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Report

REVERSED FIELD PINCH REACTOR STUDY
III. Preliminary Engineering Design

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Abingdon Oxfordshire

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(Euratom/UKAEA Fusion Association)

ABSTRACT

This report, the third of a series on the Reversed Field Pinch Reactor, describes a preliminary concept of the engineering design and layout of this pulsed toroidal reactor, which uses the stable plasma behaviour first observed in ZETA. The basic parameters of the 600MW(e) reactor are taken from a companion study by Hancox and Spears. The plasma volume is 1.75m minor radius and 16m major radius surrounded by a 1.8m blanket-shield region - with the blanket divided into 14 removable segments for servicing. The magnetic confinement system consists of 28 toroidal field coils situated just outside the blanket and inside the poloidal and vertical field coils and all coils have normal copper conductors.

The requirement to incorporate a conducting shell at the front of the blanket to provide a short-time plasma stability has a marked effect on the design. It sets the size of the blanket segment and the scale of the servicing operations, limits the breeding gain and complicates the blanket cooling and its integration with the heat engine. An extensive study will be required to confirm the overall reactor potential of the concept.

The work described in this report forms part of a Reversed Field Pinch Reactor Study undertaken by the following members, whose contributions to the study are gratefully acknowledged:

M Bagatin,[b] L J Baker,[c] S Bobbio,[d] H J Crawley,[a] P A Davenport,[a] F Gasparini,[d] R Hancox (Chairman),[a] A A Hollis,[c] J D Lawson,[e] R Martone,[f] J T D Mitchell,[a] A A Newton,[a] P Reynolds,[a] G Rostagni,[b] J A Shercliff,[g] and W R Spears.[a]

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1. INTRODUCTION

This is the third of a series of reports[1,2,3] covering studies of the Reverse Field Pinch Reactor (RFPR) done in collaboration with AERE Harwell and the Italian Universities of Padua, Naples and Calabria. The Reverse Field Pinch Reactor is a concept which is based on the stable plasma behaviour first observed in Zeta[4] and has been studied since at the Culham Laboratory.[5,6]

This present study has led to a very preliminary concept of an engineering design and layout incorporating the special design requirements peculiar to an RFPR. The dimensions used are those derived from a parameter study.

The preliminary findings of the study indicate the further design work required to establish the initial validity of the concept. The additional detailed design studies required to confirm the engineering feasibility of the concept as a practical fusion reactor will be a task of considerable magnitude. An important finding is that the need for a stabilising shell of time constant $\approx$ 0.5s sets the scale of assembly and servicing operations at a more complex and time consuming level than is indicated by a recent study of a Tokamak fusion reactor.[7]

2. DESIGN REQUIREMENTS OF THE RFPR

2.1 Initial considerations

Where practicable, the concepts for assembly and servicing first developed for the Culham Conceptual Reactor Mark II (CCTRII)[7] are used. In particular, it was decided that:-

(1) The blanket and inner structure would be removable for repair to cater for an expected service life of order 4 years and for the possibility of failures in service. To reduce reactor down-time the removal operations should be simple and conceived with the knowledge that remotely controlled machines would have to be used.

(2) The poloidal field windings should not link the toroidal field coils, in order to simplify construction by eliminating 'in-situ' winding of the poloidal coils. These would, however, have to be wound 'on-site' because of their size ($\sim$ 30m diameter).

(3) The space factor of both poloidal and toroidal field windings would be such that each would occupy no more than half their enveloping toroidal shell. This allows at least one quarter if the outer surface of the blanket/shield structure inside the windings to be used for supporting the blanket and shield, and for the coolant and vacuum ducts and other connections to the blanket system. This also provides space in the windings for their support and service access.

2.2 Special features

There are special features of the Reverse Field Pinch concept which seriously constrain the engineering design of the blanket.

(1) A "passive" conducting shell is required close to the plasma to control the short-time plasma stability, and in addition since the flux penetration time constant of the shell is short compared with the burn time, there are "active," i.e. driven, coils situated at a larger radius outside the breeding blanket.[1] The shell length and the coil dimensions are specified in terms of the plasma radius: the shell length in particular sets a lower limit to the size, weight etc. of each removable blanket segment.

(2) A separate and very effectively cooled 'first wall' is placed in front of the passive shell to absorb the radiation and particle energy from the plasma, which would otherwise by added to the significant neutron and γ ray energy input to the shell. The thermal power rating in the 'first wall' and the passive shell are both about one quarter of the reactor output (see Table 1). A consequence of these high ratings and the fatigue stresses in the structure due to the RFPR pulsed-short-burn operation is that the mean wall loading chosen is only half that adopted for CCTRII. Figure 1 shows the calculated input to the first wall throughout the RFPR cycle; the mean wall loading during the burn is 3.2MW m^{-2} total thermonuclear output$\frac{1}{2}$

(3) These additional components in the "blanket region" of the RFPR require extra space as compared with present Tokamak reactor concepts - putting a premium on using the thinnest possible breeding blanket. A thinner blanket would also minimise the magnetic coil dimensions generally, so reducing the energy supply costs. Pending our own further blanket neutronics studies, we have adopted the thin helium cooled blanket designed by General Atomics,[8] which uses Li_7Pb_2, Li_4SiO_4 and carbon, and achieves a breeding ratio of $\approx$ 1.2, though only 0.6m thick including the inner shield. Also due to the poor conductivity of the breeding materials, the magnetic field penetration time through the blanket will be short for cell diameters $\lessapprox$ 1.0m.

(4) All magnet windings have copper conductors because the pulsed short-burn cycle requires short current rise-times, ruling out the use of present-day superconductors.[1] Also the magnet windings should be as close as possible to the plasma to reduce the energy transferred between the reactor and the store during each pulse. This latter requirement is a constraint on access to the blanket and shield (para. 2.1). In addition, the level of activation of the coil conductors and the reactor structure generally would place a premium in the lifetime reliability of the windings. Replacement of coils, especially the poloidal winding coils below the reactor, would require in effect complete disassembly of the whole active structure of $\approx$ 20,000 tonne and only minor repairs to e.g. coil terminations seem practicable.

3. CONCEPTUAL DESIGN OF RFPR

3.1 General

A key decision in the design is the proportions of a "unit" removable segment which is governed by the following considerations. The passive stabilising shell of copper or aluminium is situated immediately behind the first wall and in front of the blanket proper. Initially it was not judged possible to make a reliable high conductivity electrical contact between shell sections, i.e. at joints between segments. Thus, the minimum length of the removable segment is the length of a shell - specified by Lawson as four times the wall radius - 4a. Also, each segment must be a simple fraction of the toroid and must contain 4N active stabilising coils - N being a whole number.

The choice of segment length has repercussions on the engineering design. For a wall radius a ≡ 1.75m, the weight of the removable blanket is ≈ 50 tonne per metre length and is a practical limitation on the segment length, though this cannot be quantified until the segment structural design is completed. Also, the blanket segment must be coupled with an integral number of toroidal field coils, proportioned and spaced to give an acceptably low toroidal field ripple. The present design is based on fourteen segments with two toroidal coils per segment: the shell length, 7.1 metres, closely matches the required dimension of 4a ≡ 7.0m.

The principal dimensions used for developing the engineering design and drawings are for an air-cored reactor. Both air- and iron-cored 600MW(e) reactors have been investigated in the parameter study[2] but the lowest cost air-cored reactor had a wall radius of 2.16m. Because of the requirement for the length of the passive shell (see above) this wall radius would mean that the blanket segment weight to be moved would be ≈ 600 tonne. However, the air-cored reactor costs are insensitive to reduction in wall radius and near optimum parameters have been derived for r_w = 1.75m which compares with a wall radius of 1.70m for the optimum iron-cored reactor. Not only is the air-cored reactor slightly cheaper but elimination of the iron core removes one constraint from the engineering design, allowing somewhat better access for coil supports, cooling and electrical connections etc.

3.2 Description of RFPR Engineering Design

Figure 2 is a perspective view of the reactor, and the toroidal and poloidal field coils, and inner blanket and shield structure, but excluding external structures - e.g. coil supports, vacuum and coolant ducts. Figure 3 shows the neutron blanket cross section and Table 2 lists some principal engineering data of the RFPR. In sequence from the plasma, there are the first wall or radiation shield, the passive shell, breeding blanket, inner shield, and, not shown in Figure 3, the active stabilising coils, coolant manifolds and the outer shield and magnet coils. The inner shield must limit radiation damage to the outer shield, so that the functional life of the outer shield equals the reactor life. The outer shield has several functions:

(a) to limit radiation damage to the electrical insulation of the coil;

(b) to reduce the activation and radiation levels generally;

(c) to provide a thermal barrier between the hot blanket and shield and the cooler magnet windings;

(d) to provide a pressure boundary between the internal vacuum region and the controlled atmosphere of the reactor hall - as part of the primary vacuum envelope and as the second stage tritium containment - the first stages being in the plasma and blanket regions.

The concept of blanket servicing is that the whole blanket - inner shield assembly of one segment of the reactor should be removed in a horizontal direction through gaps between the coils of the toroidal and poloidal field windings. Figures 4, 5 and 6 show the horizontal and vertical sections of the reactor and indicate how the segment may be with drawn from the outer fixed shield structure. When the reactor is assembled, two toroidal coils occupy positions on each segment midway between the centre line of the segment and the joints between segments. After removing support structure between these and adjacent coils, the two coils are moved along the segment to a position embracing the ends of the adjacent segments. This provides access to the whole length of the outer shield structure enclosing the blanket segment which is to be removed.

As shown in Figure 4, there are triangular-section ribs joining the top and bottom of the fixed outer shield to complete the shielding, strengthen the door frame and provide the vacuum joint round the door in the outer shield. After removing coolant, vacuum and electrical services to the selected segment, cutting the two joints in the radiation shield, referred to later, and releasing the door vacuum joint, the door and segment can be moved out horizontally and in a radial direction with respect to the major axis of the reactor.

From Figures 5 and 6, it will be seen that if poloidal windings are required at large major radius close to the medial plane of the reactor, such windings must be movable to permit withdrawal of the segment. The alternative of removing a segment of the blanket and outer shield complete with the associated toroidal field coils, has been rejected, because:

(i) the total weight involved in the single movement is ∿ 1000 tonne;

(ii) all the toroidal coil services must be completely disconnected;

(iii) the outer poloidal windings must be moved to provide the large vertical clearance for the toroidal coil OD of ≈ 10m.

The general arrangement of the first wall, passive shell, blanket, shield, active coils, ducting and support is shown in Figures 3, 7 and 8. The first wall must be a very efficiently cooled surface with high heat transfer and minimum nuclear heating. Accordingly, it is modelled on the membrane wall tube assembly used in power station boilers, and it is proposed to use niobium for strength and thermal and heat transfer characteristics. The tube axis is parallel to the plasma axis and the spacing is minimised by making the external ribs very short and joining them together by electron beam welding. Because of the very narrow heat affected zone in EB welding, a tube separation of only 1mm or so should be possible, (Figure 3), leaving the strength of the tube wall wholly unaffected. Helium coolant is supplied to each tube via manifolds at each end of the segment. The estimated film temperature drop is ≈ 130K (Table 2), but the wall temperatire rise in only 30K, and useful gas temperatures between 550-800K should be practicable. To eliminate arcing due to the potential difference between the first wall segments, adjacent segments are to be welded together to form a continuous liner. Detailed design of the weld joint is a difficult task in view of the difficulty of cooling the joint and the requirement for many cut and rejoin operations. However, the Joint European Torus[9] incorporates a continuous internal vacuum vessel of similar cross-sectional dimensions and an effective technique for remote welding such structures is being developed and may be applicable.

Behind the first wall, there is the copper (or aluminium)passive shell, conceived as a similar structure but with heavier metal section to increase the electrical time constant. Low conductivity is required in all directions parallel to the shell surface and in this case rectangular tubes are proposed, also welded together by EB welding (Figure 3). Coolant would pass along the tubes again parallel to the segment axis with the principal temperature gradient along the axis. The strength of copper (or aluminium) alloy will limit the shell temperature to ≈ 400^{0}C and coolant temperatures to ≈ 300^{0}C or less, and this will make it difficult to achieve efficient use of the neutron and gamma energy deposited in the shell.

The breeding blanket and primary neutron reflector/shield are behind or outside the passive shell. They are modelled on the "thin" General Atomic blanket design and would be a cellular arrangement to minimise the amount of structural material[10,11]. Preliminary neutronic calculations using ANISN and incorporating the niobium radiation shield first wall are given (Table 2) for both copper and aluminium stabilising shells. The Table gives the zone dimensions, function and compositions and the fractional heating rates, as well as the tritium breeding gains for the two models. The results are preliminary, being based on a provisional "recipe" for the blanket zone compositions and thickness.

Located immediately behind the blanket there are the "active" or powered stabilising coils arranged in sets of four round the plasma axis, and each coil length being $\approx$ a along the axis.[1] Therefore there are 16 active coils per segment of 4a length. They would be manufactured from mineral insulated hollow copper conductors to withstand the radiation levels[12] but otherwise no designs have yet been prepared. Also between the blanket and the toroidal and poloidal field coils, there are the coolant ducts and manifolds and other services to the external circuits, any secondary structure and the outer shielding. To provide radial clearance for moving the toroidal field coils, the mean dimension between the first wall and the inside diameter of the toroidal field coils is 1.8m, as compared with 1.6m used in the parameter study.[2] This results in a small increase in the dimensions of both magnet windings. Also, the toroidal coils are offset from the plasma centre-line by 0.26m, reducing the toroidal field ripple (Table 1) and providing space for the interconnecting manifolds in the coolant circuits (Figure 8).

The detailed requirements of the poloidal and vertical equilibrium field system are discussed in the companion report[3] and Figure 5 shows only a notional configuration for the poloidal windings. There is a rather large gap in these windings between the upper and lower outer coils, through which the segment can be withdrawn. If, as may turn out, this gap has to be reduced to produce the required poloidal field configuration, then provision will have to be made to move sections of the windings. This could also constrain the coolant ducting layout outside the segment.

In addition to the cooling ducts external to the segment, there is a 0.8m diameter vacuum pumping port. During the burn-quench, purified fuel will be let into the reaction space, cooling the plasma and diluting the ash and impurities. At a dilution of 20:1 - that is an increase of particle density from 2×10^{20} to $4 \times 10^{21} m^{-3}$, a vacuum pump speed at each port of $\approx 50\ m^3 s^{-1}$ will restore the particle density to the working level in $\approx$ 5s if the port is straight and $\not>$ 10m long.

4. SOME OBSERVATIONS ON THE RFPR CONCEPTUAL DESIGN

4.1 The Passive Shell

In RFPR, the need to incorporate the passive shell and active coils respectively in front and behind the breeding blanket, for control of MHD instabilities, gives rise to many complications in design and manufacture, and increases the possibility of breakdowns, adding to the reactor servicing problems. It is essential to know accurately the level of control required over MHD instabilities in a plasma of reactor parameters contained in an RFP system, with controlled reversal of the external toroidal magnetic field. It follows that in the same system, we must also know accurately the characteristics of each plasma stabilising device and the effects on those characteristics of such constraints as the length of the passive shell, size of gap between adjacent shells, the size and position of the "active" coils and their power requirements etc.

The ANISN calculation results given in Table 2 are for a 'thin' GA type blanket, but with a niobium radiation shield. They compare the nuclear heating and tritium breeding rates in the whole 'blanket' for cases with shells of (i) 25mm copper shell and (ii) 50mm aluminium. The latter is thicker to compensate for the lower conductivity of likely aluminium alloys at $\approx 300^0$C. There is no significant difference in the energy deposition fractions of the two systems: the total energy per fusion and tritium breeding gain are lower in the blanket with the aluminium shell, though both might be improved by optimisation of the compositions of zones 5 and 6. It seems that the only likely advantage over copper is the reduced energy deposition per unit volume in the aluminium shell, and therefore the larger heat transfer area. Thus optimising the design and cooling of the shell including choosing between aluminium and copper, will require detailed data of the mechanical and electrical properties of candidate alloys under high flux neutron irradiation between 200-500^{0}C, including the long term effects of transmutations (e.g. Cu $\rightarrow$ Ni and Al $\rightarrow$ Mg) on the alloy conductivities.

Because of its effect on tritium breeding and the fraction of energy deposited in the blanket, the thickness of the shell is limited, which means that its time constant is severely constrained: a reduction from 0.5s to 0.1s would be certain to improve the blanket performance, possibly increasing the energy deposition in the blanket zones by 40% or more. However, this would require faster detection of instability growth and higher peak power response from, and higher costs for, the active stabilising coil supplies.

As assumed in paragraph 3.1, it seems there is no possibility of designing a reliable high conductivity joint to make passive shell sections longer than the removable segment. Perhaps the most severe problem would be to hold the close tolerances required for electron beam welding over the dimensions of the shell for a rejoin operation, without spending excessive time in surface preparation. Also, access to the joint region is constrained by the thermal insulation, and by the radiation shield in one direction and the blanket in the other. The complication of removing small sections of blanket, radiation shield and thermal insulation to provide access to the shell would seriously increase reactor servicing down-time. Alternatively, provision of access space by increasing the radial gaps between the radiation shield and the shell or between the shell and the blanket, would increase the overall dimensions of the blanket and shield and thus the size of the magnets and the energy supply systems - and hence the reactor cost.

Generally, the practical engineering of the RFPR concept would be much improved if the size and weight of the removable segments could be reduced by reducing the length of the passive stabilising shell segments. Alternatively, the engineering concept would be greatly simplified if the passive shell could be eliminated and plasma stability provided by the field reversal and the 'active' powered stabilising system.

4.2 General Design Features

Quite large voltages are required round the major and minor circumference of the torus to establish the plasma current and reverse the toroidal magnetic field. In the parameter study[2] a current rise time of 0.5s has been assumed, and a peak loop voltage of $\approx$ 1000V would appear at the first wall/shield, which must be sufficiently resistive to limit the current in it. Also, the whole structure of blanket, neutron shields, cooling piping etc. must be designed to limit induced currents and breakdown and spark erosion damage.

Some other engineering design problems to be noted are:

(i) design of the radiation shield, its cooled joints, the passive shell segments and the associated support structures;

(ii) electromagnetic design of the main poloidal and toroidal field systems and because of the large forces involved, any arrangements required to move coils for servicing access;

(iii) integration of the separate helium cooling systems for the radiation shield, the passive shell and the breeding blanket and shield with the boiler-steam-turbine unit to give a satisfactory thermodynamic efficiency.

The advantage of a thin blanket for a fusion reactor is clear. However, in the RFPR, any advantage gained by this means appears likely to be dissipated by space requirements for the MHD stabilising system and its supporting structure, the several cooling systems and manifolds and connections, and the electrical connections to the active stabilising coils. Also, rectangular ducting has been shown in Figures 7 and 8; it may be necessary to reduce the ducting pressure losses and thus to adopt circular sections and a less compact layout. Generally, the whole "blanket" configuration and assembly is complex, is a difficult design and assembly problem, and is likely to be expensive.

5. CONCLUSIONS

An outline concept for the engineering structure of a 600 MW(e) Reverse Field Pinch Reactor has been described. The inclusion of components believed necessary to provide MHD stability of the plasma has a major impact on the design. First priority must be given to understanding in detail the requirements for MHD stability and the function of the stabilising components of the structure.

The inner blanket structure is divided in 14 removable segments for assembly/servicing. However, the segments are heavy (350 tonne) and their replacement will require some difficult operations - particularly cutting and rejoining the "first-wall-radiation-shield." All the replacement operations will have to be done by remote controlled machines: together these factors are not conducive to achieving a high reactor availibility. Some improvement might be achieved if the segment could be made smaller and if e.g. the passive shell could be eliminated from the design.

As was stated at the outset, this is a very preliminary study; excluding questions of plasma physics, the convincing demonstration of the genera-engineering feasibility of the Reverse Field Pinch Reactor through detailed design of the structural concept described will be a task of considerable difficulty, covering many closely interacting systems, each with special design problems.

Table 1

REVERSED FIELD PINCH REACTOR STUDY

Blanket & Shield Composition, Energy Deposition and Tritium Breeding Ratios

Preliminary Results

ZONE	ZONE RADIAL THICKNESS	DESCRIPTION	COMPOSITION	ENERGY DEPOSITION Fraction of output per fusion event	
-	Metres	-	Material and volume fraction	COPPER SHELL 25mm thick	ALUMINIUM SHELL 50mm thick
1	0.025	Radiation shield (first wall)	Nb 0.3 Void 0.7	0.257	0.277
2	0.05	Passive shell	Cu or Al (as stated in next col). Remainder void	0.231	0.199
3	0.135	Front blanket	Li_7Pb_2 0.60 C 0.20 Inconel 0.04 Void 0.16	0.249	0.261
4	0.18	Rear blanket	Li_2O 0.25 C 0.60 Inconel 0.04 Void 0.11	0.214	0.212
5	0.20	Shield	St. Steel 0.80 H_2O 0.20	0.040	0.042
6	0.21	Coils & ducting	Cu 0.025 St. Steel 0.05	0.005	0.005
Tritium breeding per fusion neutron - T_6			-	1.06	0.97
T_7			-	0.12	0.12
TOTAL			-	1.18	1.09
Energy deposited per 14.1MeV fusion neutron			MeV	16.35	15.06
Total output per fusion event			MeV	19.9	18.6

Note: Calculation model uses ANISN with S_8P_3 approximation and nuclear data from ORNL-TM-5249, (March 1976).

Table 2

REVERSED FIELD PINCH REACTOR STUDY

Some Engineering Dimensions and Parameters

REACTOR DIMENSIONS		
Net output	600	MW(e)
Major radius	16.0	m
Minor or wall radius	1.75	m
Aspect ratio	8.9	
Mean wall loading	2.2	MW m^{-2}
BLANKET SEGMENT (Removable)		
Number	14	
Height overall	5.6	m
Length overall	7.8	m
Width (excluding ducting)	7.3	m
Weight	370	tonne
TOROIDAL FIELD COILS		
Number	28	
Mean radius	4.2	m
Cross section	1.3x1.3	m
Weight of coil	350	tonne
Coil centre offset from minor acis	0.26	m
Toroidal field ripple (maximum)	± 0.02	
Winding space factor	0.37	
POLOIDAL FIELD COILS		
Major radius of equivalent toroidal shell	16.0	m
Outside minor radius of equivalent toroidal shell	5.88	m
Radial thickness of equivalent toroidal shell	0.88	m
Winding space factor	$\approx$ 0.5	
RADIATION SHIELD		
Niobium tubing	15mm bore	3mm wall
Wall loading during burn	3.2	MW m^{-2}
Heat input to wall facing plasma during burn	630	kW m^{-2}
Helium cooling heat transfer coefficient	3	1W m^{-2}
Film temperature drop	135	K
Wall temperature drop	30	K
Peak loop voltage on shield	$\approx$ 1000	V
Shield resistance at 770K	410	$\mu\ \Omega$
PASSIVE STABILISING SHELL		
Copper (hollow)		
Equivalent thickness	25	mm
Time constant at 670K (L/R for θ currents)	$\approx$ 0.5	s
Coolant helium		
VACUUM PUMPING		
Segment volume to be pumped including ducting	75	m^3
Flush density	4×10^{21}	particles m^{-3}
Working density	2×10^{20}	particles m^{-3}
Exhaust duct diameter	0.8	m
Pump speed	50	$m^3\ s^{-1}$
ACTIVE STABILISING COILS		
Number per segment	16	
Material - mineral insulated copper		
Coil dimensions: length along plasma axis	1.5	m
round plasma circumference	3.0	m

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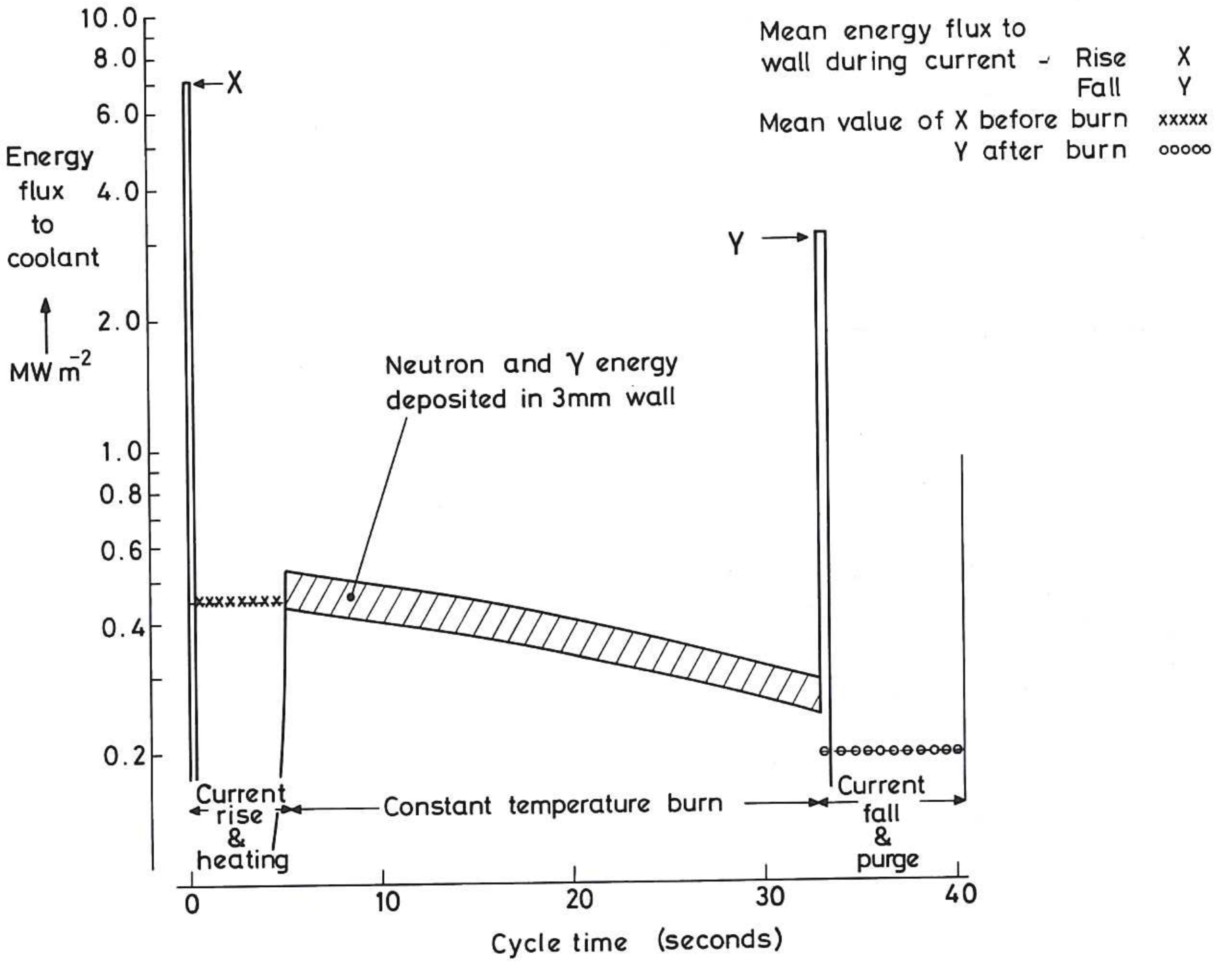

Figure 1. Energy Flux to R.F.P. Reactor First Wall during cycle, expressed as MW per square metre of tube wall inner surface.
Mean wall loading during burn - 3·2 MW m^{-2}

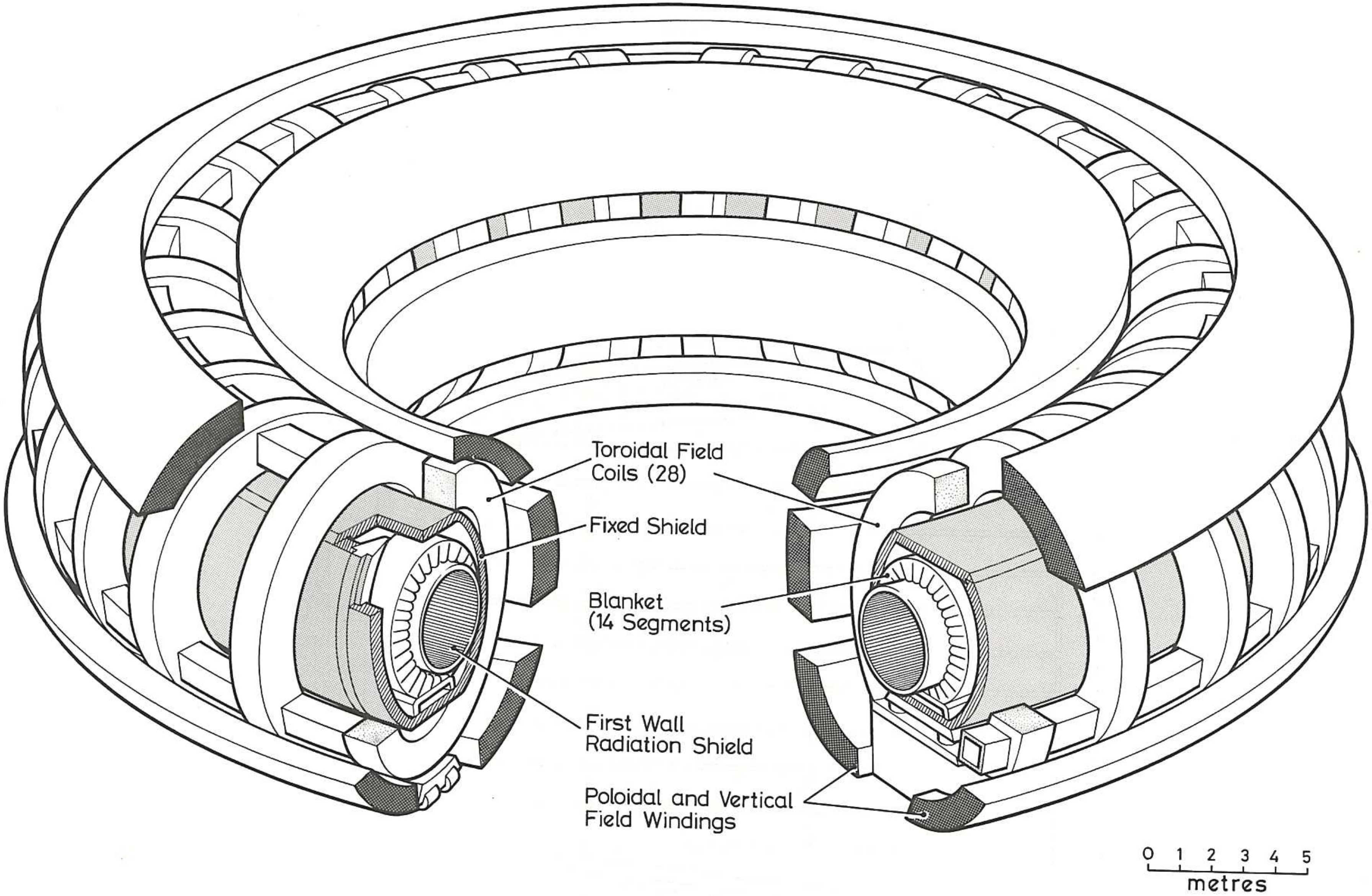

Fig.2 Artist's Impression of Reversed Field Pinch Reactor

Blanket services and magnet structure are omitted

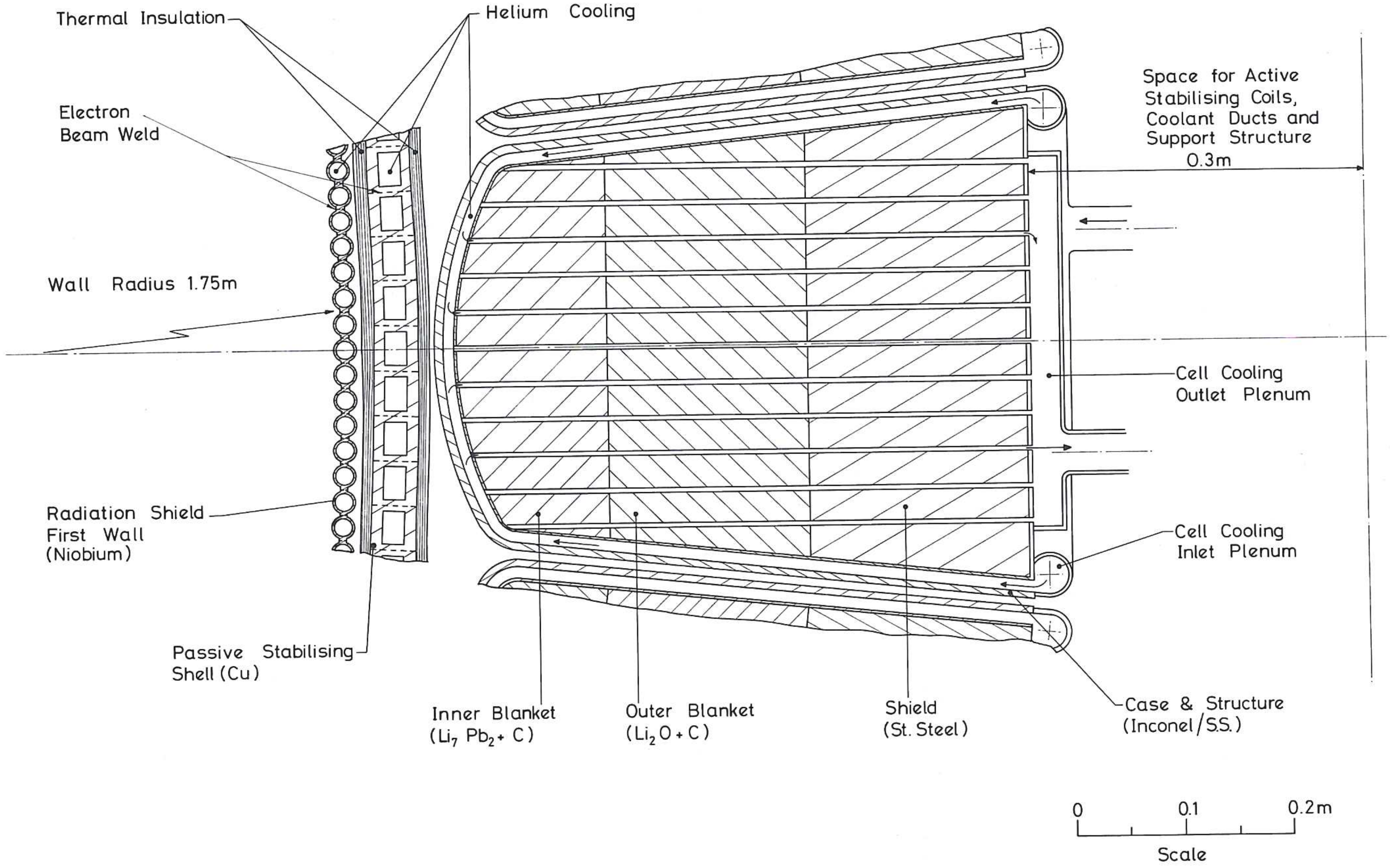

Fig. 3 Schematic Arrangement of Principal Blanket Components of Reverse Field Pinch Reactor

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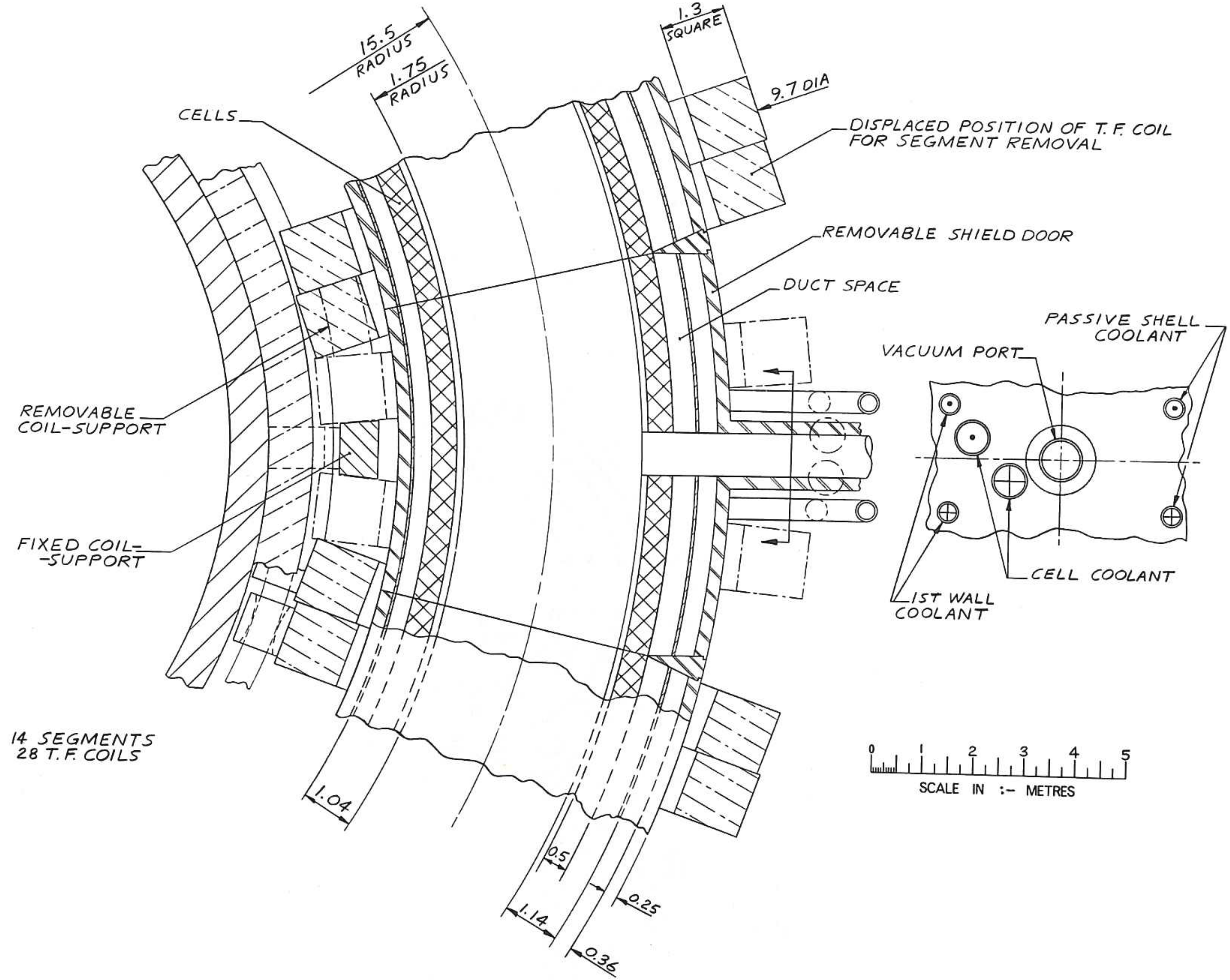

FIG. 4. MEDIAN PLANE SECTION OF THE RFP REACTOR SHOWING MOVEMENT OF THE TOROIDAL FIELD COILS FOR BLANKET SEGMENT REMOVAL

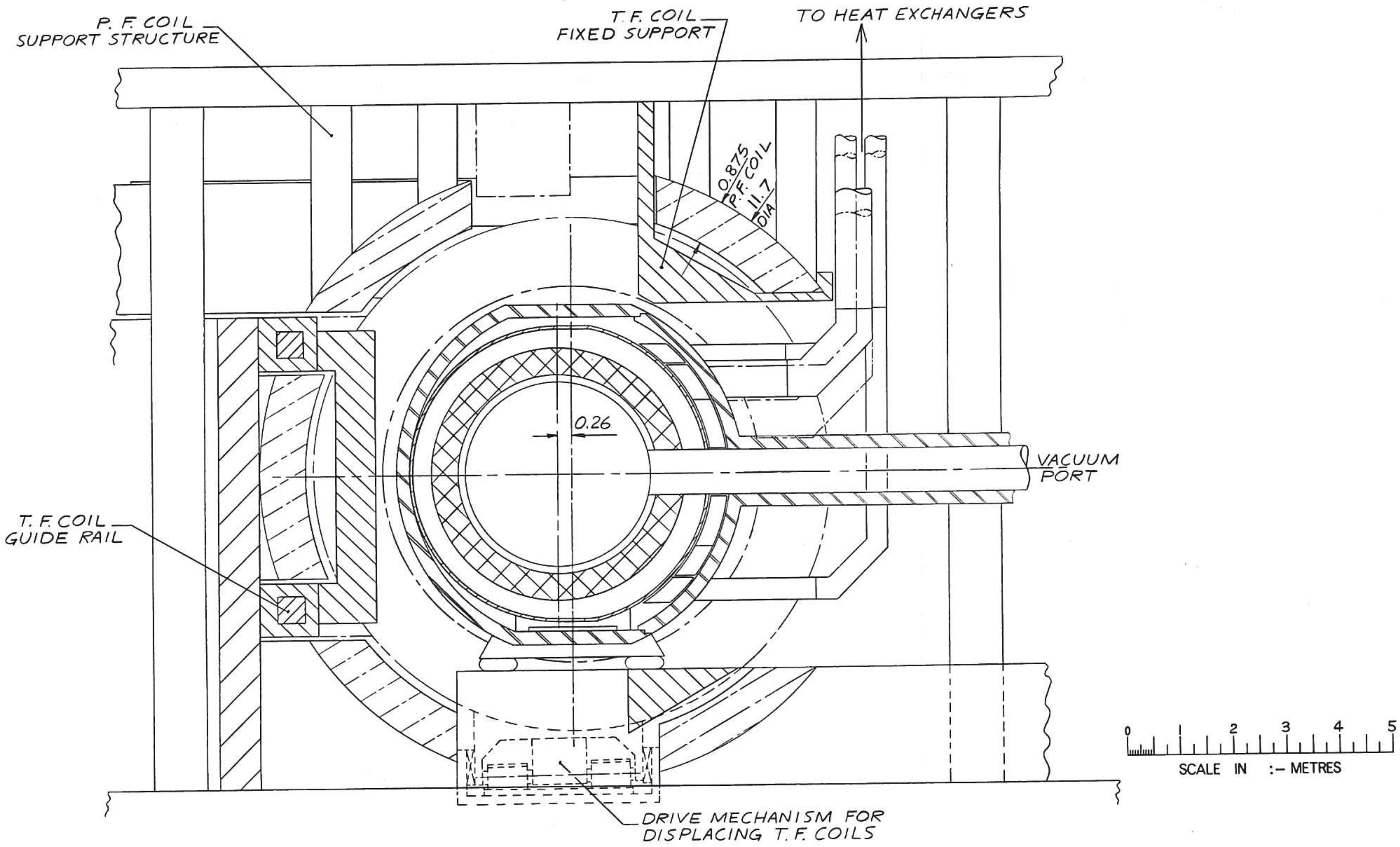

FIG. 5. VERTICAL SECTION OF THE RFP REACTOR THROUGH THE CENTRE OF THE BLANKET SEGMENT

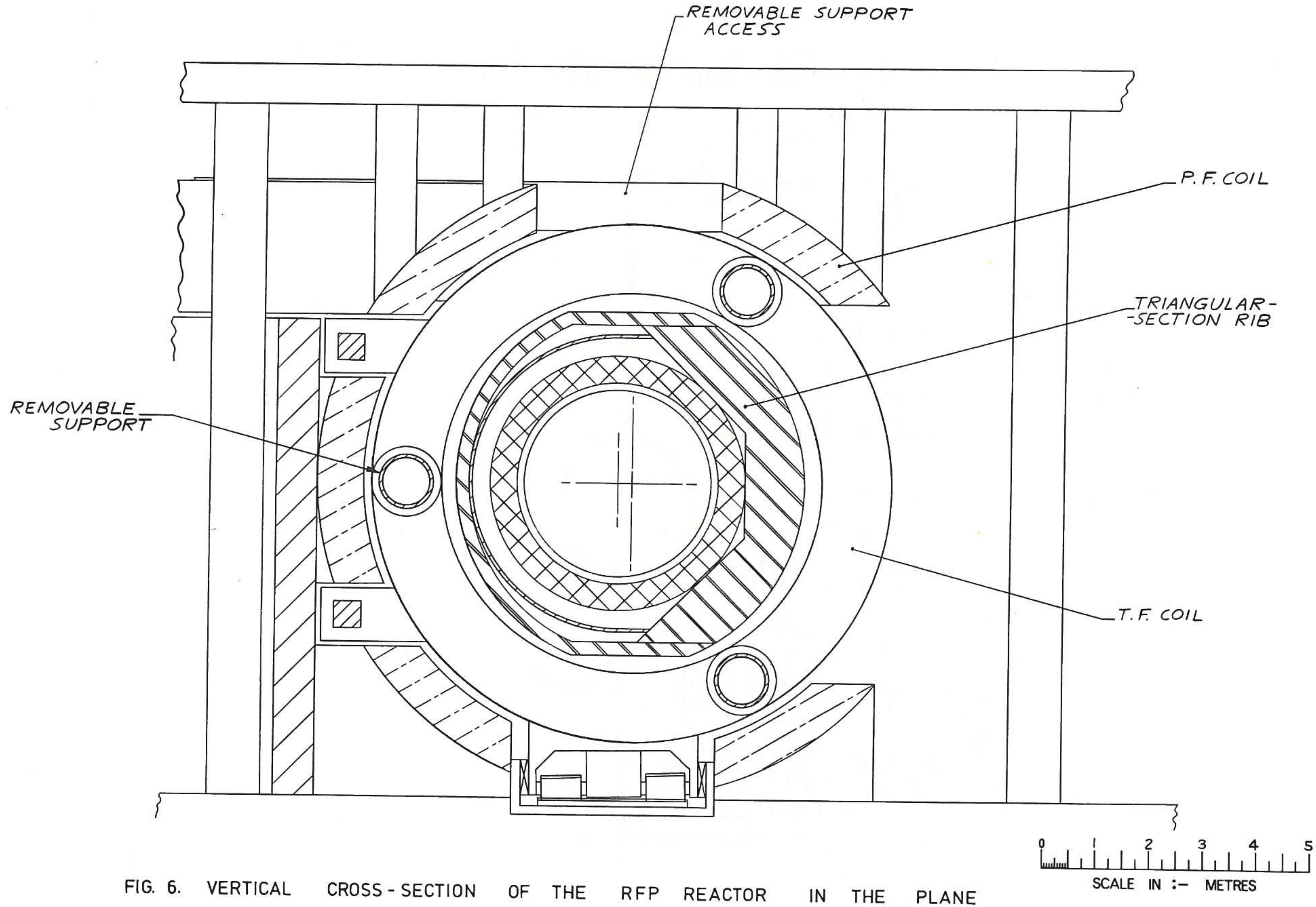

FIG. 6. VERTICAL CROSS-SECTION OF THE RFP REACTOR IN THE PLANE BETWEEN TWO BLANKET SEGMENTS

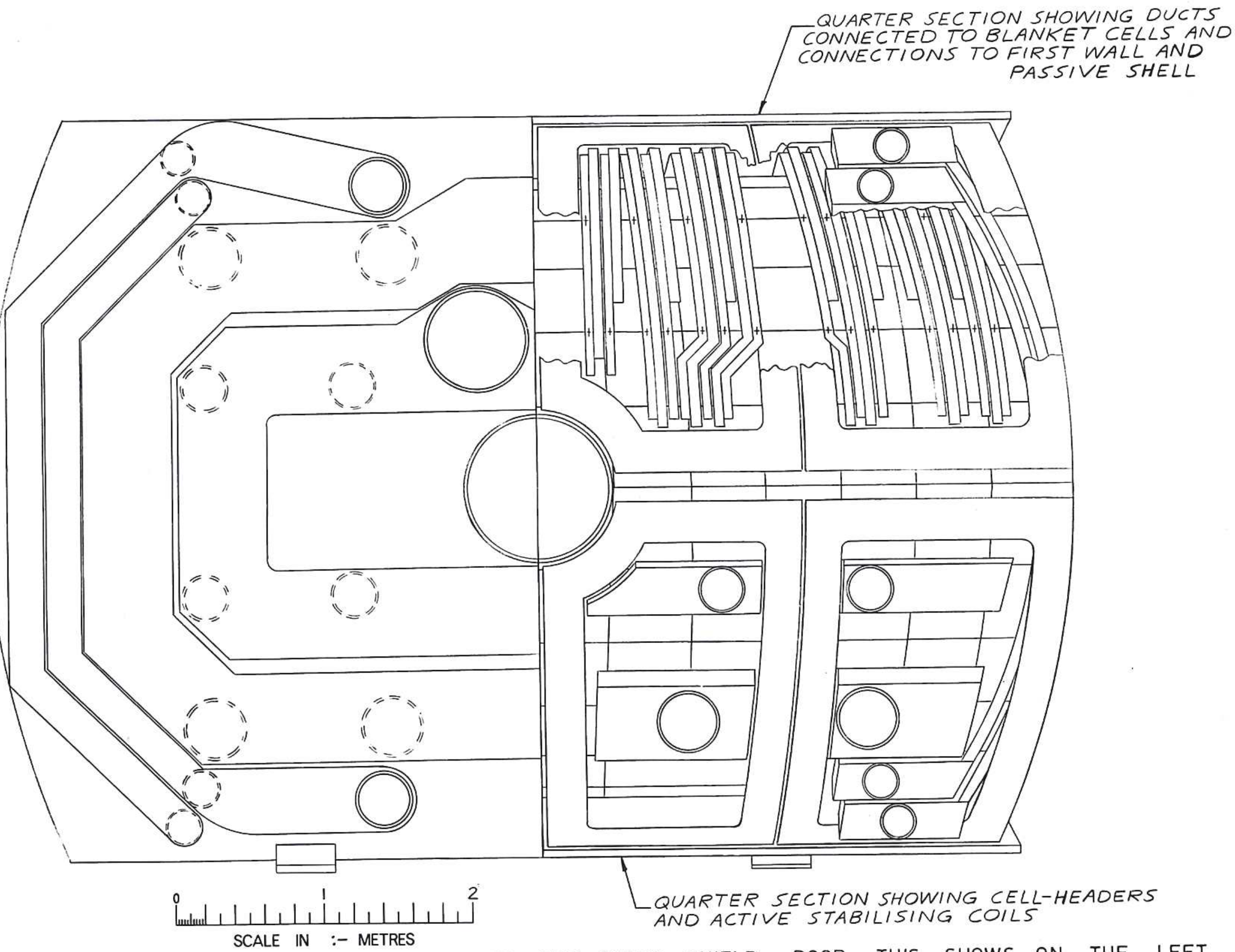

FIG. 7 ELEVATION OF A BLANKET SEGMENT WITHOUT THE OUTER SHIELD DOOR. THIS SHOWS, ON THE LEFT, THE MANIFOLDS CONNECTING THE INTERNAL SYSTEMS TO THE MAIN EXTERNAL CONNECTIONS OF THE SEGMENT. THE TWO QUARTER SECTIONS SHOW THE INTERNAL DUCTING AND THE ACTIVE STABILISING COILS.

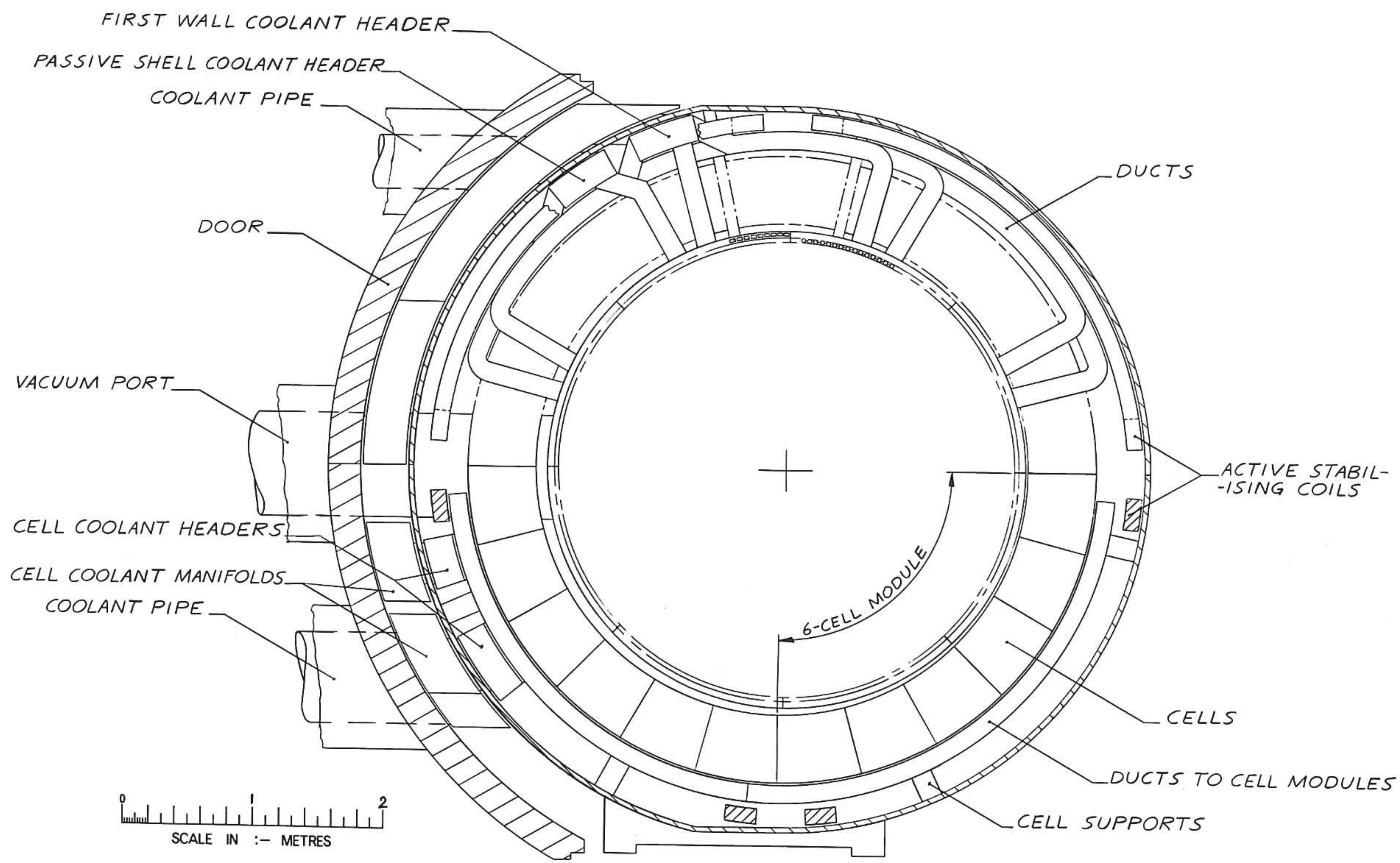

FIG. 8. CROSS-SECTIONS OF A REMOVABLE BLANKET SEGMENT. THE LOWER HALF SHOWS THE SUPPORTS FOR THE CELLS AND THE LAYOUT OF THE CELL COOLING CIRCUITS. THE UPPER HALF SHOWS THE FIRST WALL AND PASSIVE SHELL COOLING CIRCUITS.

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