Superconducting Magnet Systems
in EPR Designs

A.F. Knobloch

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Abstract

Tokamak experiments have reached a stage where large scale application of superconductors can be envisaged for machines becoming operational within the next decade. Existing designs for future devices already indicate some of the tasks and problems associated with large superconducting magnet systems. Using this information the coming magnet system requirements are summarized, some design considerations given and in conclusion a brief survey describes already existing Tokamak magnet development programs.

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1. INTRODUCTION

The technology of superconducting magnets for fusion application would have a slow start if it were dependant only on near term experiments inevitably needing it. Its proper development obviously is costly time-consuming and also risky since it must rely on extrapolations from present plasma physics achievements. It shares, however, these features with the Tokamak oriented fusion technology as a whole. Therefore, in case there is a strategy or only a cònfidence in progress towards a Tokamak reactor, there has to be an appropriate effort in this part of technology, too. The appearance of experiments like TFTR or JET gradually reveals the real engineering and sometimes political tasks of designing and building a large integral Tokamak magnet system and to make its several 100 MW pulsed power supply available. Beyond a power requirement of this order of magnitude it may be hard in the future to convince anybody of the necessity to increase further the electricity supply for fusion apparatus while promising finally to produce power but not to waste it. In addition to this argument time schedules for experiments like TFTR or JET set the case for starting substantial work towards fusion-related superconducting magnets for the next machine generation.

Introducing the superconductor technology into the complicated Tokamak fusion line in a sound manner requires due to almost unanimous opinion some previous technological feasibility demonstration by means of a large magnet test able to simulate all relevant operational features of an integral Post-TFTR or Post-JET magnet preferably in a conservative approach which means today e.g. the choice of NbTi as the superconductor material and a design for full cryogenic stability [1]. This demonstration should be achieved in a separate technical test facility preceeding the application in an experiment. Here one should mention that only a superconducting toroidal magnet system not much below JET size would yield extrapolable design information for a later TNS or EPR machine[2] whereas much smaller superconducting magnets or test facilities would not. While designing now and building the next (and possibly last) generation of experiments with normal conducting toroidal magnets [3] already much important know-how is being gathered for a relevant specification and construction of a large superconducting Tokamak magnet. This know-how includes the specific field and stress calculations[4], structural design, definition of the poloidal field configuration including the plasma[5], maintenance considerations, aspects of radiation shielding, remote handling and failure analysis. On top of this basic know-how additional information is required for a safe design and construction of superconducting fusion magnets namely on conductors, winding, cooling, monitoring and protection under the Tokamak specific boundary conditions like rapidly varying external magnetic fields, high currents and voltages, cyclic stresses and very large dimensions and fields. It is this latter group of specific problems which call for a basic development program for fusion superconducting magnets additionally to the costly task of establishing confidence in these magnets by a relevant integral test set up[6].

The following lecture deals in the above mentioned sense with the systems aspects of the integral magnet of future Tokamak machines and will add some remarks on superconducting magnet development programs and possible international collaboration.

2. REQUIREMENTS FOR NEXT GENERATION CONFINEMENT EXPERIMENTS, TNS AND EPR MACHINES, COMPARISON WITH REACTOR DATA.

2.1 Toroidal field coils

According to designs available at present the magnet requirements of large future Tokamak machines are generally the following: to establish a steady state toroidal field and pulsed poloidal fields for the confinement of a plasma ring with a diameter of 6 ÷ 26 m, an aspect ratio of 3 ÷ 4 and currents between 3 and 15 MA.

The toroidal magnet generally has to be designed for an axial flux density between 3 and 6 T. Table I shows a data comparison for a selected number of designs of next generation experiments through EPRs up to full scale reactors. Due to the introduction of a blanket and shield zone the usage of the toroidal field decreases considerably for EPRs and reactors compared to experiments. All superconducting windings in EPRs and reactors have to be radiation shielded to the extent that the stabilizing material in the innermost windings during the accumulated operational time will suffer a resistance increase less than the unirradiated value at most. This condition is roughly consistent with sufficient protection of the superconductor itself and of the coil and conductor insulation in the case of epoxy[18, 19, 20]. The number of toroidal coils is defined by the permitted toroidal field ripple at the outer plasma edge, in general about < 1 ÷ 2 % and on the required access between the outer coil parts for

TABLE I

	R (M)	a (M)	I_P (MA)	B_0 (T)	W_{MT} (GJ)	$\hat{P}_{MT}$ (GW)	N
JET	2,96	1,25/2,10	4,80	3,40	1,45	0,33	32
GA ITR (TNS)	3,50	0,80		4,00		-	16
ORNL ITR (TNS)	5,00	1,25	3,30	4,40		-	
T 20	5,00	2,00	6,00	3,50	6,00	1,20	24
ANL EPR 1975	6,25	2,10	4,80	3,40	15,60	-	16
ANL EPR 1976	6,25	2,10	7,58	4,32	30,00	-	16
ORNL EPR 76	6,75	2,25	7,20	4,90	29,00	-	20
JXER	6,75	1,50	4,00	6,00	50,00	-	16
UWMAK II	13,00	5,00	14,90	3,67	223,20	-	24
UWMAK III	8,10	2,70	15,80	4,05	108,00	-	18
CULH. MK II	7,40	2,10/3,68	11,60	4,10	~57,00	-	20

7,8,9,10,11,12,13,14,15,16,17

auxiliary heating devices and maintenance and repair concerning the plasma chamber and the blanket and shield region.

Table II gives some further toroidal field coil data, which show the size dependent increase of the forces, a major incentive to go for a noncircular coil shape adopted throughout. The constant tension, so-called modified

TABLE II

	SHAPE	DIM. (M)	AT(MAT)	I_{COIL}(KA)	B_{MAX}(T)	j_{AV}($\frac{KA}{CM^2}$)	F_T(10^6N)	F_C(10^6N)	WEIGHT(T)
JET	MOD. D	3,12 x 4,90	1,42	66,4	6,9	1,6		18,4	12,0
GA ITR (TNS)	MOD. D	4,50 x 7,20	4,25	6,0	8,0	3,0			72,0
ORNL ITR (TNS)		4,50 x 6,50							
T 20	OVAL	5,40 x 8,30	3,60	120,0	7,8	1,6	15,0	61,2	66,7
ANL EPR 1975	MOD. D	7,70 x11,90	6,50	10,0	7,5	2,4	49,3	178,2	175,0
ANL EPR 1976	MOD. D	7,78 x12,60	8,37	60,0	10,0	3,7	82,7	358,0	208,0
ORNL EPR 76	OVAL	7,40 x 9,70	6,30	78,0	11,0	4,0	76-360	220,0	
JXER	MOD. D	7,00 x11,00			11,5				
UWMAK II	EXTENDED D	18,00 x27,50	9,99	11,5	8,3	6,0	43,8		710,0
UWMAK III	REDUCED D	12,50 x23,50	9,77	10,9	8,75	4,8	34,2		174,0
CULH. MK II		9,80 x17,50	7,50		8,0				

D-shape, however, has been foreseen only for JET, one of the TNS designs, JXER and the ANL-EPRs. T20 and other designs deviate more or less from the modified D. This is an indication of both an adaptation to the inside space requirements and of economical considerations. As will be shown later, the constant tension D-shape couples the coil shape with the magnet major diameter and will not generally be the most economical solution. In a torus of D-shaped coils the axial field tries to compress the array of straight inner coil sections radially towards the torus center. The corresponding force (see table II) is so large, that it almost predetermines the mechanical structure of a Tokamak machine, in that it calls for a strong central force bearing column or cylinder to support the toroidal field coils. For large superconducting toroidal field coils the magnitude of the

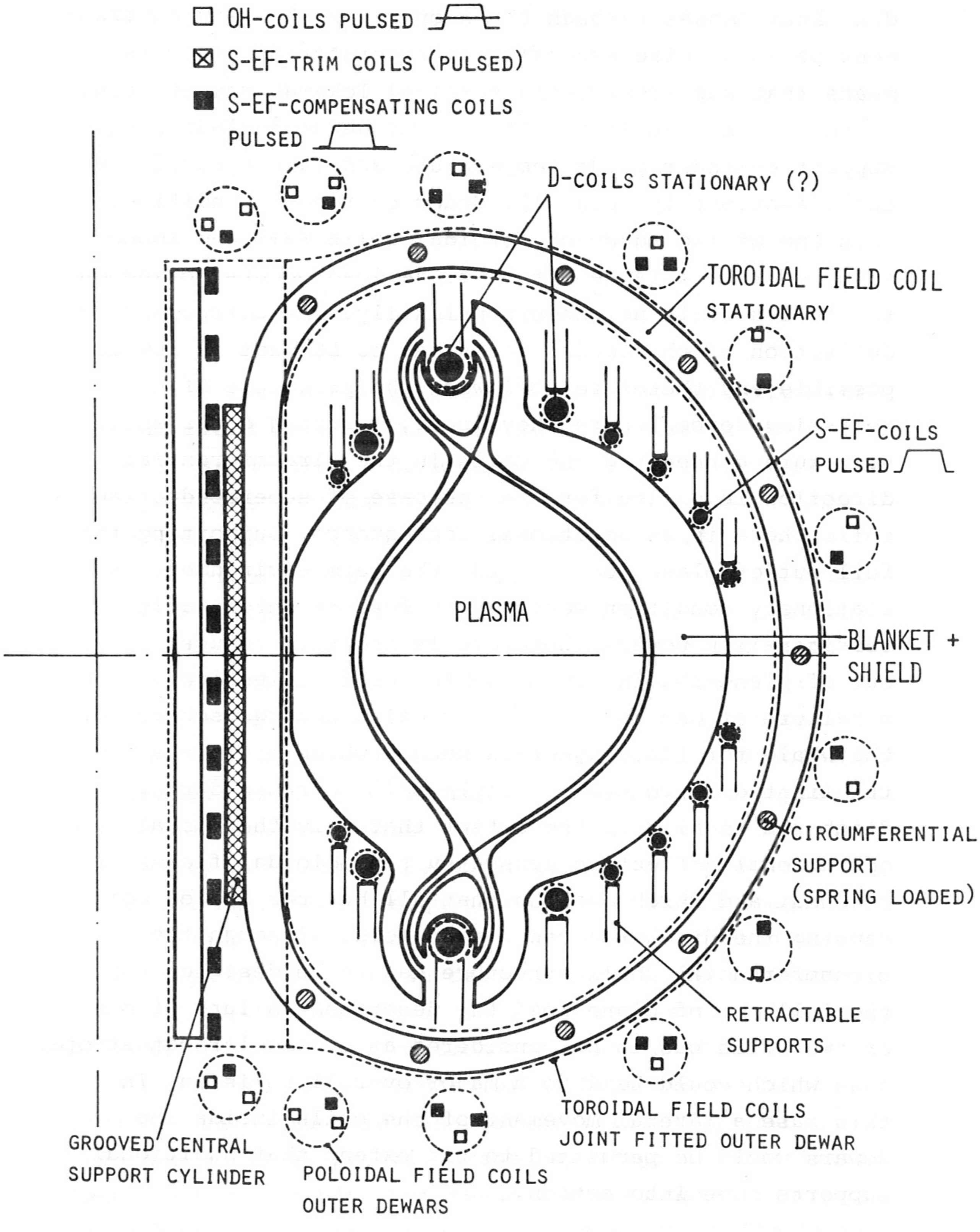


Fig. 1 E P R - M A G N E T S Y S T E M
WITH DOUBLE-0 POLOIDAL DIVERTOR

centering forces almost rules out the possibility of completely separate dewars for each coil. The correspon-ding heat losses through the support to the warm environ-ment would require excessive refrigerator power. This means that superconducting toroidal Tokamak magnets tend to require a so-called fitted joint dewar including the support cylinder at He-temperature and cannot easily be built sectionally (Fig. 1). Under emergency conditions when one or two adjacent toroidal coils fail and loose their current a large out of plane load occurs acting on the failed coils neighbours primarily. The corresponding deflection of the loaded coils has to be kept as low as possible and therefore following the principle of resisting forces where they occur, a strong outer shell structure connecting the coils in the circumferencial direction is called for. In the case of superconducting coils there is an additional consequence. Supporting the full out of plane load towards the warm environment in stationary condition would again lead to intolerable refrigeration power. Therefore in order to reduce the out of plane loading and also the coil deflection during a failure of one coil an electrical discharge scheme for the whole toroidal magnet is needed which minimizes the transient differences in single coil currents during discharge ideally to the extent that only the normal operational deflection caused by the poloidal fields is reached, and which assures that all but the failed coil causing the discharge can remain cold. Although the circumferencial shell structure has to be designed for the full out of plane load the described failure of one or two coils has to be considered as a singular catastrophic case which could lead to a major overall revision. In this case a lateral movement of the coils inside the dewars would be permitted to the extent that additional supports come into action.

Fig. 2 shows the full out of plane force distribution over the circumference of a toroidal coil under fault con-ditions (one coil without current in the ORNL EPR 1976)

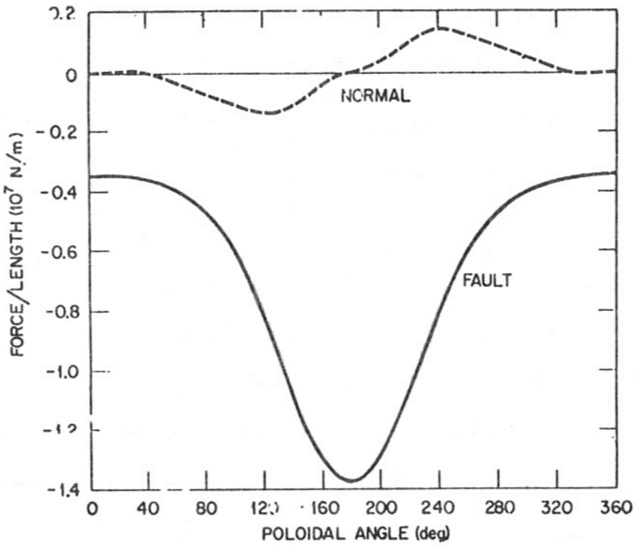


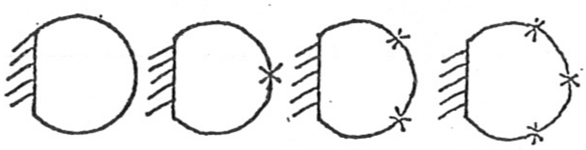
Fig. 2

compared to the normal pulsed out of plane load from the poloidal field system. The minimum stiffness of the coil and shell structure is determined by the limitation on the repetitive toroidal field magnet deflection. Taking the maximum permitted relative vertical field component of $3 \cdot 10^{-4}$ from the toroidal field magnet as defined for T20 the permitted lateral coil deflection even in a very large system is below 5 mm. In plane motions, however, in large toroidal coils are larger. Cooling contraction or charging expansion may lead to dimensional changes of the order of cm in an EPR size Tokamak magnet. Since there are elastic dimensional variations of such a size possible in a large toroidal magnet it is almost obvious that an oscillatory or instable response of such a system to mechanical excitation is quite possible. Recent magneto-elastic analysis has shown that for large magnet dimensions, high fields, and in case of insufficient circumferencial support there could be a circumferencial collapse or strong deformation following a small out of plane deflection of a single coil. The above mentioned outside shell structure can prevent this. Of course the shell has still to provide access openings for auxiliary plasma heating mainly, but these openings have to be restricted to a width not allowing excessive deflections. Table III shows a number of calculated results for an experiment and an EPR [21, 22, 23, 24].

TABLE III CRITICAL BUCKLING CURRENTS AND FIELDS

MAGNET DESIGN	1ST MODE	2ND MODE	3RD MODE	4TH MODE	DESIGN DATA
TFTR					
I_c (MA)	$\boxed{2,77}$	5,68	11,82	19,90	3,22
B_c (T)	$\boxed{3,96}$	8,12	16,89	28,50	4,6
ANL EPR 1975					
I_c (MA)	1,55	3,17	$\boxed{6,46}$	10,83	6,5
B_c (T)	0,816	1,67	$\boxed{3,41}$	5,71	3,4

SIDE VIEW OF COIL AND SUPPORTS



I_c --- CRITICAL BUCKLING CURRENT IN AMPERE-TURNS PER COIL

B_c --- CRITICAL MAGNETIC FIELD AT MAJOR RADIUS OF TORUS

A very important issue for toroidal magnet designs is the proper choice of operational current and voltage with respect to the safety discharge. Characteristic for many earlier superconducting Tokamak magnet designs was the relatively low conductor current compared to normal conducting magnets. This corresponded to the foreseen difficulties of manufacturing stable high current conductors in long lengths. More recent designs propose parallel operation of several conductors or single conductors for up to 60 kA since low currents at high stored energies could cause insuperable insulation problems during an emergency discharge of the torus magnet. As will be shown later, the safety discharge conditions couple many of the magnet design data including the permitted temperature rise in the failed winding, thus leaving only a limited parameter choice.

Since very likely Tokamaks will remain pulsed devices as they are now - even if confinement physics would become perfect, the limitation on flux swing for inducing the plasma current indicates this and the existence of a self--sustained boots trap current has not been demonstrated - under several aspects the toroidal Tokamak magnet is not a DC device. With a superconducting winding it has to be conceived of as operating in a difficult ac field environment. In normal conducting Tokamak magnets the problems from pulsed poloidal fields on the toroidal magnet like in the superconducting versions arise from the bending and tilting forces and additional heat produced; but for a superconducting magnet these forces and additional heat inputs may principally limit the performance of the magnet or in other words, conductor, winding and cooling principle of the superconducting toroidal magnet have to be taylored according to the pulsed external field pattern. There are reasons like state of the art and cost of ac superconductors as well as space limitations to use normal conducting poloidal field windings in combination with the first larger superconducting toroidal magnets. The spatial arrangement of these poloidal windings, however, in any case will have to be chosen such that a minimization of pulsed field effects in the toroidal coils will be achieved[5]. There have been proposals to use passive cryogenic or even superconducting shields[10] to protect the toroidal field coils or ease their conductor design. It appears that they either will cause additional overall heat production or in case of a superconducting shield shift the pulsed field interaction problem to a separate winding, which adds complication and can only compensate the pulsed fields for the toroidal coils for one single poloidal field pattern. This pattern is variable in time and must not be changed by toroidal field coil shields, which for efficient shielding would have to be designed for a minimum time constant like the burn cycle length. Insulating breaks in the shield would be required to allow the above mentioned safety discharge. Large local forces would be exerted on the shield especially in the vicinity of the central support cylinder. With or without a shield the stable operation of the toroidal magnet can only be assured if the pulsed field magnitude and rate of rise nowhere along the winding circumference locally can generate excessive losses and/or motion.

2.2 Poloidal field coils

The pulsed poloidal magnetic field system of a Tokamak (see Fig. 1) besides of the plasma ring current itself can consist of the ohmic heating (OH) windings, the equilibrium field (EF) windings, a shield (S) winding, a field shaping (F) winding and a divertor (D) winding[25]. It is assumed here that only poloidal divertors will offer the chance of application in a reactor. It has been shown recently that the interesting radial bundle divertor concept for the technical reasons of necessary very local high fields and forces and radiation shielding problems is not reactor relevant[26].

The OH-winding serves as the transformer primary to start the ring discharge and to induce the short turn plasma current, to keep constant its value over the burn period and to end the current flow.

Table IV is a list for existing designs showing the assumed volt-second requirements and their subdivision between OH and EF windings, the type of operating cycle together with the relevant times as well as the energies, voltages and powers involved. Also the flux density swing in the core is indicated. As can be seen from the list the start-up or recharge requirements for the OH coil in terms of $\dot{B}$ should become easier in large machines. This may have an important influence on the experimental step at which to introduce superconducting poloidal windings. Because of their required large flux density variation the OH coils have to be placed outside the toroidal field coils in such a spatial distribution, that the field of the OH windings in the plasma region is of the order of 10s of Gauss (Fig. 1). This means that most of the

TABLE IV   OH/EF WINDING DATA

	V (kV)	I (kA)	P_{MAX} (MVA)	E_{MAX} (MJ)	$\Delta\phi$ (Vs)	ΔB (T)	$B(\frac{T}{s})$	ΔT_{RISE} (s)	COND.
T 20	28/26	120/120	3450/3100	220/460	71,5/46,5	± 4	4/	2	Cu
ANL EPR 1976	51/21	80/80	1910/420	1200/1500	85+4/50	± 5	6,7/1	2	NbTi
ORNL EPR 1976	69/	50/	3500/180			± 7	7/	2	NbTi/Cu
UWMAK II		65/	$\overbrace{950}$	$\overbrace{10400}$		$+5,7$ $-8,4$	0,16/	10	NbTi
UWMAK III		10/5	$\overbrace{900}$	$\overbrace{8860}$		$+6,9$ $-8,6$	0,77/	20	NbTi

OH coils will form a straight long cylindrical coil inside the torus magnet with the remaining ones distributed outside along the torus magnet surface in such a distance as not to cause large local pulsed fields. The straight cylindrical OH magnet section is one of the most critical parts of the overall magnet system since it influences directly the machine design and the Tokamak scaling. In order to achieve economically interesting regimes all Tokamak designs are being squeezed towards a low plasma aspect ratio, which mainly involves the inside blanket and shield dimension and the flux density swing in the machine core to produce the required flux swing which in many designs contains a safety or uncertainty margin of 2. This then leads to as high as possible fields in the inner cylindrical OH coil. Because of its moderate radial dimensions, even considering fatigue during the machine life, tensile stress in this coil is not a major problem. Consequently one finds the highest B values in Tokamak designs in the self field of the central OH coil. These characteristics have led so far to almost an exclusion of iron cores in the very large Tokamak concepts and e.g. in JET to a highly saturated core. There are estimates for iron core versions of OH coils which seem to indicate that a complete iron core with return yokes will yield no major advantage, but heavily burden the design and construction with thousands of tons of magnetic plate[27]. One concern, however, remains about the air core concept, and that is the range of the poloidal magnet dipole field, which e.g. for the UWMAK II design is still about 30 G at a distance of 100 m from the reactor center line. This may call for some kind of flux shield close to the reactor. Another effect of the pulsed poloidal fields of course is the necessity of insulating breaks for about 1000 V in all metal structures like mechanical supports, blanket and shield structure and cryostats arranged around the torus axis. Despite these insulating breaks which only avoid direct OH transformer short turns detailled eddy current loss assessment especially in the cold support structures is important in order to minimize excess heating there. Fig. 3 shows some eddy current decay time constants at different temperatures[28]. Recent more detailed studies

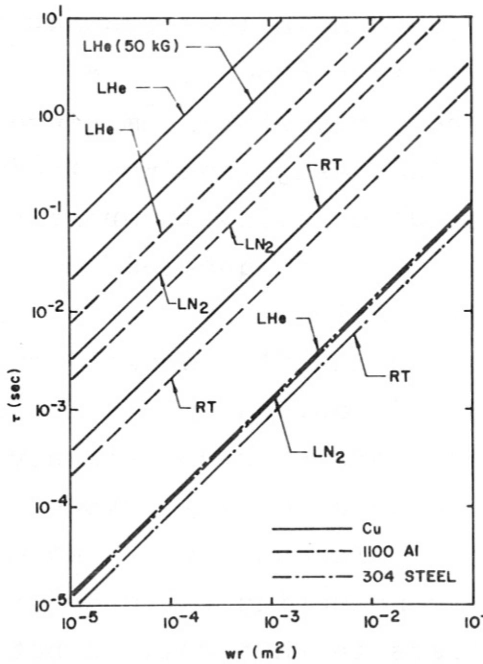


Fig. 3 Decay time constant for currents in a cylindrical shell of radius r and thickness w.

of EPR magnets and their support structure have revealed the eddy current problem in large Tokamak magnet structures as a major design constraint[29].

The equilibrium field winding which may be combined with a shield winding to be explained later has to provide an essentially plasma current proportional but adjustable vertical field with a spatial distribution according to the plasma current distribution which controls the plasma position according to the plasma current and radius, plasma current distribution and the poloidal plasma pressure. Table V lists the vertical field values required

TABLE V	$B_V(T)$	$I_P/R(10^4 \frac{A}{CM})$
JET	~0,47	1,62
T 20	0,42	1,2o
ANL EPR 1976	0,46	1,21
ORNL EPR 76	~0,34	1,o7
UWMAK II	~0,44	1,15
UWMAK III	~0,85	1,95

for some Tokamak designs together with the quantity $\frac{Ip}{R}$ mainly determining the vertical field. Since the vertical field should increase in large Tokamaks especially for noncircular plasma cross sections the requirements of $\dot{B}$ in the EF coils may not much easier to be met in future apparatus unlike to OH windings. Another difference between the OH windings and the EF windings concerns their maximum field impact on the toroidal field coils. Whereas with an outside OH winding the maximum local pulsed field in the region of the toroidal field winding can be restricted to about O.1 T [9], about 3 - 7 times higher fields have to be expected from a simple EF winding almost regardless whether this winding is placed outside or inside the toroidal field coils. For symmetry the equilibrium field windings have to be distributed above and below the plasma ring. Placing them outside the toroidal field coils would be desirable for easier fabrication and assembly. Great care has to be taken concerning their spatial distribution under the constraints of access for pumping ports, cooling ducts and auxiliary heating systems but also avoiding excess local field peaking in the toroidal field winding due to large EF ampere turn concentrations close to it. Fig. 4 shows this for one of the EPR designs[9]. Complete penetration of the vertical field generated by the EF coils across the lower and upper parts of the toroidal field coils (see Fig. 5)[13] will cause there undesirable heating

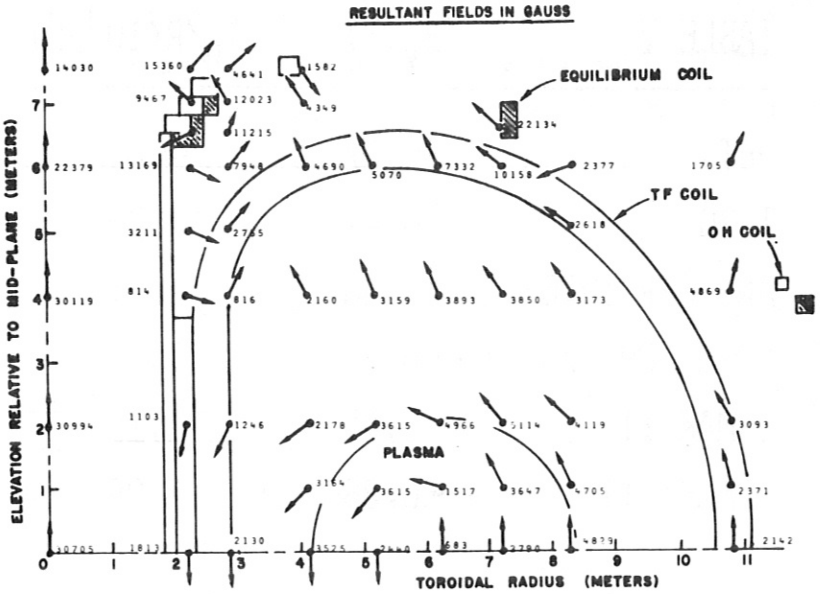


Fig. 4

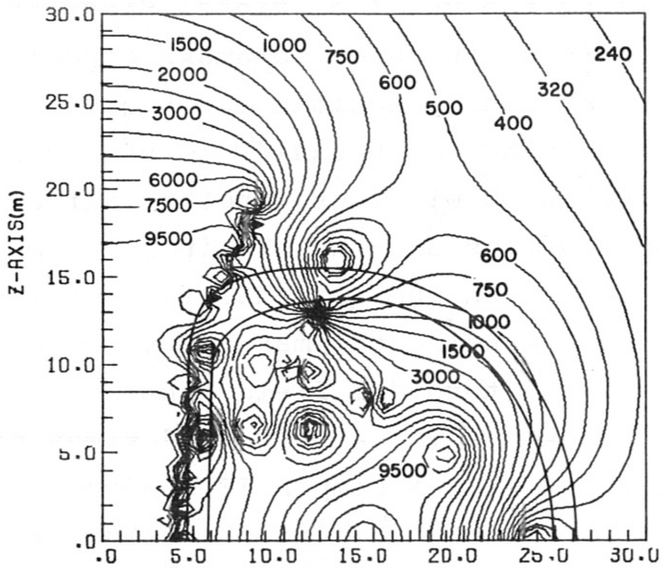


Fig. 5

and local twisting and overall bending moments. The latter can be rather large (see also Fig. 2). Table VI gives some corresponding figures for Tokamak designs. In UWMAK II the maximum repetitive lateral toroidal coil movement due to the combined forces from all the poloidal coils has been calculated to be about 1 cm which could mean about $7 \cdot 10^{-4}$ relative vertical field component from the toroidal coils. This is twice as much as has been permitted in the T20 design. In future Tokamaks of the flux conserved type[30] or with a non-circular cross section, the required vertical fields will go to the upper limit just mentioned. This would mean that the corresponding ac losses in the toroidal field magnet and the pulsed forces would become then rather

TABLE VI

MAXIMUM OUT OF PLANE FORCE / LENGTH ON TOROIDAL COIL FROM POLOIDAL FIELDS

ANL EPR 1976	$1{,}0 \times 10^7$ N/M
ORNL EPR '76	$1{,}4 \times 10^6$ N/M
UWMAK II	$1{,}2 \times 10^7$ N/M

large. The ORNL EPR indicates therefore the possibility to place the EF winding inside the toroidal magnet which of course has its great draw-backs like winding them in place and the necessity of vertically movable supports to enable maintenance and repair in the plasma chamber, blanket and shield region (Fig. 1). The special feature of the ORNL EF winding [5] is that it serves at the same time as a shielding winding for the plasma currents poloidal field, thus reducing the maximum vertical field component in the toroidal field windings. The combined so-called shield-equilibrium winding has a current distribution consisting of a first shielding part corresponding to the shielding currents in an infinitely conducting shell around the plasma and a second part of the EF winding with a zero poloidal net current. In order to decouple it from the OH winding the S-EF winding is series connected with a de-coupling winding placed close to and with a similar current distribution as the OH winding. This additional winding with a sum current equal to the plasma current causes some space problems in the region of the straight inner OH coil section. The S-EF winding thus defined will only correctly shield the toroidal field coils for one definite plasma current distribution and one single value of the poloidal plasma pressure. This is the reason for the reduction of the vertical field by only some factor (about 3 - 5) whose exact value depends on the range of β_{pol}

and plasma current profile to be corrected for. The correcting part comes from an additional EF trim winding outside the toroidal coils. Fig. 6 shows a comparison between the poloidal flux line pattern of an outside EF and an inside S+EF winding scheme[11]. Although proposed for

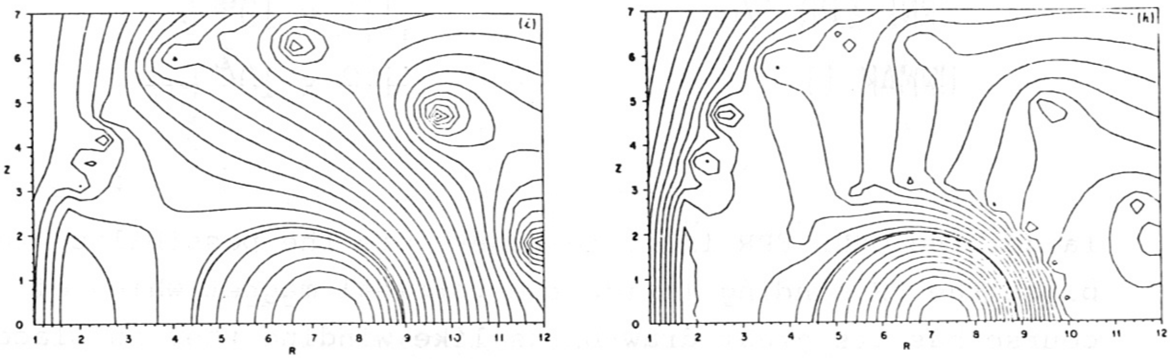


Fig. 6 Poloidal flux line pattern

with EF winding with S + EF winding

the ORNL-EPR with normal conducting inner S+EF windings, this scheme applying inner poloidal superconducting windings seems to be inescapable for a pulsed full scale reactor, especially should a non-circular plasma cross section turn out feasible. In this case of a preferably vertically elongated plasma cross section additionally a field shaping winding, roughly a quadrupole winding with a zero poloidal net current is required. Since the quadrupole field components have a rather short range and the required fields are relatively high, this F winding can only be situated inside the toroidal magnet. The F winding is needed to control the plasma cross section independent of the plasma current distribution. For a certain current distribution it can be combined with the EF or S+EF winding. It is assumed that this does not considerably increase the pulsed fields at the toroidal magnets, but the spatial distribution of the pulsed fields will be different and may lead to additional local twisting moments on the toroidal field coils.

In addition to this machine concepts of the flux conserved Tokamak type will require the possibility that single conductors or single coils of the EF or F+EF winding can be controlled independently such as to conserve the poloidal flux like in an infinitely conducting shell surrounding the plasma.

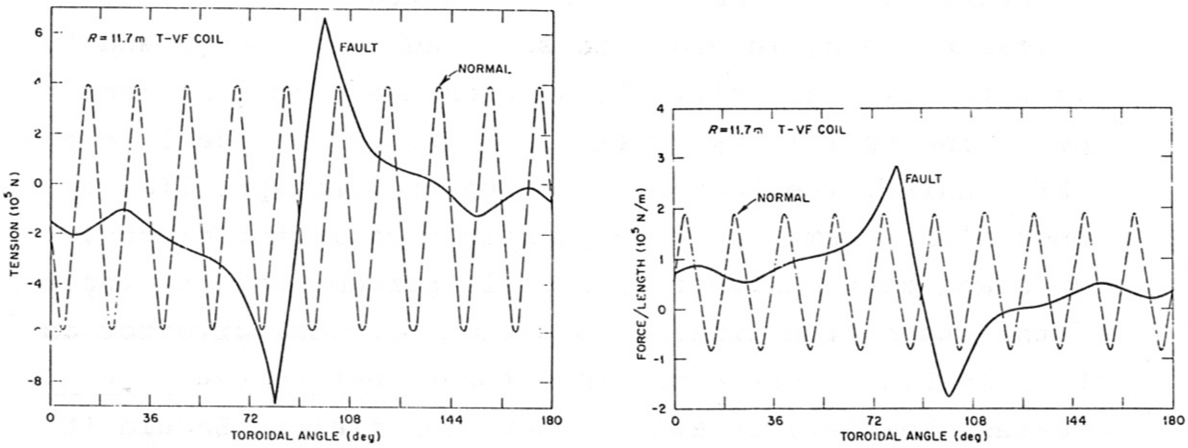
Since the blanket and shield dimension will separate the plasma surface from the location of the EF or F+EF winding, there will be in the case of the flux conserved Tokamak the need for a vertical field trim winding to center the plasma ring and a quadrupole trim winding to enforce the required plasma cross section despite of the distance between plasma surface and EF or F+EF winding.

As already mentioned a poloidal D winding seems to be compatible with an EPR or reactor situation. For topological reasons, in order to save magnetic energy and to limit the external field loading on the toroidal magnet the D winding will have also to be placed inside like the S+EF winding. The topological aspects more specifically concern the geometry of the particle collector plates, the shape of the diverted particle streams and the location of the stagnation line. A localized poloidal divertor as shown in Fig. 1 has a winding of zero net current. It creates, however, locally rather high fields. Should it turn out necessary to substantially vary the divertor winding current in time, then the outer two D windings could be distributed to shield the toroidal magnets against the divertor fields. Concerning its radial position such a distributed divertor winding could be integrated in the S+EF winding.

For the divertor windings as well as for the other poloidal coils the local magnetic fields at the windings and the field variations are rather large, typically about half the values occurring at the inner OH-winding.

2.3 Mechanical structure, safety

All poloidal windings arranged inside the toroidal magnet are subject to an additional constraint when compared to the outside ones: they have to operate in a high parallel field approaching the maximum toroidal flux density at the inner circumference of the torus magnet. Besides of the need to design these windings, if superconducting, especially for these high parallel fields, there is a mechanical influence on them from the toroidal magnet. With a necessarily limited number of toroidal field coils there is a large spatial modulation of the toroidal field. Since the large radius inner poloidal windings are rather close to the toroidal coils they will see considerable radial field components which produce in plane and out of plane forces on the poloidal field coils (Fig. 7)[11]. This



in plane   out of plane

Fig. 7 Forces from the toroidal field on one poloidal field coil

calls for a rather stiff mechanical structure especially for the support of inside poloidal coils which otherwise can experience a helical instability as recent magneto-elastic calculations have shown (Fig. 8)[24]. In the case of a failure in the toroidal magnet system like loss of current in a coil the interaction forces on the poloidal coil system would become even larger. Since, however, it is very unlikely that under such circumstances the plasma ring could remain stable any further but would

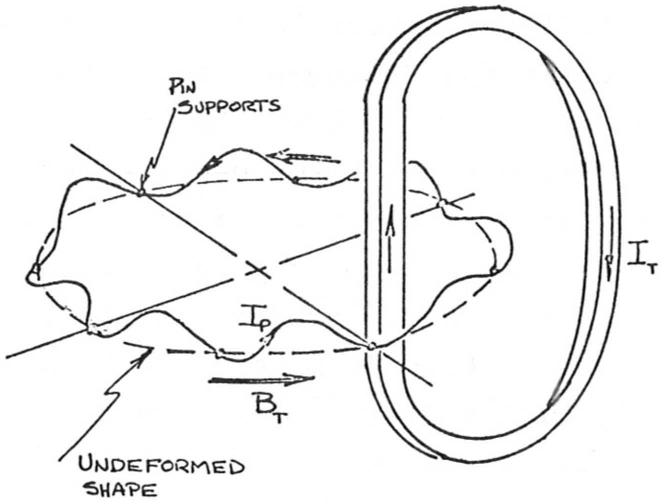


Fig. 8 Helical "instability" of poloidal coil

rapidly disintegrate and possibly damage the first wall, it is inferred here that in any case of a major failure in the integral magnet system calling for a discharge of the toroidal magnet, the poloidal magnet system would have to be discharged first. This implies further consequences on the ability of the toroidal coils to stand faulty condition for some time, that is to be built with a high degree of electrical stability and to have an early fault detection scheme.

The possible failure modes in the poloidal field coil system are manifold in principle since they involve the plasma behaviour itself. Up to now the experience tells that a disruption may cause a plasma current decay several times faster than the initial current rise. In this situation the S+EF winding as described above may help in preventing a disruption or to slow it down. Emergency shielding to slow down the local field variations in order to protect the superconducting inner poloidal windings has not been considered so far. As has been calculated for JET, the sudden loss of plasma current with the other poloidal currents unchanged will lead to approximately doubling of the tilting moments on the toroidal coils. Such an effect can be smoothed out by an inner S+EF winding.

To cope with failures occurring in the poloidal coil system means very roughly the following: First the plasma current must be driven to zero. This can be done either by the OH or the inner S+EF winding, provided the latter has been decoupled from its compensation winding. Both can provide roughly the volt seconds for a plasma current decay. After that a safety discharge to protect the failing superconducting poloidal winding from being overheated can follow. The relatively high voltages necessary in the poloidal field system (see Table IV), however, increase the likelyhood of a dielectric failure there. Such failures of course can lead very rapdily to asymmetries in the poloidal field pattern implying again the danger of the plasma hitting the wall. Therefore a series connection of poloidal field windings symmetric about the machine mid-plane and redundant parallel coils to preserve the S+EF field pattern may be required. The definition of the whole poloidal magnet system needs a lot more experience with windings closely coupled to the plasma especially for non--circular plasma cross sections. It may be appropriate to build the first TNS or even EPR size machines relying on inside normal conducting poloidal coils like proposed in the ORNL and GA designs[31, 32, 33, 34].

Designing the inner poloidal windings and laying out their manufacturing process has a prerequisite the detailed logistics of the whole machine assembly-procedure. Maintenance and repair operations for the plasma chamber, blanket and shield have to be specified before inner poloidal coil design and they cannot cover inner superconducting poloidal windings in an easy way, since so far no obvious solution for a sectional superconducting poloidal field coil[35] design is known. Reflecting the problem of dielectric failures in the toroidal or poloidal magnets and remembering again the possible catastrophic effect of a rapid field distortion one must state that the electrical insulation in the toroidal and poloidal magnets must not involve the slightest likely risk. This may have the consequence that only high current hollow conductors or more generally such winding concepts can be permitted which decouple superconductor cooling from electrical insulation.

The toroidal and poloidal magnets both determine together the basic mechanical structure of a Tokamak. The cryogenic envelope of the toroidal magnet will partially include also the OH-coil following the favourable tendency of low aspect ratio. The toroidal coils will be essentially hanging from and pressing against the grooved vertical cold support cylinder and need an additional support from below against the dewar wall. Depending on the stiffness of the coil together with its helium dewar another support from above may be required. These additional supports have to take into account in their design the dimensional changes of the toroidal coils depending on temperature and current which should remain symmetric about the reactor mid plane. The main lateral support of the toroidal coils themselves will be most likely against their dewar walls via sliding thermally insulating struts designed for carrying the transient out of plane forces during safety discharge or the poloidal field interaction forces which ever are larger[13, 14]. Retractable vertical support pillars from above and below for the inner poloidal field windings seem to be the most compatible solution for horizontal access requirements of maintenance and repair operations in the vacuum chamber, blanket and shield region. In order to stiffen the poloidal coil system it will be necessary to mechanically connect the retractable supports in the radial direction. The necessary supports to resist gravitational forces will be separate for the toroidal and poloidal coils. The central column besides of carrying essentially the weight of the toroidal coils can serve as a mechanical restraint for the outer poloidal coil system in separate or common annular cryostats.

The overall picture would not be complete without mentioning that a high reliability of the toroidal magnet system is called for because it can hardly be dismantled and repaired during the lifetime of the machine without causing excessive downtime. To dismantle any of the poloidal coils except for the upper few seems almost equally impractical. The safe operation of a complete Tokamak magnet will have to be ensured also by redundancy in the cooling capacity and by an efficient sensing system to monitor the regularity of the electrical and mechanical behaviour. The repetitively pulsed operation can produce cumulative effects such as fatigue or excessive motion. An automatic fast failure evaluation and emergency discharge system seems inescapable.

So far a rough description of the requirements for future Tokamak magnet systems. This description combines a large number of different features to be found in existing designs. By putting them into a suitable combination, one may gradually evolve a viable Tokamak EPR or reactor magnet configuration. The above is an early attempt to crystallize it.

3. SOME DESIGN CONSIDERATIONS

3.1 General

Today no realistic detailed EPR design is possible yet, but the EPR design studies, many of them from the United States but also from Europe and Japan, have brought forward interesting proposals for future technical solutions. These proposals together with some existing detailed work on fusion relevant superconductors and background experience from building much smaller than EPR-magnets still form the narrow basis for the design of large superconducting Tokamak magnets. Reference has been made and will be further made to the existing designs. Fig. 9 - 11 show JET, TNS, T20 for comparison at equal scale as the ANL, GA, JXER and ORNL EPRs[11, 36] and at a scale further reduced by a factor of 2 FINTOR EPR, UWMAK II and IIItogether with Culham MKII reactor design to

Fig. 9 TOKAMAK MAGNET SYSTEMS AND DESIGNS FROM JET TO EPR

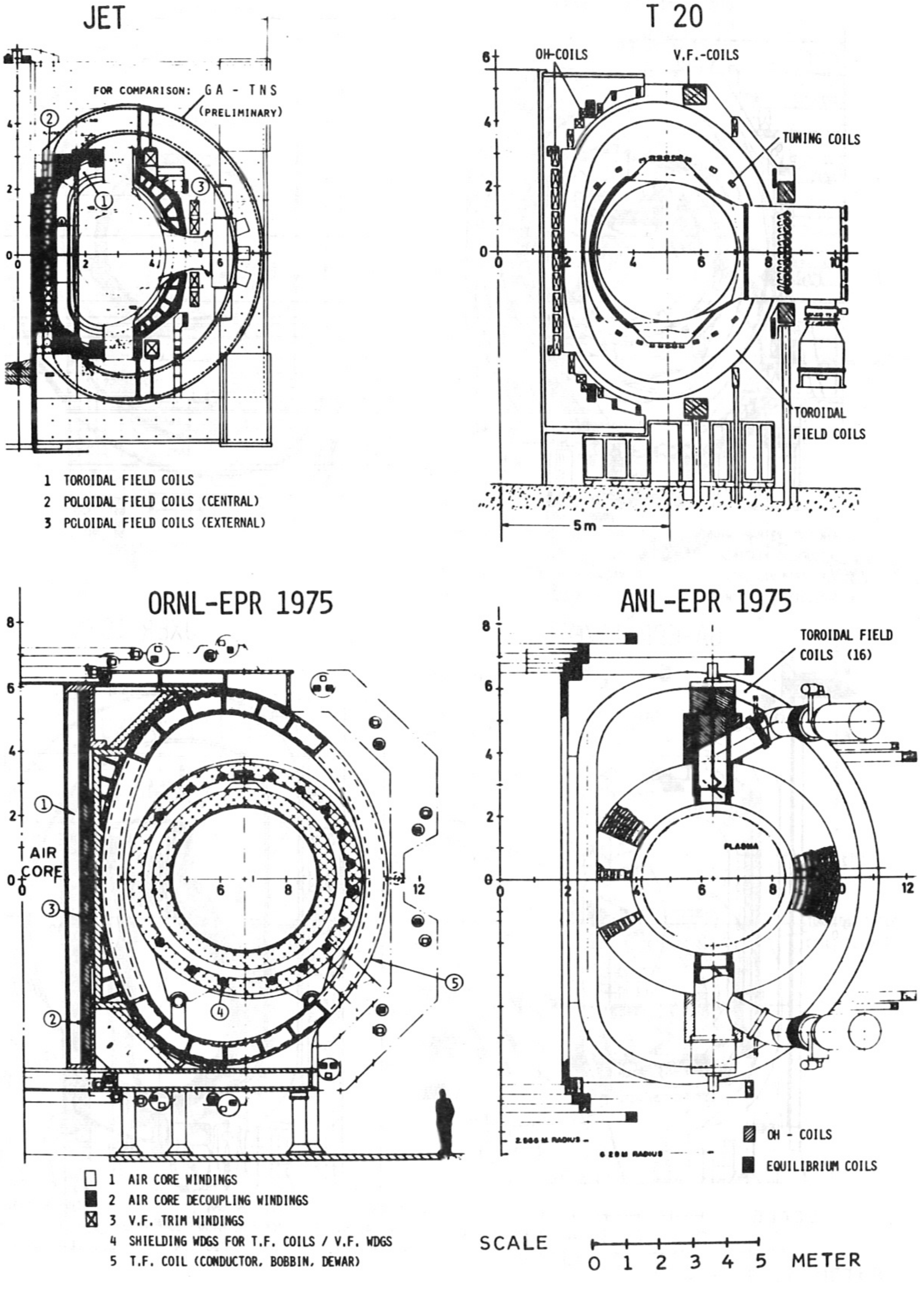
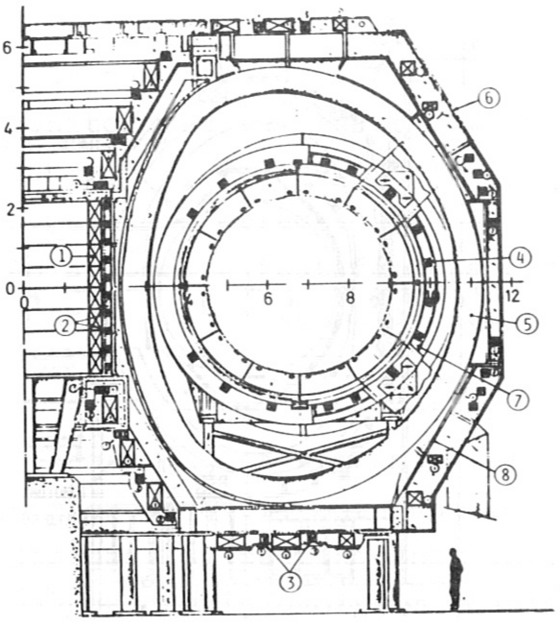


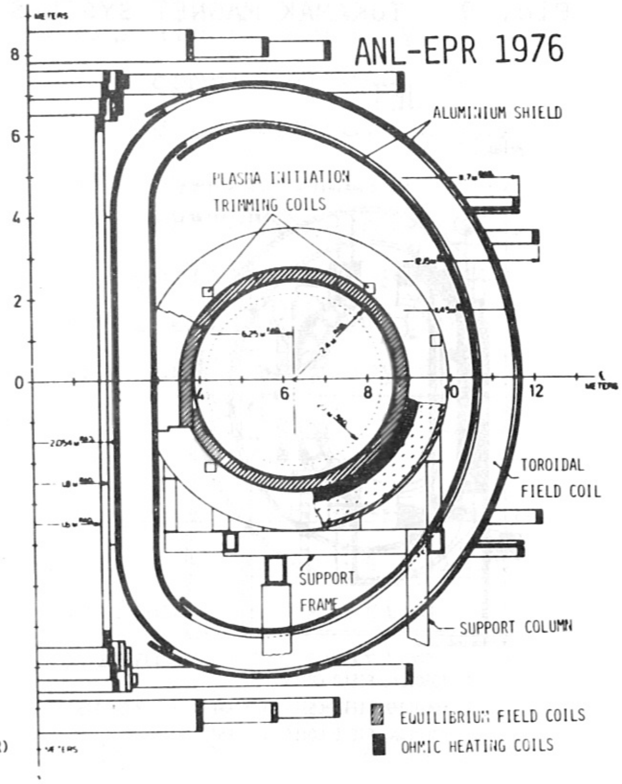
Fig. 10 TOKAMAK MAGNET DESIGNS

ORNL-EPR 1976

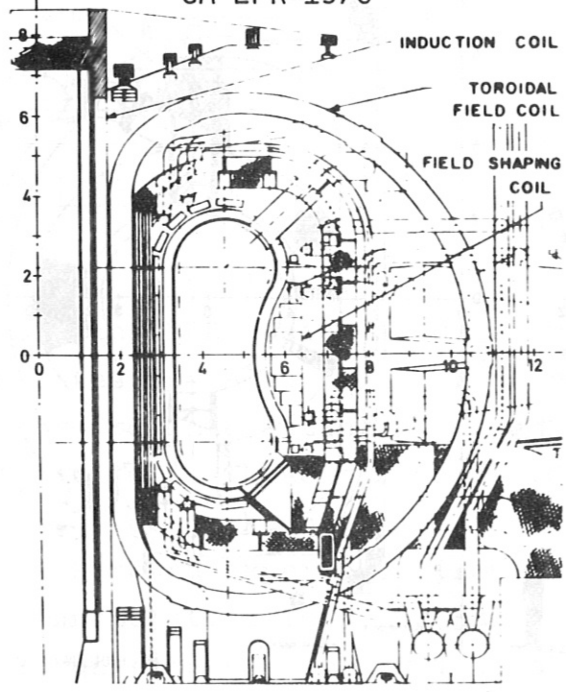


1 AIR CORE PRIMARY WINDING
2 DECOUPLING WINDINGS
3 V.F. TRIM WINDINGS
4 SHIELDING / V.F. WINDINGS
5 T.F. COIL (CONDUCTOR,BOBBIN,DEWAR)
6 POLOIDAL COIL SYSTEM SUPPORT
7 VACUUM PORTS & SHIELDING WDGS. SUPP.
8 SHIELD & BLANKET SUPPORT

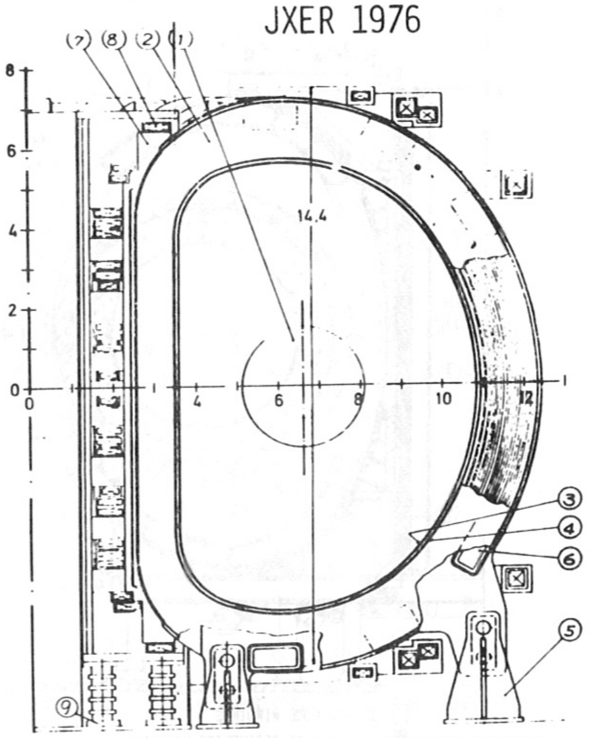
ANL-EPR 1976



GA-EPR 1976



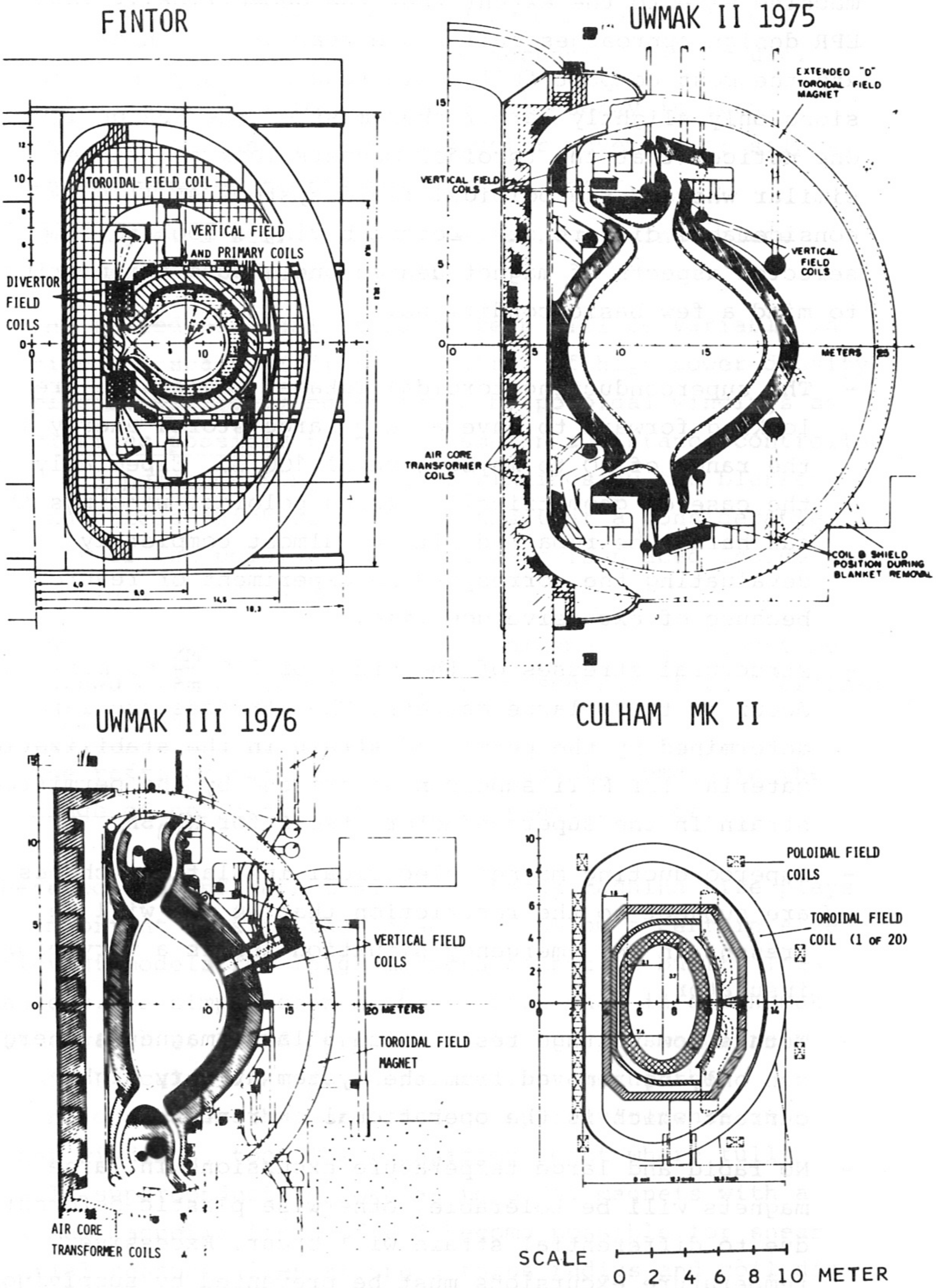
JXER 1976



SCALE 0 1 2 3 4 5METER

1 PLASMA
2 TOROIDAL MAGNET
3 VACUUM VESSEL
4 LIQ - HE CAN
5 TOROIDAL MAGNET SUPPORT LEG
6 ANTI - TORQUE BEAM
7 SUPPORT STRUCTURE
8 POLOIDAL MAGNET
9 CENTER POST SUPPORT LEG

Fig. 11 TOKAMAK MAGNET DESIGNS FROM EPR TO FULL SCALE REACTOR



illustrate their size and magnet configuration. Quite characteristic for the present situation there is still a large variety in proposed reactor data versus physical machine size to the extent that the geometrically largest EPR design approaches full scale reactor design dimensions and the most compact full scale reactor design has dimensions only slightly larger than most of the EPR designs. One notices that the toroidal magnets look all rather similar whereas the poloidal field systems still show considerable differences. Before having a look to some selected aspects of magnet design one should again bring to mind a few basic conditions:

- The superconducting toroidal Tokamak magnets we are looking forward to have a very large stored energy in the range of 10 to maybe several 100 GJ. Especially in the case of geometrically linked poloidal windings they can hardly be repaired without almost completely devaluating the corresponding experiment or reactor because of excessive downtime.
- Structural stresses of the order of 200 $\frac{MN}{m^2}$ or more will occur in these large magnets. The stresses are largely determined by the permitted strain in the stabilization material for NbTi superconductors and by the permitted strain in the superconductor itself for Nb_3Sn.
- Superconducting magnet electrical insulation schemes are subject to the restriction that He gas will be present in any emergency situation and is a very poor insulator.
- With a low voltage restriction a large magnet's energy can only be removed from the system at very high current which is the operational current.
- No rapid and large temperature excursions in large magnets will be tolerable, otherwise plastic deformation due to differential strain will occur. Excessive temperature excursions must be prevented by supplying enough stabilizer, sufficient cooling and proper energy removal in an emergency case.

- Superconductors fulfilling the above requirements will have substantial dimensions like heavy electrical machinery normal conductors.
- The maximum out of plane load by one toroidal coil failed has to be taken by the mechanical structure.
- The elastically stored mechanical energy in large magnets is considerable. Conductor movement or slippage in the winding therefore must not occur.
- The magnet's stored energy per unit winding volume increases with magnet size.
- The pulsed poloidal field pattern will be variable in time and space. Efficient control of high power density plasmas will only be achieved by poloidal windings as close as possible to the plasma in a distance controlled for superconducting poloidal windings by the blanket and shield thickness. This means that a considerable part of the poloidal field magnets is placed inside the toroidal field magnet.
- In a working large Tokamak the plasma current must be turned off before an emergency discharge of the toroidal field magnet.
- The refrigerator electric power should remain in the range of percents of the rated reactor power.

These points indicate that the Tokamak machine size plays an important role in the sense that below a certain size relevant models for large magnets cannot be built. This can e.g. be simply seen from the fact that in a Tokamak for a given a, A and toroidal field the plasma current density scales like $\frac{1}{R}$. Therefore increasing the dimensions eases the current density requirements in other windings, too. The transition point where fully stable superconducting Tokamak toroidal magnets with a maximum magnetic field of 8 T become possible for shear spatial reasons, lies around a torus radius and coil diameter of 3 m. Since the forces increase with magnet size, the size dependant relationship between current density and stress is important. Smaller magnets have to be built partially stabilized; they will quench rapidly at any surplus heat input, protection will scarcely be possible without quenching all the rest of the coils, too, and without excessive refrigeration such an event causes long interruption times.

One should just state that JET size is about the smallest stable superconducting Tokamak magnet coil one should consider , which in turn would leave a further range for scaling up in linear dimensions by a factor of 2 for the TNS or EPR and after that maybe by another factor between 1 and 2 for the full-scale reactor magnet.

3.2 Toroidal coil shape, strain

Regardless of the size, the appropriate shape for large toroidal coils is at first glance the constant tension D-shape (modified D) which can be described according to Fig. 12 in an approximate analytical expression taking into account a finite coil number[38, 39]:

$$\frac{\rho}{R} = \left[\frac{\rho_2}{R_2}\left(1+\frac{1}{N}\right) - \frac{1}{N}\ln\frac{R}{R_2}\right] \Big/ \left(1+\frac{1}{N}\right)\cos\phi$$

As has been shown this analytical expression agrees quite well with an iterative solution developed elsewhere[40] (at least for $N > 8$). There are problems, however, with this shape; mainly because the width of the coil determines its height depending on the inside torus radius and therefore may yield a magnet too high and storing too much energy for the purpose.This shows up in the T20 coil design or the ORNL-EPR coil (see Figs. 9, 10, 11) where the inner straight section has been replaced by a curved one. The price to be paid for this is an increase of tension in the bent inner section and a consequent shear zone at the transition of the bent inner section to the outside D-shape. In the UWMAK II and III coil designs the

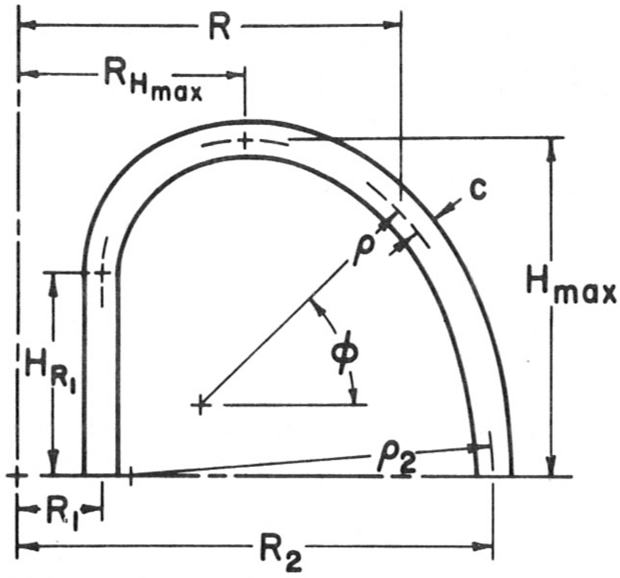


Fig. 12

straight inner section is retained but it is connected to a D-coil which is too broad in UWMAK II or too narrow in UWMAK III when compared to the practical D-shape. This leads to additional or less tension in the straight section and bending moments in the curved part of the coil. Quite naturally such deviations from the constant tension case, where of course even in the modified D-shape the tensile stress is never really constant but shows deviations of $\pm$ 10 to $\pm$ 20 % across the coil cross section, requires a special fixation for almost any turn of the winding in order not to sum up the shear forces but to transfer them directly into the force restraining structure. This can e.g. be solid grooved coil disks like in UWMAK reactor designs or a honey comb structure as proposed in the ORNL-EPR[11]. In any case the wall thickness and conductivity of this force restraining structure has to be adjusted to the magnetic field variation of the poloidal fields for limitation of eddy currents. This may rule out high strength aluminium structures for EPR size machines with their assumed rapid field variations.

An interesting coil shape resulting from using the modified D-shape for every winding layer is presented in the FINTOR I design[27]. Apart from the above mentioned coupling of the shape to the torus radius with such a coil shape additional magnetic energy is stored in the extended upper and lower coil cross sections.

Since a variation of the conductor and load bearing winding cross section along the circumference of the coil is not feasible because there is no conductor joining method producing a perfect superconductor transition, because such a variation would be exceedingly expensive and also since shear stress in a winding should be small in order to exclude the possibility of conductor slippage, a coil shape close to the D-shape in principle makes the best use of the material spent. The amount of material in the conductor is given first by the stabilizer requirements. Depending on field and coil size additional restraining material such as stainless steel or high strength aluminium has to be added. Within the given strain limits there is a certain limited choice of coil shapes deviating from the practical D-shape as mentioned above. The locally permitted strain limit, which sets essentially the mechanical design, is given for the very ductile NbTi superconductor only by the properties of the stabilizer material (for copper 0.1 - 0.15 %), for Nb_3Sn the superconductor itself very likely will determine the upper strain limit, which is a more risky situation. Since it appears that the permitted strain on Nb_3Sn is very low, it may turn out that it will be hard to achieve more than 12 T maximum field in large magnets.

Copper as a stabilizer material shows a strong magneto-resistance effect. Therefore in order to minimize the material in the coil stabilizer grading in the conductor has been proposed[42], e.g. by using conductors with 3 different fractions of copper stabilizer for the higher, medium and low field regions at the inner part of the coil. Because of the strong $\frac{1}{R}$ dependance of the toroidal flux density this means, however, that with copper stabilizer grading but necessarily constant conductor cross section along the coil circumference still too much stabilizer material is being spent. It also means that in terms of magnet stability the inner coil section is the most critical one. In the case of an emergency discharge this section would be heated preferably, thus causing differential strain. An equal fraction of stabilizer in the whole winding could be envisaged with aluminium whose magnetoresistive effect saturates at low field[43]. Also its lower resistance at 4 K compared to copper may reduce the amount of material needed. Since high resistance ratio aluminium is very soft, its application depends on more information about its repetitive resistance-strain behaviour. In any case it needs e.g. stainless steel restraint, possibly directly bonded to it in order to enforce an apparent elastic behaviour possibly at strains which permit full usage of the stainless steel like proposed in the UWMAK III design[44].

3.3 Safety discharge

Protection of fully stabilized magnets by a safety discharge is governed by two simple equations[45, 6]: The sum voltage over all the coils connected in n parallel branches is per branch

$$V_{branch} = \frac{\sum V}{n} = j^2 \frac{W_{mt}}{n \cdot J \cdot f(\Theta)}$$

W_{mt} total stored energy

$f(\Theta)$ protection function for given temperature rise

and the relationship between stabilizer current density j and conductor current J is

$$J = \left(\frac{K\dot{q}}{\rho}\right)^2 \cdot \frac{1}{j^3}$$

K geometry factor

stabilizer resistivity

$\dot{q}$ heat flow to the coolant

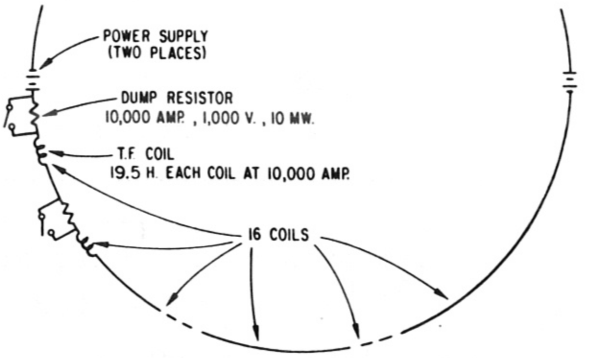
The corresponding time constant of the magnet discharge is

$$\tau = \frac{2 W_{mt}}{J \cdot \sum V}$$

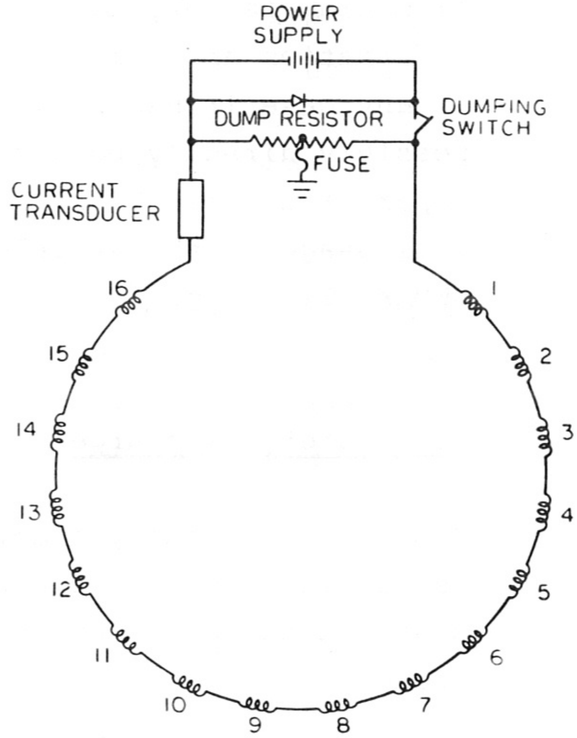
Connecting the coils and/or conductors in parallel branches can help to reduce the maximum voltage level at safety discharge at the expense of multiplying the number of separate power supplies.

A safety discharge will be started after a voltage sensing system monitoring the sum of resistive and conductor internal inductive voltage has detected a persistent or increasing normal conducting zone, which in a fully stabilized magnet can occur e.g. by local loss of coolant

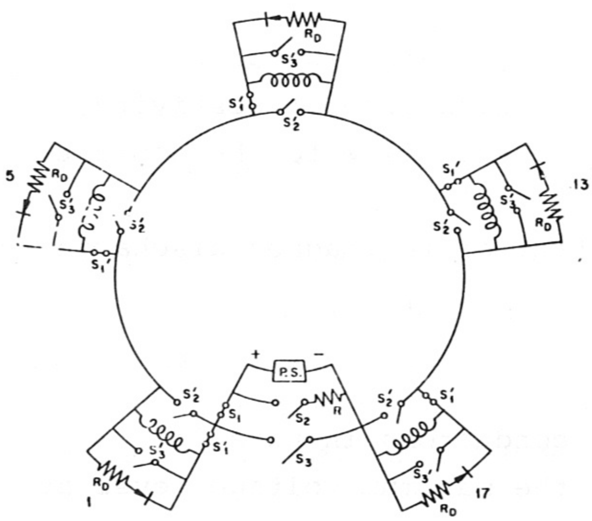
Fig. 13

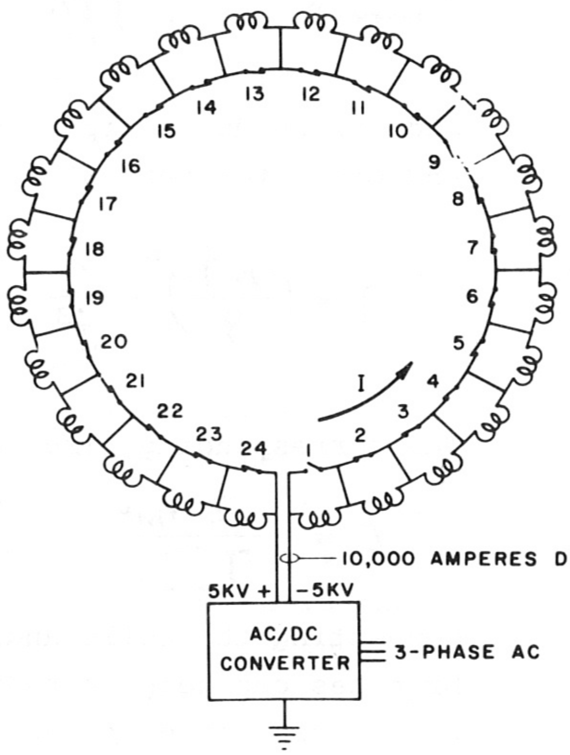


TF Coil Quench Protection System



1. FOUR SYMMETRIC GROUPS
2. NORMAL OPERATION:
   S_1 AND S_1' CLOSED; S_2, S_2', S_3 AND S_3' OPEN
3. DISCHARGE:
   (A) CLOSE S_2' AND S_3', THEN OPEN S_1' FOR THE QUENCHED COIL
   (B) OPEN S_3' TO DISCHARGE THE QUENCHED COIL THROUGH ITS R_D
   (C) THE POWER SUPPLY CAN BE ISOLATED THROUGH S_1, S_2, AND S_3, IF DESIRED





due to channel blocking, excessive energy input to a conductor by movement or a rapid field change causing excessive losses beyond the design limit. Fig. 13[9, 10, 11, 14] gives a couple of discharge schemes suggested for the toroidal magnets of above mentioned designs. Table VII lists the relevant data for the safety discharge according to these schemes. None of the proposals presents a transient analysis so far. Fig. 14 taken from the JET design[46] shows, how the maximum out of plane load in discharging the faulted and the other coils together can be diminished using the mutual coupling of the toroidal coils. J_2 is the current in the faulted single coil suffering a low resistance short at the terminals, J_1 is the current in the other series connected coils shorted by a switch with a time lag ΔT.

The safety discharge of the poloidal coil system has yet to be studied in detail.

TABLE VII

	I_{COIL} (KA)	$V_{MAX_{COIL}}$ (V)	τ (s)	PARALLEL BRANCHES	LEAD PAIRS	COILS DISCHARGED
ANL EPR 1975	10	1000	195	1	16	ALL
ANL EPR 1976	60	2000	506	1	1	ALL
ORNL EPR 76	78	1300	28	4	4	ALL
	78	300	10	4	4	5 OF 20
UWMAK III	10	10000	32	1	24	1 OF 24
	10	417	2280	1	1	ALL

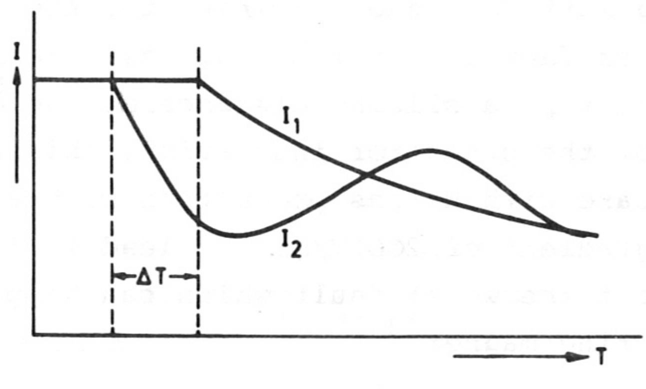


Fig. 14

3.4 Cooling

The field of cooling for a fully stable magnet is still under broad discussion and even the definition of full cryostability still has to be settled. In detail it requires to take into account any possible superposition of steady state and transient heat loads, the transient temperature distribution and the transient cooling properties in the different spatial directions relative to the conductor. A thorough general treatment of these items does not yet exist. There appear 2 main possibilities, each of which having two alternatives: firstly the immersion of the magnet into a bath of liquid or superfluid helium is possible. Liquid helium at 4.2 K may provide a heat flux of 0.3 W/cm^2 of cooled conductor surface for vertical surfaces and a temperature difference between conductor surface and helium of about 0.5 K, but this in a large magnet would appear to be a rather uncertain figure, since for a large surface to be exposed any surface orientation can occur in a coil, and other than vertical orientation may decrease the possible heat flux by at least an order of magnitude. The consequence is to use only the vertical surfaces for cooling and this preferably leads to pancake toroidal coils like in the UWMAK designs. Immersion in superfluid helium at 1.8 K would drastically improve superconductor performance but at the expense of a high relative room temperature refrigerator power and at the risk of large sealing problems. All solutions where the winding is simply immersed in the coolant may cause voltage problems since the cooling surface cannot be insulated and thus the conductor may in fact be insulated against the restraining structure just by a sliding distance of the order of the thickness of the conductor insulation. This in an emergency case with He gas production with a corona inception gradient of 200 V/mm may lead to internal arcing, about the worst fault which can happen to a superconducting magnet[43].

The second possibility of forced cooling in hollow conductors or ducts between conductor bundles allows to increase the cooling heat flow by increasing the cooled surface and a consequent reduction of the stabilizer material. There is, however, only a short maximum length of conductor which may go normal after a sudden heat input and still will recover after that. Also there is of course a restriction in the pressure drop from channel inlet to outlet, which determines the pumping power needed. Since pumping must be done at He-temperature, the pumping power adds to the refrigeration power. There are two possibilities of forced flow cooling. One is two-phase helium at a pressure drop of about 0.3 bar with 4.5 K inlet temperature and 4.2 K outlet temperature. One can reach 0.4 W/cm^2 at 0.5 K temperature difference between conductor surface and helium in cooling channels $\sim$ 150 cm long[47]. This is about the same cooling rate per unit surface cooled as for 4 K helium bath cooling but now the orientation of the cooled surface is rather irrelevant and the conductor can be fully insulated. Another advantage of the scheme is that in case of a local sudden energy input almost, like in the case of bath cooling - but somewhat reduced according to the two phase situation - latent heat of liquid helium will be available. In the case of two phase forced He cooling a typical length of conductor which can recover from a normal transition is about 5 m. Such a figure is favourable in that it sets the lower limit of the normal zone detection voltage to about 0.3 V. There are, however, questions about the applicability of this scheme because of the two phase situation. The second possibility of forced cooling applies high pressure supercritical helium with a minimum outlet pressure of about 3 bar[11]. Supercritical helium applies naturally in the case where an increase of the heat transfer is possible via an increase of the helium wetted perimeter per unit conductor length, e.g. when a finally subdivided conductor is needed in order to lower the eddy current losses. In the case of such a bundle[48, 49] conductor the pressure drop along a conductor length of about 400 m may be 5 bar, thus calling for a large pumping power[11]. The maximum length which can recover from a sudden heat pulse is of the order of only one meter, thus reducing the minimum voltage to be detected to ∿ 70 mV. Of course rather high average current density can be achieved in a force cooled bundle conductor but due to this high current density the discharge in case of emergency must be rather fast.

In order to give a rough impression about the energetics involved in superconductors and their cooling[50], table VIII gives a comparison for the ANL-EPR, ORNL-EPR, FINTOR and UWMAK III. Looking into the heat inputs per unit volume

TABLE VIII		ANL EPR 1975			ANL EPR 1976		ORNL EPR 1976			FINTOR	UWMAK III	
COND.		TF	PF	PF	TF	PF	TF	TF	PF	TF	TF	PF
COOLING		BATH	2PHASE	BATH	BATH	BATH	SUPCRIT	SUPCRIT	SUPCRIT	2PHASE	BATH	BATH
S C		NbTi	NbTi	NbTi	NbTi	NbTi	Nb_3Sn	NbTi	NbTi	NbTi	NbTi	NbTi
I	kA	10	40	40	60	40	19,5	19,5	25	10	11	14,5
N_{FIL}		425262	1904877	1889360		1756500	677040	19530	5621114	2760	182	182
D_{FIL}	μ	10	5	5	10	5	1/10 ∅	40	4	260	380	380
SUDDEN HEAT LIMIT	$\frac{mJ}{cm^3}$						100	300				
HEAT LIMIT		250	~ 2250	250			300	700		1184	~250	~250
AC LOSS PER CYCLE		12,2	18,3							7,56		20,1
AC LOSS PER TRANSIT							48	144				
PUMPING LOSS		—	—	—	—	—	82	82		—	—	—
COND 0,1mm 30 MN/m		100	100	100	100	100	100	100	100	100	100	100
STORED EN./COND. VOL.	$\frac{J}{cm^3}$	97,5			226		232	232		137	394	56,8

of conductor it is interesting to note that the capability of all 3 cooling systems represented in the list of Table VIII are rather similar. Owing to a 10 % helium fraction the ANL and UWMAK III toroidal coil conductors can take a maximum heat load of 250 mJ/cm^3 and the ORNL EPR shows values of the same order. Its forced cooling scheme has been optimized versus pumping power. Despite this fact, the heat load from pumping is of the same order of magnitude as the ac-losses. For all 4 designs the heat input from ac-losses per cycle is in the range between

3 and 20 $\frac{mJ}{cm^3}$. These are averaged data. Locally the difference between heat input and cooling limit may be much smaller. It should also in comparison be born in mind - which throws a light on possible difficulties with conductors and winding schemes which cannot be absolutely fixed - that at a transverse pressure of e.g. 30 MN/m^2 a conductor movement of 0.1 mm will cause a heat input of about 100 mJ/cm^3 [50]. Conductor and winding designs such as the two phase helium cooled poloidal field cable conductor for ANL-EPR or the FINTOR toroidal field hollow conductor, however, can offer 1000 to 2000 $\frac{mJ}{cm^3}$ at the expense of a large helium inventory. It may be further of interest to note that the stored energy per unit conductor volume, which of course increases with the machine size, varies between 90 and 400 J/cm^3 thus again pointing to the crucial safety discharge problem.

As can be seen from table IX showing the single loss contributions, one important conductor design criterium for Tokamak magnets is to minimize the ac-losses and possibly about equalize the single loss contributions. As can be

TABLE IX HEAT LOADS IN SUPERCONDUCTING MAGNETS (W AT 4 K)

	ANL 1975	ANL 1976	ORNL 1976	UWMAK III
TF COILS	WITHOUT FIELD SHIELD	WITH AL-SHIELD		
CONDUCT.	208	160	1400	1430
SURFACE RAD.			500	375
HYST.+ EDDY	55832		1300	0 (?)
RADIAT.				
	256	1500	16000	600
PUMPING	—		6380	
SHIELD	—	4800 + 136000 (18 K)	—	—
OH + EF COILS				
CONDUCT.	2090	331	1500	1600
SURF. RAD.			1600	231
HYST.+EDDY		3594	1360	1740
RADIAT.		80	200	
PUMPING				
			8240 (EF COILS NORM.COND.)	
TF COILS LEADS		120	780 L/H (4 PAIRS)	1718
PF COILS LEADS			120 L/H (2 PAIRS)	6730
CONDUCTOR JOINTS				300
TRANSFER LINES				800

seen, going from EPRs to the full scale reactor eases the conductor design, since less and larger superconductor filaments will suffice. The proposed poloidal conductors for currents between 15 and 40 kA - one ORNL EPR[36] again foresees paralleling them up to 175 kA - in the EPR designs require millions of filaments compared to about 200 for UWMAK III. The very high hysteresis losses of the earlier ANL-EPR toroidal field conductor shows the problems connected with outside vertical field windings and the high pumping power in the coil system of the ORNL-EPR seems to indicate that supercritical helium cooling of bundle like conductors if at all can only work with more space for the coolant flow, that is at a lower overall current density. This might call e.g. for a configuration like the Karlsruhe conductor as shown in the previous lecture[51]. The large radiation heating in the ORNL EPR magnet certainly would be reduced by 10 cm more shielding thickness to about an order of magnitude less. Also it would appear that the helium ducting losses in large installations can easily come into the kW-range and should be considered.

Figs. 15 - 19 give a collection of conductor and winding cross sections for a few selected designs[9,10,11,14,27]. Looking at the various conductor and winding proposals one should mention that a thorough comprehensive study of the different cooling schemes also including transient cooling and their application to different conductor and winding configurations is a task which has just been identified[11,44]. For the detailed design of a fully stabilized magnet there are still many alternatives and we do not yet know the optimum solutions.

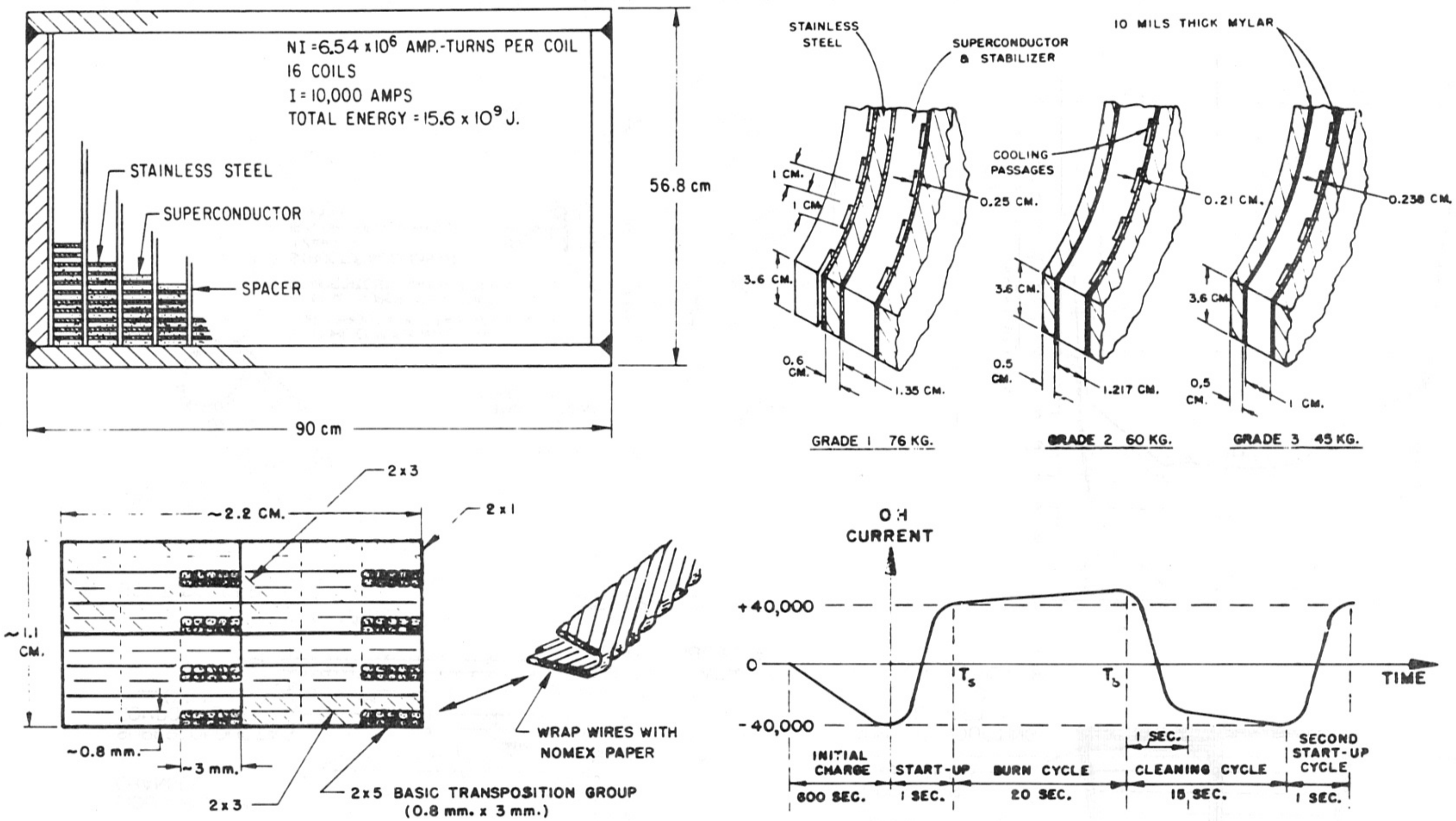


Fig. 15 ANL-EPR 1975: TOROIDAL COILS WINDING AND CONDUCTOR CROSS SECTIONS
OH-COIL CONDUCTOR CROSS SECTION AND OH-CURRENT VS. TIME

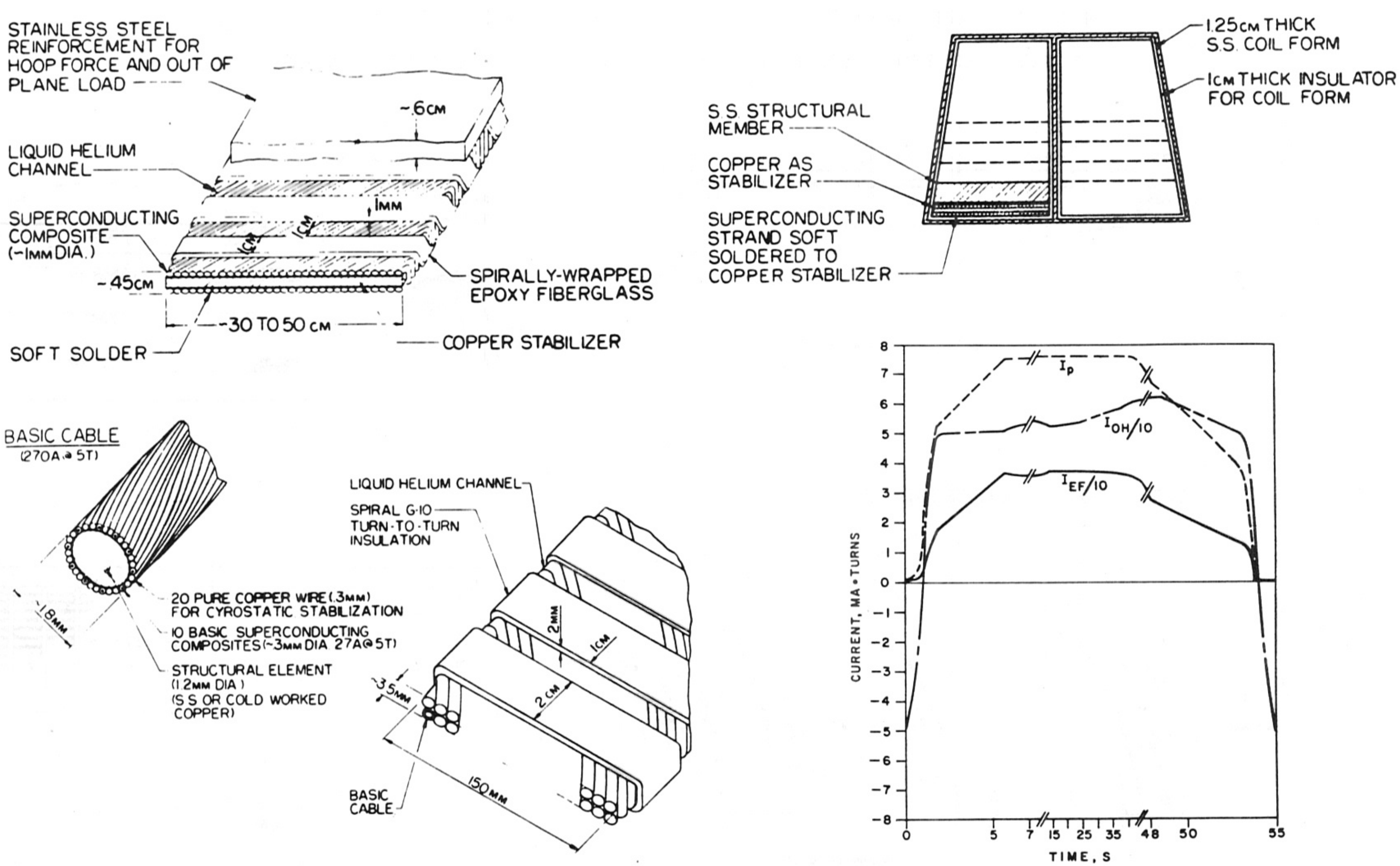


Fig. 16 ANL-EPR 1976: TOROIDAL COILS WINDING AND CONDUCTOR CROSS SECTIONS

OH-COIL CONDUCTOR CROSS SECTION AND OH-CURRENT VS. TIME

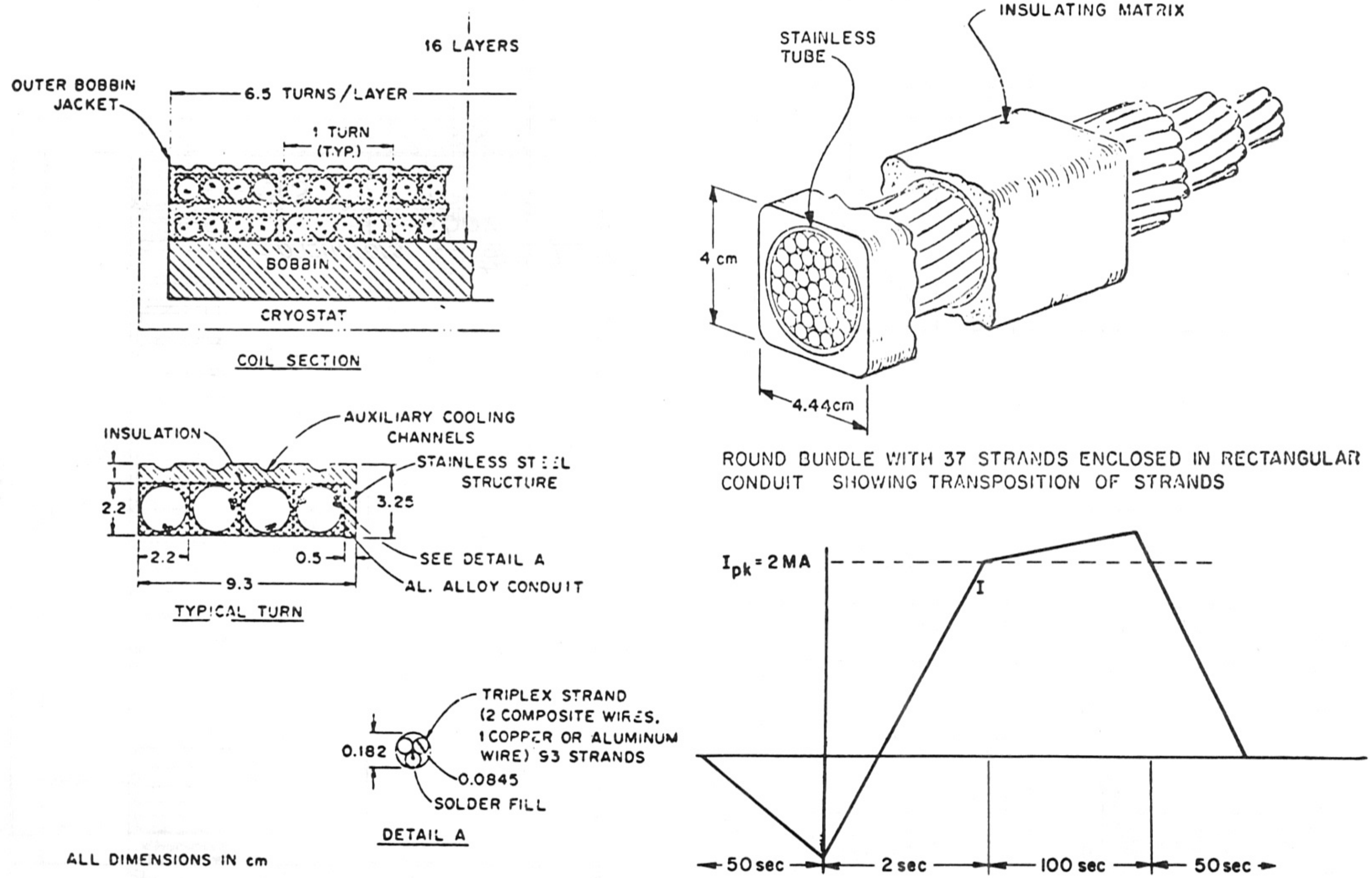


Fig. 17 ORNL-EPR 1976: TOROIDAL COILS WINDING AND CONDUCTOR CROSS SECTIONS
OH-COIL CONDUCTOR CROSS SECTION AND OH-CURRENT VS. TIME

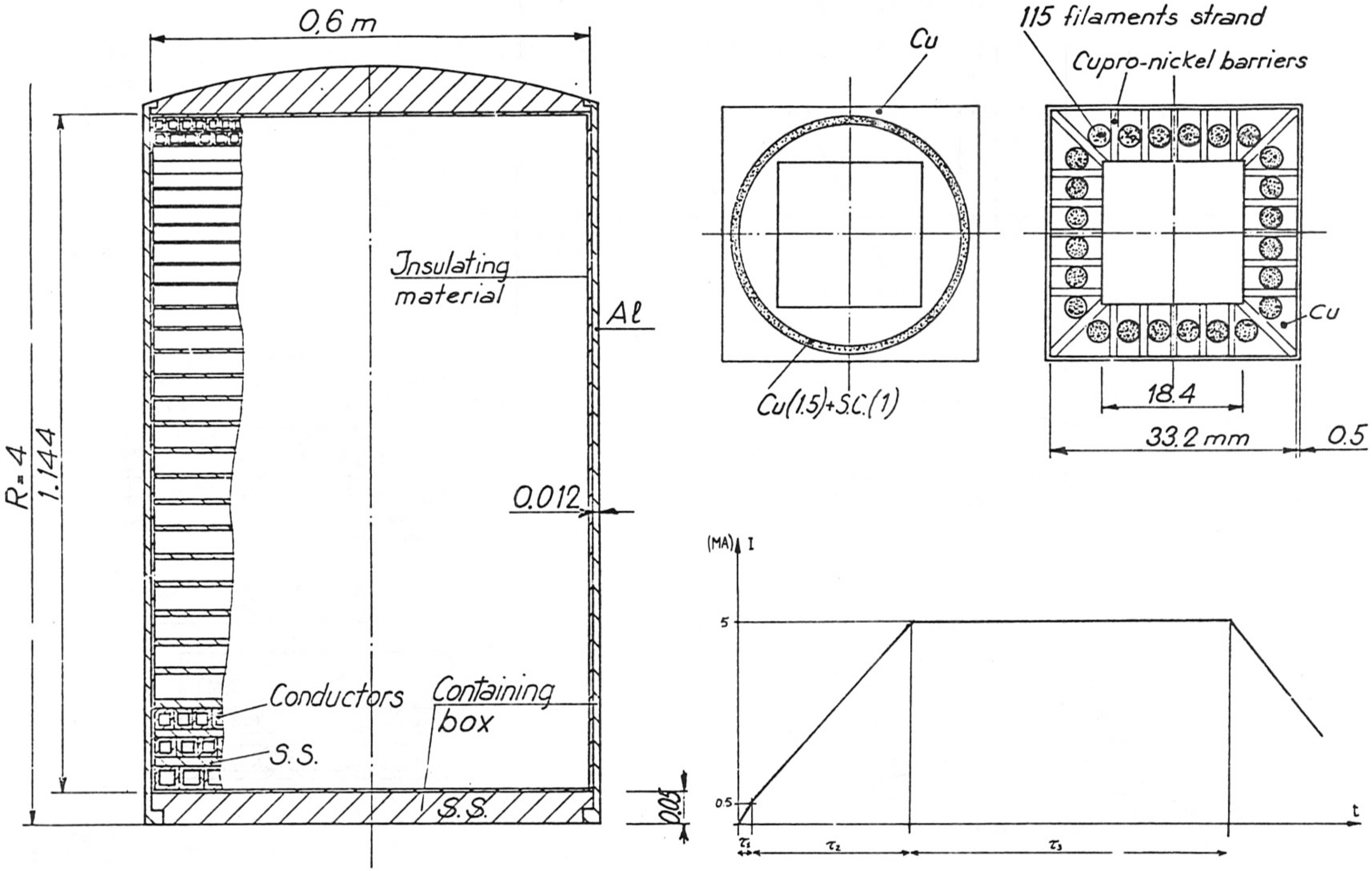


Fig. 18 FINTOR 1976 : TOROIDAL COILS WINDING AND CONDUCTOR CROSS SECTIONS

OH-CURRENT VS. TIME

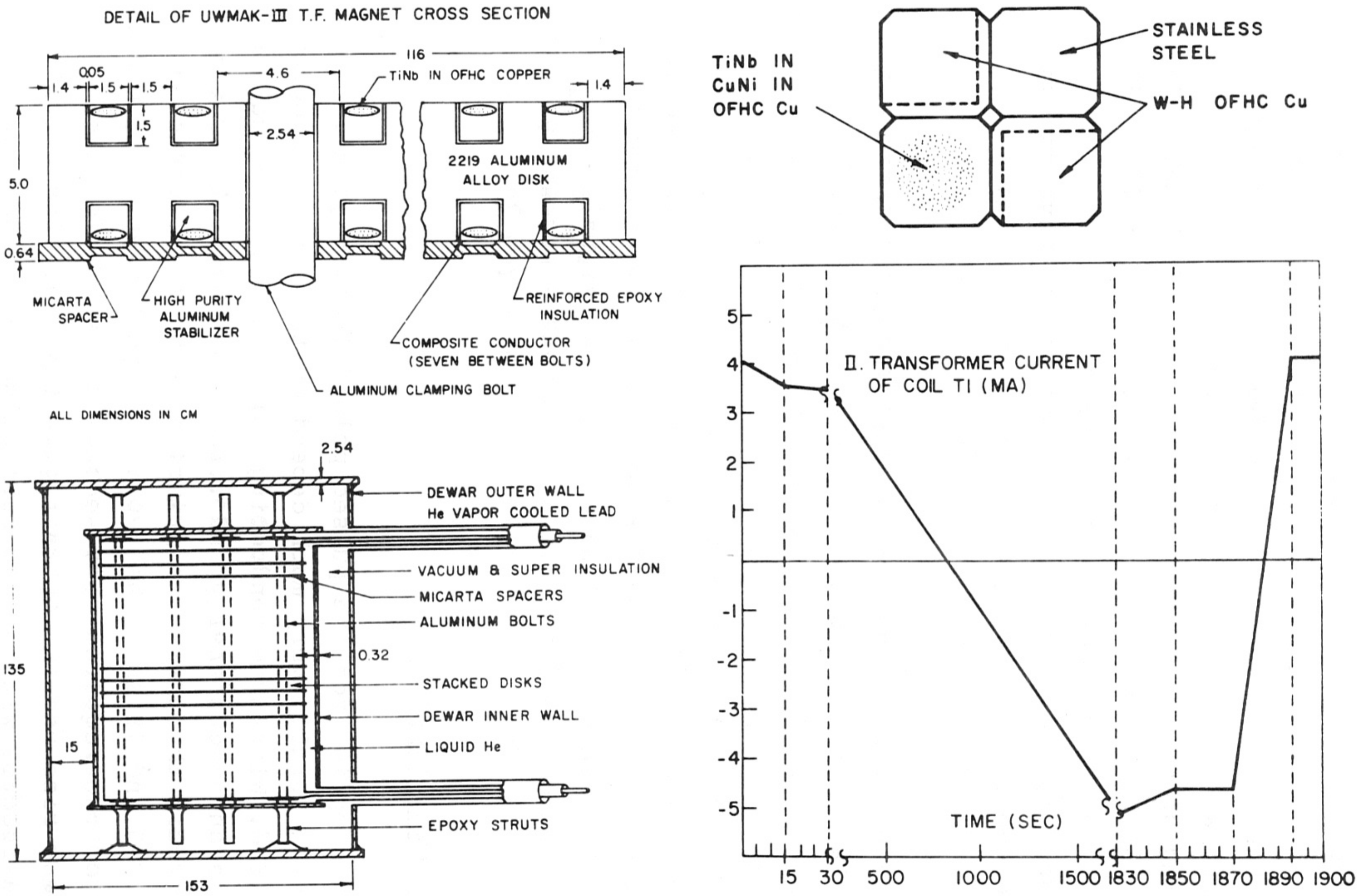


Fig. 19 UWMAK III 1976: TOROIDAL COILS WINDING AND CONDUCTOR CROSS SECTIONS

OH-COIL CONDUCTOR CROSS SECTION AND OH-CURRENT VS. TIME

3.5 Reliability

The availability of an integral Tokamak magnet system necessarily depends on the reliability of all its components. An easy repair scheme in case of a major magnet failure can hardly be thought of. It has been said that the toroidal and poloidal magnets must operate without any major failure during the plant lifetime. Warming up of the magnets to room temperature may occur once per 1 or 2 years for annealing radiation induced damage in the conductor stabilizer and for routine inspection. The refrigeration plant should have enough spare capacity to bridge the failure of one major refrigerator section.

For safety reasons one of the most important components in a large magnet system is a redundant monitor system to detect any irregularity as soon as possible. Especially important is the detection of effects like fatigue or gradual movement which may accumulate over many cycles. For coincidence of 2 signals required this might require up to 5 sensing devices for one quantity monitored in order to ensure cancelling of erroneous signals. Thousands of sensing devices will thus be necessary[43].

3.6 Power supplies

Power supplies for the pulsed poloidal coils of a Tokamak magnet naturally very much depend on the current program required. With superconducting coils, where minimization of instantaneous energy and forces is essential for the OH circuit there are two principal schemes: Inserts in Figs. 15 and 19 show the typical shape for an EPR[9], where most of the required volt seconds are spent for the current rise and in contrast give the shape for a full scale reactor with a long burn pulse where the volt second requirement of the burn pulse is much larger than for the current rise[14]. In the first case the large power demand occurs during the plasma current rise and fall and a high power fast storage device is required for these periods whereas in the case of a long burn pulse the large power demand occurs in the recharge phase of the OH coil. In these phases GVA power requirements may occur for short periods of time which have to be supplied by some special storage device which for cost reasons may be a homopolar generator. The remaining pulse power requirements could in principle be taken from the grid via invertor rectifiers. Recent experience, however, indicates that the use of solid state controlled rectifiers directly fed from the grid may not be a straightforward solution even if the pulsed power demand would not be a problem for the grid[52]. The possible influence from the grid on the rectifier in emergency cases and vice versa calls for a link which electrically decouples the grid from the pulsed magnet system such as a motor-generator set preferably storing energy in the mass of the generator rotor itself or a superconducting storage coil[10, 11]. The reliability of the overall Tokamak magnet system strongly depends on the reliability of the pulsed power supplies also. These have to be almost absolutely reliable since the safe controlled shut down of the plasma current is the first and most important step in the sequence of any safety discharge of the overall magnet system.

3.7 Scaling

Having touched a number of technical design problems associated with Tokamak magnets in conclusion it may be interesting to look briefly into the connection between Tokamak magnet and overall Tokamak scaling. One can set up a simplified geometrical model of a Tokamak according to Fig. 20[53, 54]. Some basic assumptions about the plasma quality (essentially $\beta_{pol} = \sqrt{A}$; $q = 2.5$) together with

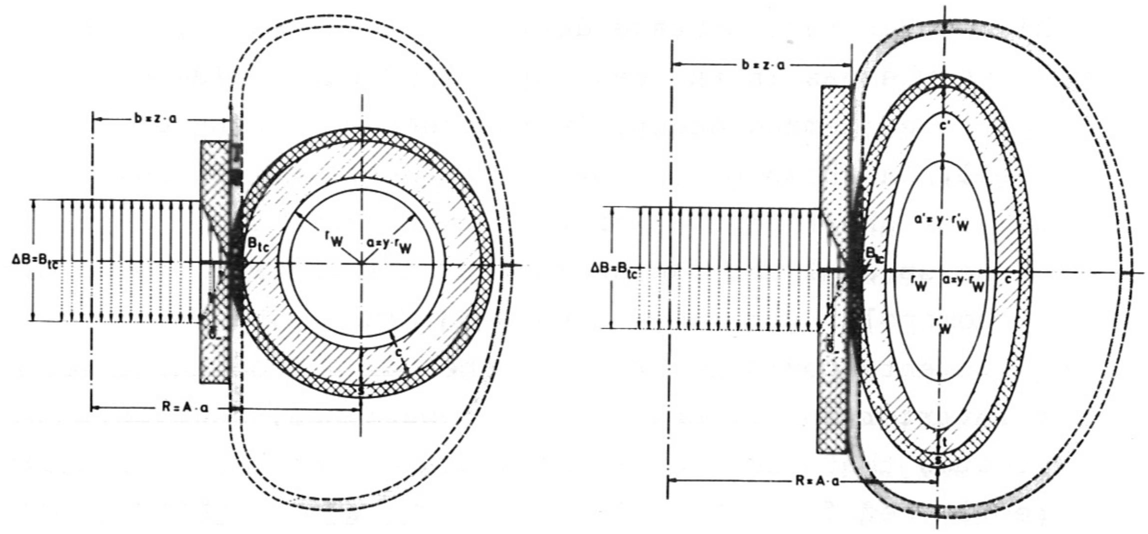


Fig. 20 Tokamak reactor - scaling model

Cylindrical plasma cross section (CCS) Elliptical plasma cross section (ECS)

technical constraints such as mechanical stress limits, full stability of the toroidal magnet, flux density swing in the transformer core and space requirements for the blanket and shield lead to a set of consistent parameters for any size of reactor subject to the constraints imposed. Comparing toroidal magnet stored energies vs. reactor power for circular and modest-elliptical ($\frac{a'}{a}$ = 2) plasma cross sections (Fig. 21) shows for the machine size between EPR and full-scale reactors the strong dependance of Tokamak magnet requirements on the further achievements in plasma fusion experiments and materials development. With an elongated plasma cross section NbTi magnets at least for full-scale reactor size would fully suffice under the restrictions imposed by the first wall load limit to be envisaged at present. Higher magnetic fields calling for Nb_3Sn would seem to be required for low power reactors rather than for large ones in order to increase their low power wall loading. Fig. 22 shows the plasma current and the plasma poloidal field energy vs. reactor power according to the same scaling. When taking into account that the

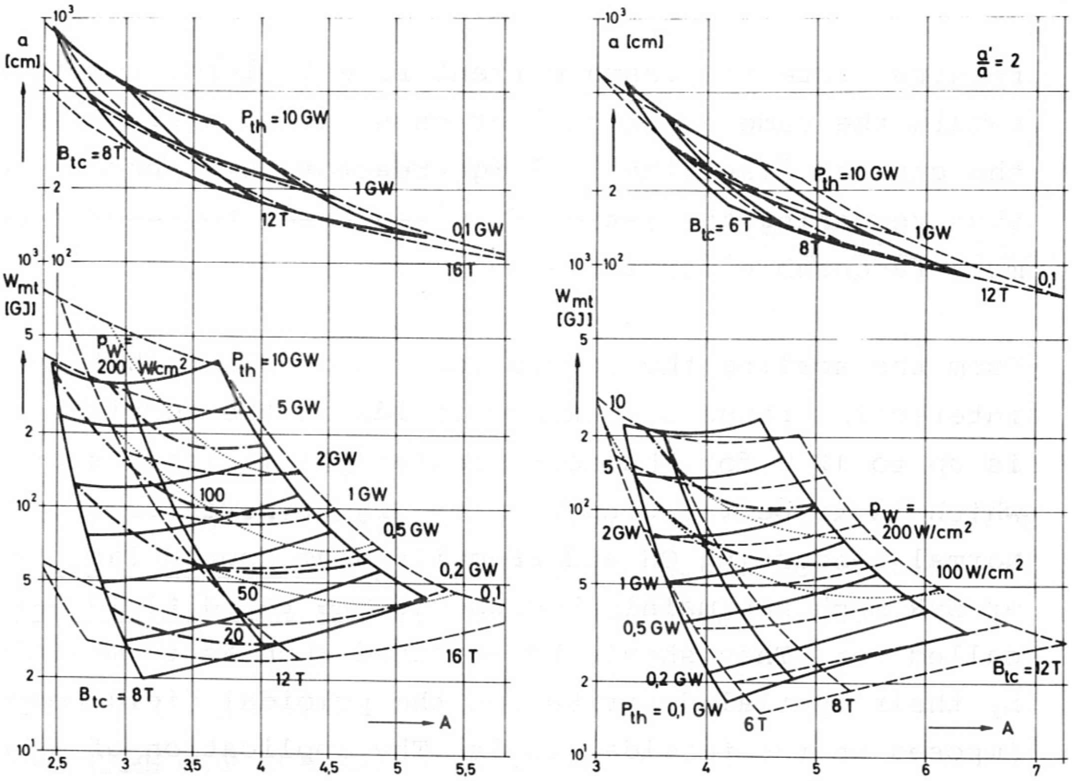


Fig. 21 Plasma radius and toroidal magnet stored energy vs. plasma aspect ratio

(CCS) (ECS)

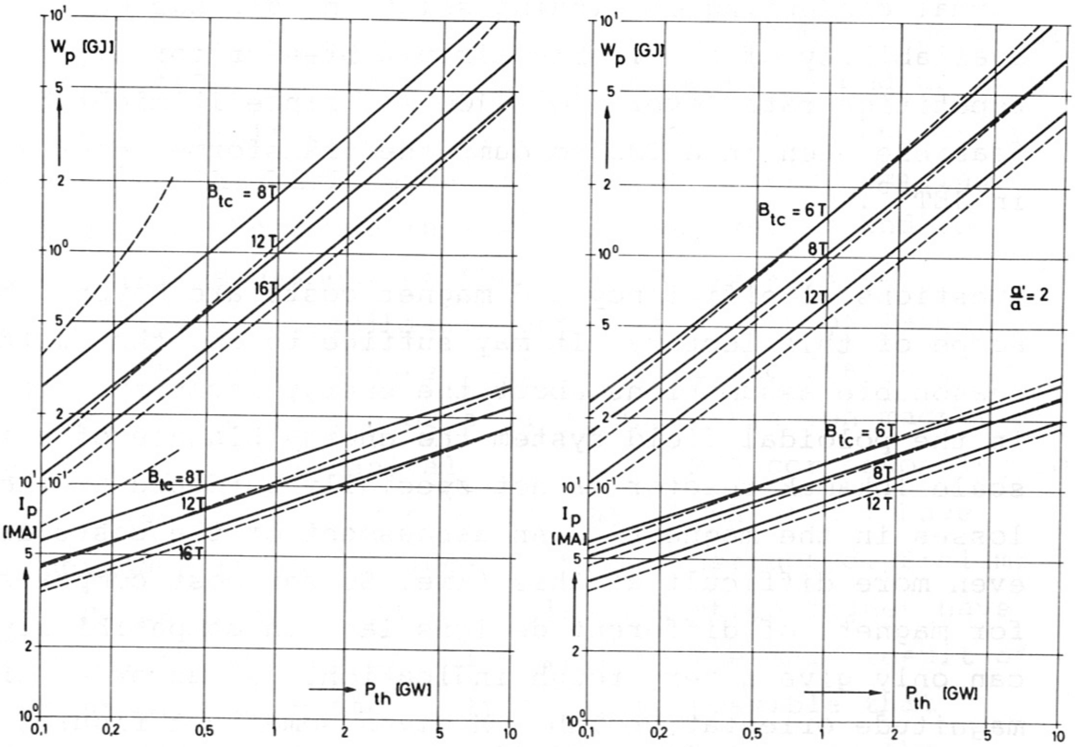


Fig. 22 Plasma current and poloidal magnetic energy vs. reactor power

(CCS) (ECS)

required rate of plasma current rise in large reactors may remain the same as in smaller ones, it is again seen that the current rise time in large reactors will become longer thus rendering the ratio of pulsed power to reactor power more favourable for large units.

From the scaling the conclusion would follow that the interesting range of maximum fields in the toroidal magnet is up to 12 T for the non-circular plasma cross section which is to be favoured for its higher power density. Normal conducting OH and EF-coils seem appropriate in TNS or EPR machines mainly because of the rapid field variations called for. They should be arranged such as to minimize by their spatial distribution the poloidal field components imposed on the toroidal coils. The application of normal conducting OH coils may imply a larger plasma aspect ratio than shown in the scaling because of the otherwise very high field in the transformer core associated with an asymmetric flux swing for loss minimization[35]. Also the normal conducting OH-winding solution depends on the availability of a reliable circuit breaker for high repetition rate (every 30 - 100 s), since it might be still feasible even in a TNS to dump the transformer energy like in JET[55].

Questions of efficiency and magnet costs are beyond the scope of this lecture. It may suffice to say that with reasonable assumptions about the energy recovery percentage in the poloidal field system the energy blanace of a full-scale Tokamak reactor is not specially burdened by the losses in the magnets[56]. An assessment of the costs is even more difficult at this time. So far cost comparisons for magnets of different designs lack in compatibility and can only give a very rough indication. For an order of magnitude orientation Tab. X gives some cost figures from published designs[10,13,14,16].

TABLE X MAGNET COST INCL. REFRIG. AND POWER SUPPLIES IN M $

	TF	OH	EF	REFRIG.	POW.SUPPL. TF,	OH+EF	Σ
GA ITR (TNS)	48						
ANL EPR 19/6		100		20	~16		136
UWMAK II	162	21		19	4	33	229
UWMAK III	77	6	21	19	2	33	157

The obvious manifold tasks in large Tokamak magnet development have caused some corresponding program activities in several countries which will be briefly reported.

4. PROGRAMS FOR TOKAMAK MAGNET DEVELOPMENT

The European associated laboratories in the frame of the third pluriannual program 1976/80 which up to the final agreement on JET is only partially approved have established a development program for superconducting magnets the first two years (1976/77) phase of which has been agreed upon. This first definition phase of the program will be devoted to extending the basic understanding of superconducting materials and their use in large magnet systems and to the detailed assessment of the alternative ways of demonstrating the feasibility and reliability of large magnets. Subsequent phases would involve the design, construction and operation of a large demonstration magnet at an estimated (1976) capital cost of 12 MUA corresponding to 17.5 M $. So far NbTi conductors for 10 kA at 8 T have been designed and prototypes of them constructed. Preliminary studies of the form for a demonstration magnet have been made, the choice of which will be the main result of the first program phase[57]. It is quite possible that opportunities for international collaboration in the testing of large coils will strongly influence both the form of a demonstration magnet and the time scale of its development. The ongoing discussions also include the possible construction of a complete Tokamak experiment with superconducting toroidal field coils. An organizational structure for the first 2 years phase is being formed. Table XI[58] shows a list of tasks to be covered and the associated manpower foreseen. The work during the definition phase is intended to be carried out mainly in the so-called GESSS laboratories (Karlsruhe, Rutherford, Saclay) under agreement with the corresponding fusion laboratories (Garching, Culham, Fontenay-aux-Roses) and by groups in Frascati and Petten. The program foresees an investment expenditure for test equipment of 1.5 MUA $\hat{=}$ 2.2 M $ during the first 2 years.

TABLE XI

EUROPEAN SUPERCONDUCTING MAGNET PROGRAM 1976 / 77

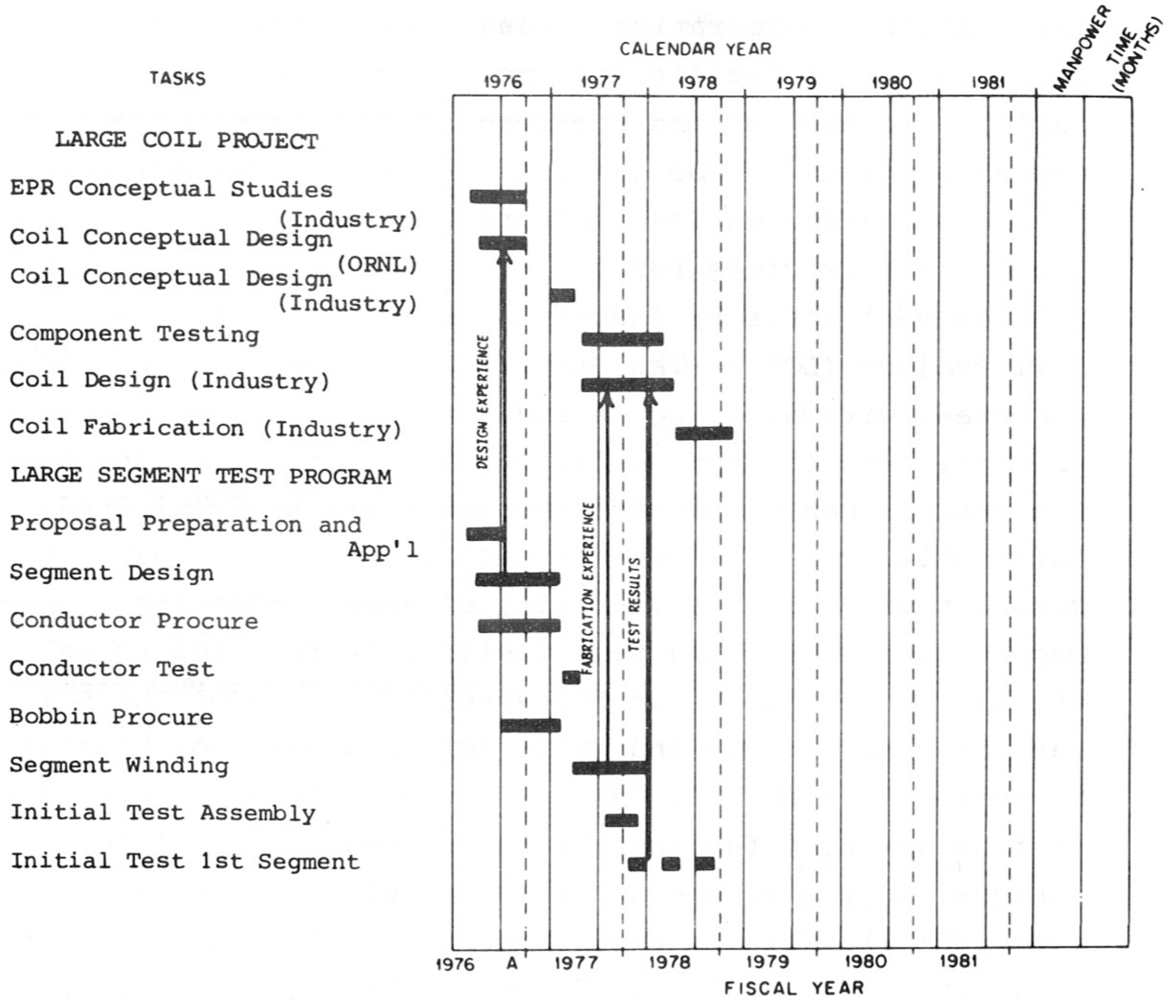
Summary of manpower requirements during first two years

TASK		Professional Manpower (pmy)	Sub-total
1.	Outline Studies of post-JET Experiment	3	
2.	Alternative Designs		
	2.1 Full torus	4	
	2.2 Single coil or cluster	4	
3.	Definition and Preliminary Design	4	15
4.	Specific Tests		
	4.1 Conductors	7	
	4.2 Computer programmes	6	
	4.3 Cryogenics	4	
	4.4 Electrical insulation	3	
	4.5 Protection	5	
	4.6 Test facilities	5	30
5.	Supporting Activities		
	5.1 Filamentary A-15 conductors	4	
	5.2 Alternative stabilisation and cooling	2	
	5.3 Energy transfer	3	
	5.4 Radiation damage	2	11
	TOTAL		56

Presently negotiations are under way concerning possible international cooperation through IEA at the proposed large coil test facility in the United States. This facility is part of the American Tokamak superconducting magnet program[59] to be carried out in or under guidance of the Oak Ridge National Laboratory supervised by ERDA. The program provides for the early fabrication of large EPR relevant coils by industrial subcontractors (Large Coil Project (LCP)). The aim is to utilize the experience and creativity existing in industry at every stage of the project, from conceptual design through fabrication of the coils. When completed, the industrially fabricated coils will be assembled in a test facility to provide a demonstration of the viability of superconducting magnets for EPRs. This test facility is referred to as the Large Coil Test, but the specific nature of the LCT has not been decided and still is the subject of studies. Concurrently with LCP, ORNL will undertake basic development of advanced fabrication techniques and novel coil designs. In the superconducting magnet development program (SCMDP), details are given for the design, fabrication and testing of Large Coil Segments which are similar in bore to the coils produced in the LCP but reduced in cross section so that thin coil would be an appropriate description. The large size fabrication experience will enable ORNL to provide guidance, technical input and evaluate the industrial participation in the LCP. The interrelation between the large segment tests and the LCP is shown in Table XII for the next two fiscal years.

The small scale experiments in the frame of SCMDP are in direct support of the above activities as well as providing a base for the poloidal field requirements. The initial pulse field experiment will be on a small scale and are aimed at selecting the conductor and winding configuration which can meet the rate of rise constraints with acceptable losses. The second phase will require increase in size to achieve the field magnitude as well

TABLE XII



INTERRELATION OF SCMDP LARGE SEGMENT TESTS AND LCP IN INDUSTRY

as the rate of rise of field simultaneously. The third phase will be construction of a model coil which will be large enough to require structural constraint. The model coil can be scaled to the prototype poloidal field coil when a reference design is accepted for application in a particular superconducting Tokamak.

Table XIII gives a summary of the tasks and manpower foreseen at ORNL throughout 1976 and 1977 for SCMDP and LCP. The cost of the LCP has been evaluated in some detail for the compact torus version leading to a total investment cost of 28 M $. The program parts shown here represent only the Tokamak oriented part of the US superconducting magnet program. There is a strong inter-

TABLE XIII ORNL-Superconducting magnet program. Summary of ORNL-manpower requirements during first two years 1976/77

	Professional manpower (pmy)
Subprogram A: System Design	2,0
Subprogram B: Coil Design	5,0
Subprogram C: Conductor Selection and Test	12,5
Subprogram D: Radiation Effects on Superconducting Coils	2,3
Subprogram E: Coil Protection, Eddy Current Shielding, and Power Supply	3,7
Subprogram F: Structural Analysis and Materials Investigation	9,0
Subprogram G: Cryogenics and Refrigeration	2,0
Subprogram H: Subsize Coil Fabrication	6,6
Subprogram I: Large Coil Project	5,0
Subprogram J: Coil Testing and Evaluation	8,9
	57,0

connection between this part and the mirror- and energy storage oriented program parts.

Japan having a very strong national Tokamak program is considering the future activities for superconducting magnets in fusion. Japan together with the European Communities and Switzerland takes part in the IEA-discussions on international cooperation. In Japan the main effort will be directed to the development of superconducting magnets for a Tokamak device coming after JT60[60]. Preliminary figures for such a machine which is called fusion core mock-up test facility have been quoted as follows. Preliminary dimensions are 6 m major radius, 1.2 m plasma minor radius and the axial magnetic field is 5 T. The first priority in Tokamak application is given to the development of superconducting toroidal coils, followed by Ohmic coils and EF coils. In future the superconducting energy storage and transfer device for Tokamak and other applications will also be studied. The development work will be carried out mainly at JAERI in collaboration with national laboratories having experience in superconducting magnet development. Although a long term program is not yet formulated, these organizations are now making plans for the next year (1977); development of a large current stabilized superconducting conductor and construction of a simulation facility of Tokamak toroidal coils at JAERI the design of which has already started. Poloidal effects and high field materials are studied at ETL and NRIM. The industries will support these activities on contract basis with these laboratories.

In the USSR already smaller Tokamaks like T-7 and possibly T10 in a second phase will be built with superconducting toroidal magnets. The current Soviet reference design (EPR) proposes D-shaped toroidal magnet coils built with NbTi conductors (B_{max} = 7.8 T), the plasma relevant data being I_p = 6.3 MA, R = 7.2 m, B_{t_o} = 4 T [61].

Of course in other countries fusion oriented technology programs including superconducting magnets are being considered, too.

In the frame of the US superconducting fusion magnet program, which seems to be the most elaborate and strategy based of the so far known programs the already mentioned large coil test facility plays a major role in early involvement of industry and conclusive feasibility demonstration for fusion application. The concept which was preferred among 5 alternate arrangements the so-called compact torus foresees 6 approximately D-shaped coils arranged in a symmetric toroidal geometry with a slightly tougher magnet aspect ratio compared to a

Tokamak experiment. Fig. 23 shows the compact torus coil of the LCP and for comparison the JET toroidal magnet cross section. The compact torus definition has been arrived at by means of the following procedure [62]:

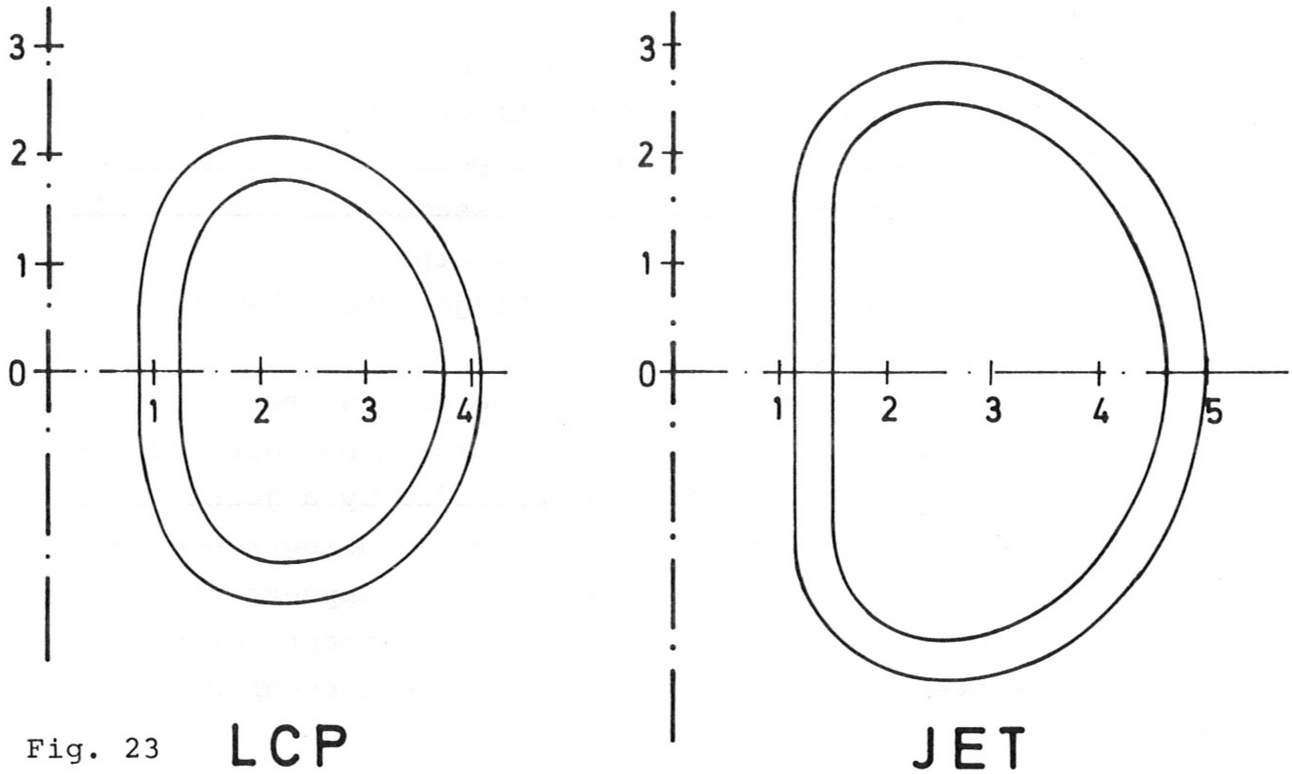


Fig. 23

1. The basis is the reference EPR design.
2. By decreasing the coil size, but keeping the current density and maximum field at the inner coil circumference the stresses will be somewhat higher than in the EPR and no extrapolation in current density coil be necessary later.
3. The result is that with 6 coils and a reduced magnet aspect ratio the circumferencial force distribution can be held within 10s of % of that in the EPR size coil.
4. Starting from an EPR design with 8 T maximum field and 2500 A/cm^2 current density in the winding, inner coil dimensions of about 2.5 m horizontal diameter and about 3.5 m vertical diameter - which are within the capability of existing machining equipment - can simulate EPR coil conditions relatively closely. Further conditions are:

- The conductor and coolant characteristics are the same as for EPR. Local heat inputs by built in heaters shall test the coils response and also simulate radiation heating.
- The EPR field force and strain pattern in the coils shall be realized.
- The pulsed poloidal field effects have to be simulated, either locally or integrally. This point is under discussion since it depends on the choice of the poloidal configuration assumed for the EPR and strongly influences the conductor and winding design.
- An appropriate safety discharge scheme has to be tested in LCP.
- Another point under consideration is the dewar configuration for the compact torus. The most realistic test conditions would be provided by a joint fitted design with a common inner part housing the cold central support ring, however, it appears that in an overall large dewar housing the compact torus a convenient flexibility including the test of coils with fitted dewars is provided.
- The LCP has to demonstrate reliable operation.

LCP has been offered for contributions by one or more coils from other countries. Fig. 24 shows a preliminary sketch of the large coil test facility as it looked in August/ September 1976 [62].

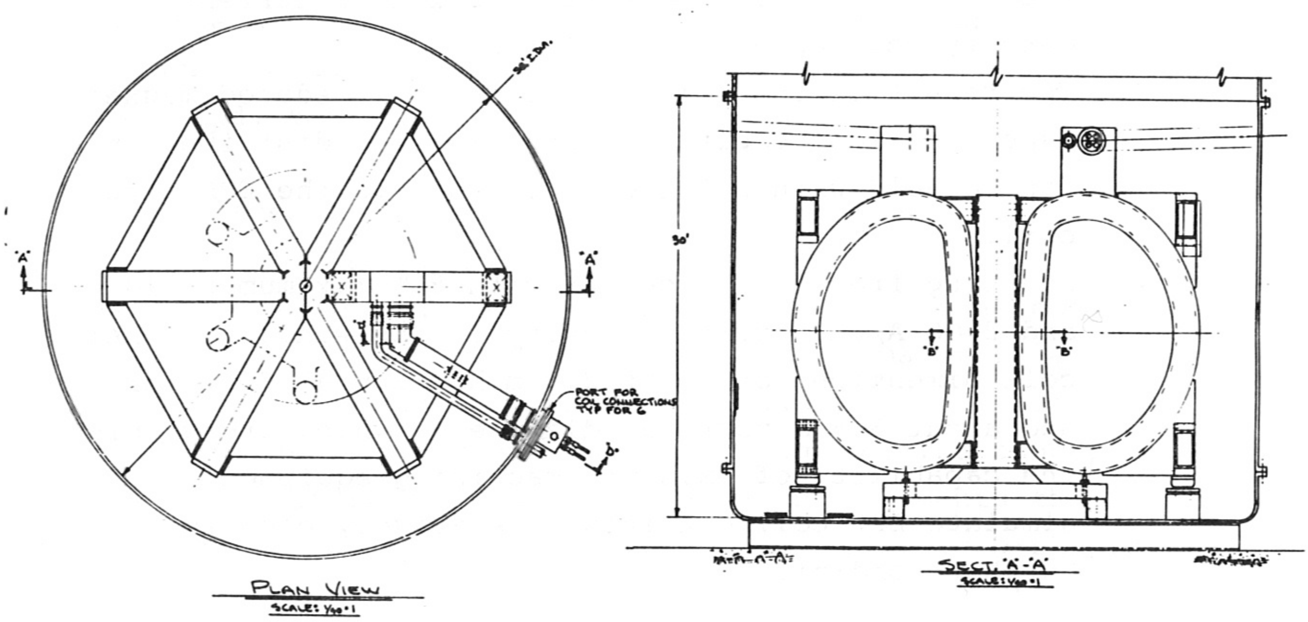


Fig. 24 Preliminary configuration obsolete 9-2-76

5. CONCLUSION

The technology of superconducting Tokamak magnets provides challenging tasks both for specific developments in laboratories and heavy conductor and hardware construction in industry. The need to increase the efforts in this field is widely acknowledged at least to the extent that the generation of toroidal experiments after JET, TFTR and JT-60 [63] provided this generation of experiments is going to be successful will require a superconducting toroidal magnet.

Relevant experience can be gained from a demonstration setup with torus relevant coils in a Tokamak relevant environment and at a size comparable to e.g. JET coils, but not much smaller. While the specific nature of such a demonstration set-up is still being considered, Table XIV[64] including preliminary LCP data briefly summarizes the present situation. The justification for substantial efforts in superconducting magnet technology for fusion lies in the belief in the success of the coming toroidal fusion experiments.

TABLE XIV

COIL SHAPE	DIMENSION	DESIGNED	OPERATION	$\hat{P}_{MT}$(GW)	B_{MAX}(T)
LCP	2,5 x 3,5	1976	1980 ?	-	8,o
JET	3,12 x 4,9	1975	1981 ?	o,33	6,9
GA ITR (TNS)	4,5 x 7,2	1976	1985 ?	-	8,o
T 20	5,4 x 8,3	1975	1985 ?	1,2o	7,8
ANL EPR	7,7 x 11,9	1975	1992 ?	-	7,5
ANL EPR	7,78 x 12,6	1976	1992 ?	-	10,o

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