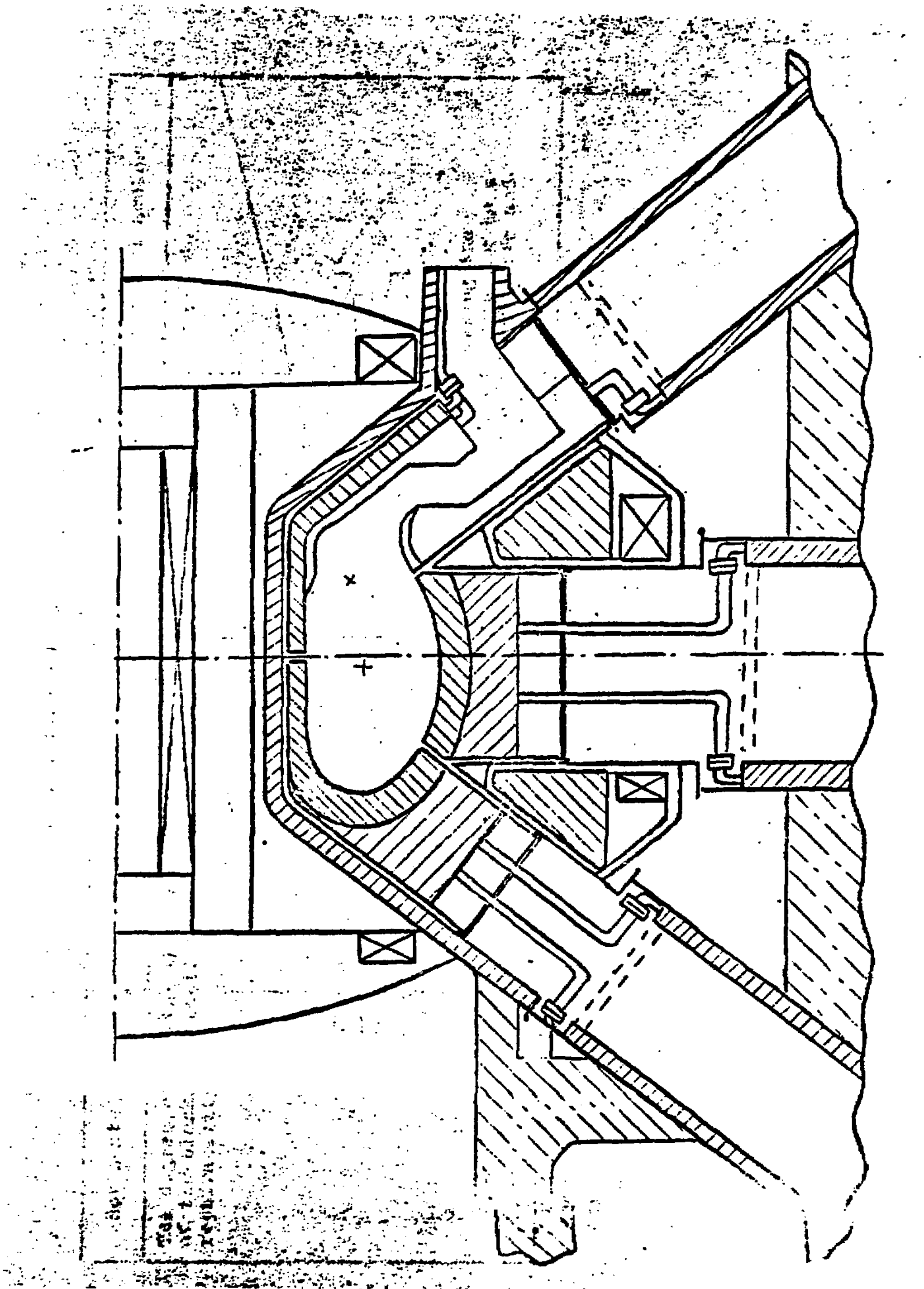


Fig. VII-29. Double-null configuration with full blanket coverage.

If we assume that a new tokamak design will require a higher plasma elongation, the *recommendation* is to arrange the PF coils in a cost-effective manner while providing reasonable midplane access for heating interfaces and test modules (i.e., large port openings). In addition, vertical (not oblique) access between TF and PF coils is desirable to replace first wall structures and possibly divertors. An example of this approach is shown in Fig. VII-27.

If a new design study is not based on a high elongation plasma, it still appears prudent to consider the above approach so that in-vessel maintenance can be accomplished without moving very massive structures such as the bulk shield. The shield, which makes up most of the weight of a sector, could be considered a semipermanent structure. It is not expected to "wear out" like a first wall and is best left undisturbed. The same applies to peripheral equipment like heating and test modules. They should be left in place for maintenance of the first wall and other in-vessel components. The result should be reduced downtime for a carefully thought-out configuration.

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