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Report

A PULSED TOKAMAK REACTOR STUDY

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ABSTRACT

This report considers the choice of parameters for a pulsed Tokamak reactor with net electrical output of 600 MW_e. The physics and engineering assumptions are described in detail with particular emphasis on the magnetic field design. This leads to reactor parameters with 2 m minor radius, an aspect ratio of 3.9 and energy multiplication factor Q of 13. The sensitivity of the chosen parameters to various changes in the assumptions is analysed. A comparison is made with a recent study of the Reversed Field Pinch Reactor, showing that the two containment systems appear to have similar potential as fusion power sources.

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1. INTRODUCTION

The Tokamak is commonly considered the most likely confinement geometry to be developed as a fusion power reactor, and has been taken as the basis of many design studies of a deuterium-tritium burning reactor. It is often assumed in these studies that quasi-steady state operation will be possible, with refuelling to maintain the plasma density and an exhaust system to remove both the helium formed by the fusion reactions and plasma diffusing to the wall. Such operation may not be possible, however, because new fuel is unable to penetrate to the centre of the plasma, because the helium does not diffuse sufficiently quickly from the centre of the plasma or because the rate of build-up of impurities in the plasma is too rapid. It is therefore of interest to consider a pulsed reactor operating cycle without either refuelling or exhaust during the burn, and in which the unreacted fuel, reaction products, and impurities are removed and new fuel is introduced between relatively short burn periods.

This study is based on the experience gained in two previous Culham studies of a Tokamak - the Culham Conceptual Tokamak reactors Mk II A and Mk II B [1]. These studies assumed quasi-steady state operation, and the latter included a single null poloidal divertor. The present study, designated Mk II C, employs the same general engineering features and construction. As before an elliptically shaped plasma cross-section, a low plasma aspect ratio, and a similar value of the plasma pressure ratio β_ϕ are used. The physical dimensions of the reactor are also similar to the previous designs as a result of using a lower value for the power loading of the first wall. In this new study, however, the spatial profiles of plasma parameters have been calculated in much greater detail and the distribution of poloidal field windings defined more accurately. The optimum durations of the heating and burn phases have also been considered as these are important in defining the parameters of a pulsed reactor.

This study is also based on a recent collaborative study of a pulsed Reversed Field Pinch reactor[2-7]. The net power output of the two reactors has been made the same, as have several other assumptions, so that a comparison of the parameters and performance of the two confinement systems can be made. This new Tokamak study builds on the experience gained in the Reversed Field Pinch study, particularly in relation to the assumptions for the various stages of the operating cycle and the plasma conditions during the burn stage. The same methods for evaluating the energy multiplication factor Q and the cost have been employed. In making comparisons, however, it must be remembered that the present knowledge of plasma confinement in the two systems is very different, and that assumptions concerning plasma profiles and the stability of equilibria will strongly affect the apparent relative merits of the two systems.

The following sections describe the physics assumptions (section 2) and engineering assumptions (section 3) upon which the choice of reactor parameters (section 4) is based. The sensitivity of the energy multiplication factor Q and reactor capital cost to some of these assumptions is estimated (section 5), before comparing the Tokamak design with the Reversed Field Pinch reactor (section 6).

2. PLASMA PHYSICS ASSUMPTIONS AND CONSTRAINTS

In order to design a reactor certain assumptions must be made concerning the relationship between the spatial profiles of fuel density, temperature, current density and toroidal field during the burn phase of the cycle. The absolute magnitude of any one of these parameter profiles is then determined by the plasma dimensions which in turn depend on the engineering constraints described in section 3. It is also necessary to establish the reactor operating cycle and to analyse the energy balance of the heating and burn phases using these profiles. The assumptions needed to carry out this detailed analysis are described below and summarised in Table I. Several assumptions have been directly derived from previous studies of the pulsed Reversed Field Pinch (RFP) reactor[4,5].

TABLE I

SUMMARY OF PHYSICS ASSUMPTIONS

Plasma elongation, b/a	1.68	
D shape parameter, d	0.17	
Poloidal beta, β_p	(R/a) 0.66	
Toroidal beta, $\bar{\beta}_\phi$	0.08	
Safety factor on axis, q_o	1.0	
Safety factor at edge, q_a	$\gtrsim$ 2.0	
Neutral beam heating efficiency, η_I	0.4	
Ratio of heating to cycle time, τ_h/τ_c	0.15	
Burn temperature, T	10	keV
Fractional burnup, f_b	0.3	
Recoverable thermonuclear energy/ reaction, Q_R	20	MeV
Blanket multiplication	1.135	
Fraction of energy recovered during rundown	0.4	
Current rise time	1.0	sec
Current rundown time	1.0	sec
Off time	6.0	sec

2.1 Operating Cycle

Operation of a pulsed, unrefuelled reactor can be divided into five phases with the following general characteristics:

i) Setting up of plasma current, induced by the changing primary current. A proportion of the magnetic energy supplied is lost to the first wall due to poor energy confinement. With the present lack of understanding of this "startup loss" process the scaling is assumed to be the same as in the RFP[4], that is, the fraction of energy lost is taken to be proportional to current rise time and inversely proportional to first wall radius and plasma current. This phase is assumed to last one second.

ii) Heating, during which neutral injection is used to augment ohmic heating against the loss processes of bremsstrahlung, cyclotron and line radiation, until ignition is reached. This is in contrast to the RFP where ohmic heating alone is thought to be sufficient to overcome the loss processes. The contribution of nuclear reactions during this phase to the total thermonuclear power released during the cycle is negligible. The optimisation of the heating phase is discussed in section 2.2.

iii) Burn, when thermonuclear reactions occur and energy is released. During this phase the plasma temperature is assumed to remain constant. The plasma resistance is likely to be so low that the change in primary current to maintain plasma current is negligible. The recoverable energy per reaction is taken to be 20 MeV, corresponding to a blanket multiplication of 1.135. The phase terminates when the overall fractional burnup reaches a given value.

iv) Run down of plasma current, during which a changing primary current drives the plasma current to zero. Magnetic and kinetic energy inside the reactor vessel are assumed to be recoverered at 40% efficiency. This phase is assumed to last for one second.

v) Off time, the period during which the reactor vessel is evacuated to remove unreacted fuel, helium and impurities. New fuel is then supplied and the operating cycle repeated. Based on estimates of pumping requirements, a period of six seconds has been allowed.

2.2 Plasma Heating

There are several proposed methods for heating the plasma to ignition, but for the present study it was assumed that neutral beams would be used. A high beam energy would probably be necessary for plasma penetration, implying the use of negative ions during the acceleration process or positive ions with direct conversion of the unneutralized component of the beam. In the recirculating energy calculations an overall efficiency for plasma heating, η_I, of 40% has been assumed.

In an optimised reactor there is a preferred level of heating. The heating power is reduced by slow heating, reducing the cost of the heating equipment, but the cost per unit of reactor output is increased since the plasma burn becomes a smaller proportion of the cycle. Thus, at fixed reactor power and wall loading the total reactor cost can be minimised by varying the ratio of heating to cycle time. Such a variation is shown in Figure 1, which was based on calculations carried out on a simple zero dimensional model with high thermal conductivity within the plasma, low conduction losses to the wall and ignoring the effects of plasma impurities. The figure shows

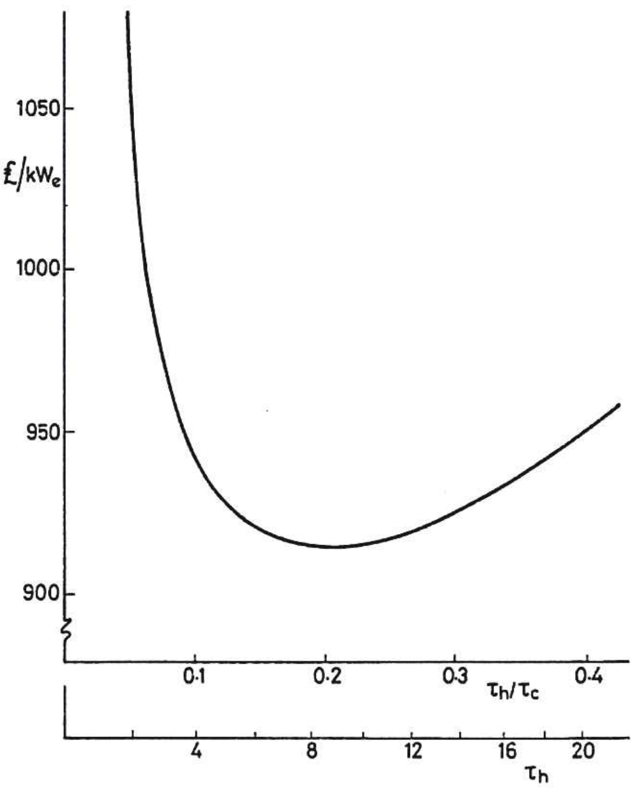

Fig. 1 Nominal reactor capital cost variation with the ratio of heating to cycle time at fixed reactor power of 600MW$_e$ and fixed mean neutron wall loading of 1·5MW/m^2.

that a ratio of about 0.2 is optimum. However, this calculation assumed a constant energy multiplication, which would not be present in a more detailed analysis. The energy multiplication is likely to decrease with larger ratios of heating to cycle time, because higher plasma densities and fields are required, and thus a lower value of 0.15 was chosen for the ratio. This was close to the value chosen for the RFP studies.

It should be noted that these calculations are zero-dimensional and thus neglect the spatial variations of plasma density and energy deposition from neutral beams. Much more extensive codes, which model several energy and particle loss processes, are available for determining the optimum injection current and energy. It is not clear that they give more accurate results than the simple model used here.

2.3 Plasma Equilibrium

A D-shaped plasma with elongation of 1.68 and D-shape parameter[8] of 0.17, similar to the design values of JET[9] and the Culham Mk II A and Mk II B reactors, was chosen as a balance between the conflicting intuitive requirements of increasing the plasma beta for the same investment in toroidal field, without increasing excessively the poloidal field energy required to maintain the plasma shape.

The value of poloidal beta, β_p, the ratio of plasma kinetic energy to poloidal field energy, (definitions of plasma parameters are given in Appendix A, to be discussed later) has been shown to be limited to less than the aspect ratio, A,[10] if the field separatrix is to lie outside the plasma. A further restriction to $\sim \sqrt{A}$ may be required if all toroidal current is to flow in the same direction[11]. On the other hand, if ballooning modes are considered alone, much higher values may be attainable. In the light of this uncertainty, values somewhat greater than $\sqrt{A}$ were considered here. The effect of possible restrictions on β_p are described later (§5). The value of current beta, β_I, defined by the modified Bennett relation[12] appears from numerical calculations described below to be quite close to that given by an analytic expression in terms of dimensions and β_p[11].

Average toroidal beta, $\overline{\beta}_\phi$, the ratio of plasma kinetic energy to toroidal field energy is often claimed to have a significant effect on reactor cost. Present day experiments[13] have achieved values $\sim$ 2% for this parameter and in JET it is hoped that values $\sim$ 5% might be achieved. Theoretical studies[14] indicate that somewhat higher values might be obtained and here 8% has been chosen to represent a reasonable compromise between optimism and pessimism. The value of $\overline{\beta}_\phi$ has been taken as fixed rather than the value of β, the ratio of mean plasma energy to total magnetic energy, or β^*, the ratio of root mean square plasma energy to toroidal field energy, which is directly related to reactor power, because it is the most often quoted.

The safety factors, q_o and q_a, measured at the plasma centre and edge respectively, required for stability must also be chosen. It has been shown[15] that for ideal internal modes $q_o > 1$ is a necessary condition for stability. A somewhat optimistic assumption made here is that the stabilisation of the m = 2 surface kink mode by specifying $q_a > 2$ is sufficient for reactor operation. This remains to be shown.

The plasma equilibrium current density profile has been specified here in terms of the plasma pressure and toroidal field profiles using a model due to Sykes, Wesson & Cox[14]. This model permits a wide range of β values to be studied. Equilibrium calculations were carried out inside a fixed boundary, of the required plasma shape, at which plasma current and pressure were set to zero. Such a boundary can be considered to be perfectly conducting and thus considerations of how to provide the maintaining field for the plasma equilibrium using external vertical field coils can be delayed until section 3.

For a given plasma aspect ratio, the relationship between the profiles of plasma density, current and the magnetic field in possible reactor plasmas can be derived. The reactor plasma dimensions can then be obtained by scaling of the profiles in the light of burn calculations, described below, and the engineering constraints described in section 3.

2.4 Plasma Burn

The assumed burn conditions are based on the conclusion of the RFP study in which it was shown that the recirculating power fraction could only be kept down to acceptable levels by assuming a control mechanism which allowed the burn to proceed at constant plasma temperature. This implies that the energy confinement time must be considerably less than the burn time, whereas the particle confinement time must be longer than the burn time. No known mechanism has been observed to date which would provide such control of burn temperature. Impurity injection after the plasma heating phase enhances radiation losses but cannot be considered since it does not give a thermally stable burn temperature. Possible mechanisms might be adiabatic plasma expansion and compression controlled by the poloidal field system during the burn, or through instabilities associated with operation at some critical β limit. If such mechanisms are practical there exists an optimum fractional burnup and burn temperature, which were taken to be 0.3 and 10 keV respectively from the RFP study.

The plasma temperature was assumed constant across the profile, with the density falling towards the edge of the plasma to match the required plasma pressure. As a result the highest rate of burn occurs in the central core of the plasma. For the profiles of plasma pressure obtained from typical Tokamak equilibrium calculations, the fall in fuel density during the burn is illustrated in Figure 2.

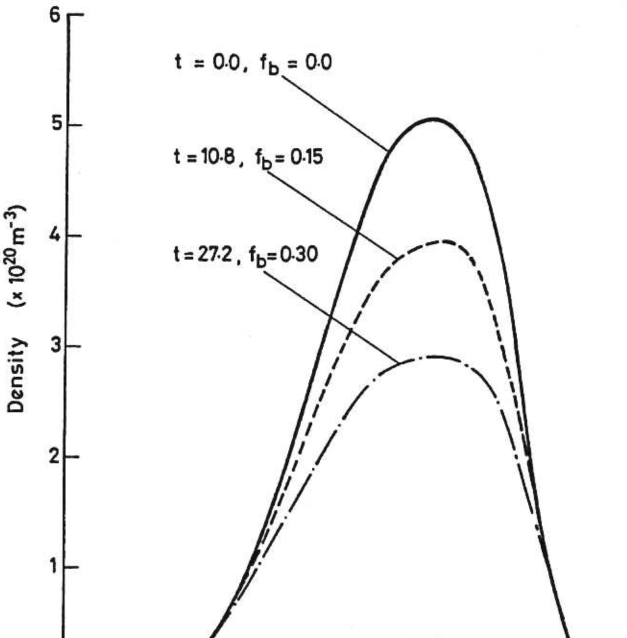

Fig. 2 Changes in the plasma density profile across the midplane as a function of position, time and fractional burnup.

3. ENGINEERING ASSUMPTIONS AND CONSTRAINTS

As stated in the introduction, one of the aims of this study is to derive parameters for a 600 MWe Tokamak reactor in order that realistic comparisons can be made with an RFP reactor under similar ground rules. The Culham Mk II A and II B Tokamak reactors were designed with higher output, but it is unclear whether the past trends in the growth of power system networks on which these power outputs were based will continue in the future. Also it is likely that when fusion power stations are first introduced smaller units will be required in order to test reliability and acceptability, and for these reasons a small unit size was proposed.

One of the consequences of the pulsed operation of a reactor is that low wall power loadings must be used. No divertor is present to protect the first wall from the energy deposition implied by the assumed temperature control mechanism, and the thermal fatigue problems caused by the cyclic nature of this energy deposition are made worse by the pulses of energy reaching the wall during the startup and run-down phases of the cycle. Thus a mean neutron wall loading of 1.5MW/m^2 has been assumed compared with 4.2 and 3.2MW/m^2 used in the previous quasi-steady state Culham reactor designs.

A consequence of fixing the power output and wall loading is that the product of plasma major radius (R) and minor radius (a) is fixed. However, for plasma stability the plasma aspect ratio (A=R/a) should be low. The net result is that the plasma inner major radius (i.e. R-a) must be minimised. Working against this fact is the need to include the necessary blanket, shield, ducting and clearances for remote maintenance, toroidal field coils and poloidal field coils, without exceeding the design limiting field of the coil systems. In order to take full account of these interacting constraints, reactor parameters have been calculated by using the methods described in Appendix A.

Under the assumptions of this section and section 2 reactor dimensions are found to be very similar to those of the Culham Mk II A and II B studies, and it has been assumed that many of the engineering construction and maintenance concepts of these designs can be applied to the pulsed Tokamak. Therefore this study has concentrated on the magnetic field design of the reactor, paying particular attention to poloidal field coil location, toroidal field coil shape and the losses occuring in the magnetic field systems as a consequence of pulsed operation. A further difference from the previous Culham Tokamak studies is that an air core transformer is proposed for the pulsed device. The engineering assumptions discussed below are summarised in Table II.

3.1 Toroidal Field

The Tokamak toroidal field is steady state. Its level in the plasma region is determined by the plasma density required to achieve the specified reactor output. The peak field at the toroidal field coil is determined by the separation between the coil and the plasma. For this study the distance between plasma and toroidal field coil was taken to be 2.1 m along the inside median

TABLE II

SUMMARY OF ENGINEERING ASSUMPTIONS			
Net reactor output power, P_{out}		600	MW_e
Mean neutron wall loading, $\bar{P}_n$		1.5	MW/m^2
Thermal efficiency, η_{TH}		0.4	
Blanket/shield/ducting thickness	inner	2.1	m
	outer	3.3	m
Shield door clearance		0.5	m
Refrigeration efficiency		0.35	%
Fraction of electrical power output to auxiliaries		0.05	
Magnetic field energy transfer efficiency		0.95	
Blanket replacement dose		4×10^{26}	n/m^2
Toroidal field winding		Nb_3Sn strip	
Stabiliser		Zr/Cu	
Winding current density		20	MA/m^2
Number of coils		20	
Toroidal field ripple		<2	%
Winding radial thickness		1.0	m
Poloidal field winding		NbTi filaments	
Stabiliser		Cu	
Current density within conduit		44	MA/m^2
Current per turn		30	kA
Transformer		air cored	
Separation of centre from toroidal field coil edge		0.5	m

and 3.8 m along the outside median. These are conservative estimates and are consistent with the values used in the superconducting RFP study[2] and take account of the absence of the stabilising shell in the Tokamak. They are the result of more detailed engineering and neutronics studies since the Culham Mk II A and II B designs which used 1.6 m along the inside median. No room is included for the primary or vertical field coil systems as these have been considered to lie outside the toroidal field coil system (see later). The peak toroidal flux density that could be supported was assumed to be about 8 Tesla.

Niobium-tin strip conductor[16] was proposed since it provides a simple winding construction which should be cheaper and stronger than a filamentary conductor at the same field. The coil consists of consecutive layers of superconductor, zirionium-copper for stabilisation, coolant and insulation, wound like a watch spring. Unlike filamentary superconductors, such a conductor is "self screening" so that the pulsed poloidal fields only cause currents to flow on the edges of the conductor. The pulsed field losses due to the changing plasma, primary and vertical field coil currents during startup and rundown are not excessive because they are also restricted to the edges. Previous loss calculations[17] indicate that this type of conductor is attractive, provided that the pulsed fields do not exceed about 0.7 T, which is the case in this design.

Twenty toroidal field coils are used, with the coil centreline conforming to the "Princeton-D" shape for discrete coils[18], a current density in the winding of 20 MA/m^2 and a coil radial thickness of 1 m. Care has been taken to ensure that toroidal field ripple at the outer edge of the plasma[19] is less than a nominal value of 2% (peak to peak) and that with constant cross-section coils, blanket segments can be removed between coils for replacement. A minimum all-round clearance of 0.5m for the removal of the outer shield door between adjacent toroidal field coils (i.e. 2 x 0.25m) has been assumed.

The refrigeration load for the toroidal field coil system includes contributions from pulsed field losses in the superconductor and stabiliser from both longitudinal and transverse pulsed poloidal fields[17]. An operating current density per unit width of 70 A/mm in the superconducting strip and a substrate resistivity of 2×10^{-9} Ωm have been assumed in calculating these losses. Heat losses of 5 W/MN are assumed in the mechanical supports, for twisting forces caused by the poloidal field, the centralising force on the toroidal field coil itself and the coil weight, and 4 W/kA in each of the two current leads per coil. Thermal radiation (0.7 W/m^2 of cryostat surface) through a cryostat 5 cm from the coil boundary, and nuclear heating add to the refrigeration requirements but helium cooling is provided by pool boiling and thus incurs no pumping loss. The heat deposited in the coil is averaged over the cycle and is removed by refrigeration consuming 290 W for every Watt of heat at 4 K.

3.2 Transformer Core Operation

A further dimension which determines the minimum plasma inner major radius is the radius of the central core measured to the outer edge of the primary winding. This in turn is specified by the operating sequence of the primary and vertical field coils and by the peak core flux density attainable (∿ 8T as for toroidal field coils). The use of an air cored transformer seems necessary, as preliminary calculations for an iron cored transformer with saturated central core and unsaturated limbs, as in JET, indicate that such a solution would be more expensive.

As with the RFP the storage of pulsed field energy can be minimised if a symmetric current swing is assumed in the poloidal field coil system. However, the vertical field system has to control plasma equilibrium during the heating phase and maintain plasma stability during the burn so it is reasonable to assume that it will have to be supplied by a separate energy storage and switching system to that used by the primary. Furthermore it will be necessary for there to be low flux linkage between the vertical field and primary systems so that the flux swing provided by the primary does not induce currents in the vertical field coils. On the other hand the vertical field coils can provide some flux linkage with the plasma, and reduce the primary flux swing requirement. Thus the most suitable mode of operation is for the net current in the vertical field coils to be zero and for the primary field in the plasma region to be negligible. Consequently the energy storage for the poloidal field system has been calculated by assuming a symmetric primary current swing, and this implies that primary current will be flowing, in one direction or the other, throughout the cycle.

3.3 Poloidal Field

Poloidal field coils have been placed outside the toroidal field coils for the following reasons:

i) To ease construction and maintenance - there is likely to be a considerable cost penalty associated with interlinked coil systems.

ii) Interlinking reduces the effectiveness of superconductor in both toroidal and poloidal field coil system since each system creates high field at the other, reducing the critical current and thus causing increased materials costs.

iii) For plasma and shielding of fixed dimensions, placing either the vertical or primary coil system inside the toroidal field coils decreases the toroidal field on the plasma axis for a given peak toroidal field and will therefore lower the attainable level of reactor power.

However, the use of external poloidal field coils results in a higher level of poloidal field energy, and this is considered in more detail in section 5.3.

The primary and vertical field coils have been considered to be centred on a toroidal surface 0.5 m outside the surface of revolution of the toroidal field coil exterior about the reactor centreline. The method for determining the continuous current distribution on this surface which satisfies the plasma operating conditions has been described elsewhere[20]. For primary windings creating negligible field inside the toroidal field coil and plasma regions it is found that most of the current is concentrated on the inner limb of the toroidal surface, with the current direction during the burn opposing that of the plasma. The vertical field coil current distribution is "quadrupole", with currents flowing with the plasma burn current above and below the plasma, and against it elsewhere. In practice these ideal current distributions must be replaced by discrete coils to permit access for pumping. In concentrating the current at specific locations care must be taken not to increase the level of pulsed field in the toroidal field coils due to the poloidal field to such an extent that pulsed field losses become intolerable. This applies particularly to the vertical field system, but the use of too few coils in the primary field system can also cause the "zero-internal field" requirement to break down in the toroidal field coil region. Furthermore, because of the choice of plasma shape, it is not possible to eliminate vertical field coils from certain regions. Twenty primary windings and fourteen vertical field coils were used in the reference reactor design. Studies indicated that this was a reasonable compromise between reduction of pulsed field losses while maintaining accessibility.

The refrigeration requirements of the poloidal field system can be estimated with similar assumptions to the toroidal field coils. The heat input to the coils consists of pulsed field losses, heat influx via supports and current leads, nuclear heating and thermal radiation. Strip conductors cannot be used for the poloidal field coils because of the high level of pulsed flux density ($\sim$ 8T) and thus the winding must be constructed from a transposed cable of superconducting multifilaments in a copper matrix[2]. Niobium-titanium was assumed here. The cable is force flow cooled by liquid helium inside an insulated stainless steel conduit 1.5 mm thick. Stainless steel reinforcement is provided between turns to support the different hoop stress in each coil. This design is the same as that proposed for the RFP with the conductor carrying a current of 30 kA at an overall current density inside the conduit (26 mm square) of 44 MA/m^2. The volume ratio of coolant to superconducting cable is assumed to be 1.2 and a total of 4 mm of insulation is provided between turns and between conductor and reinforcement. Losses due to the pulsed field consist of hysteresis loss in the superconductor, coupling loss between filaments and eddy current losses in the copper matrix, conduit and reinforcement. For a steady state toroidal field there is no longitudinal component of pulsed field and this helps to keep losses at a manageable level. For the helium coolant a pumping loss is incurred and this has been calculated for a conductor in which the temperature rise caused by helium pumping plus that caused by pulsed field losses is minimised. This results in a figure of 2.55 W/MA of poloidal field coil current per metre length of winding.

3.4 Energy Multiplication and Costs

The energy multiplication factor Q, defined for pulsed reactors as the ratio of thermal energy from the reactor to recirculating energy, is often taken to be the figure of merit in deciding the potential of particular reactors and this approach has been followed here. An alternative is to consider the recirculating power fraction $\varepsilon_c = 1/Q\eta_{TH}$.

The basis for the energy multiplication factor calculation is the energy flow diagram described in Appendix B. The flow of energy between the poloidal field coils and the energy transfer and storage system is characterised by a 95% transfer efficiency. Magnetic energy is lost in plasma startup and rundown and energy is used in neutral injection heating of the plasma and for refrigeration of the toroidal and poloidal field coil systems. In addition, reactor auxiliaries are assumed to consume 5% of the gross reactor electrical output. These components together make up the recirculating energy of the reactor. The efficiency of thermal conversion of the heat deposited in the reactor blanket and shield is assumed to be 40%.

The basis on which the capital costs have been estimated is the same as that used previously for the Culham Mk I and Mk II Tokamak reactor studies[21] and for the RFP studies[5]. The major reactor components such as the blanket, shield, windings, etc are costed according to their area or volume and the heat transport and generating equipment according to their thermal or electrical power ratings. An adjustment has been made to the toroidal field coil costs to reflect the use of simpler construction. The quoted costs are for a complete generating unit including reactor, boiler, turbine, generator, auxiliaries, and building, but excluding customers on-costs or interest during construction and assume quantity production for a series of commercial power stations. Replacement sections of breeding blanket are included, discounted to the date of construction and used to replace sections which have reached a total radiation dose of 4 x 10^{26} neutrons/m^2. This leads to blanket replacement about every five years throughout the 25 year reactor life. The cost data is based on 1976 values and has previously been used to estimate the capital costs of other Tokamak reactor designs[22]. Where possible the data has also been checked with figures used in a study of the cost optimisation of the JET experiment[9]. In order to facilitate direct comparison with previous studies, costs have not been updated.

4. REFERENCE REACTOR PARAMETERS

4.1 Choice of Parameters

The process of determining reactor parameters is iterative, beginning with an estimate of reactor dimensions from the method described in Appendix A and using the tightest plasma aspect ratio permitted by the engineering constraints in order to achieve a stable value of β approaching that required. In contrast to the recent RFP study, a full optimisation has not been undertaken, but earlier Tokamak studies have shown that the tightest aspect ratio produces the lowest reactor cost. The various reactor components, in particular the magnetic field systems, are then analysed in more detail, and the reactor dimensions are modified to satisfy the engineering constraints and the required power output.

Table III gives a list of reactor parameters derived in this way and Table IV tabulates the components of circulating energy in more detail. The main contributions to the circulating energy come from the losses during the transfer of poloidal field energy and the energy required for neutral injection heating, coil refrigeration and auxiliaries. Some improvement may be possible in the energy transfer efficiency but a detailed study of the behaviour of the switching and energy transfer system is beyond the scope of this report. Careful coil design and positioning may improve the refrigeration requirements. A more careful analysis of the requirements of reactor auxiliaries should also be considered for future study. It is interesting to note that in the present design the plasma magnetic and kinetic energy losses do not make a major contribution to circulating energy and this suggests that the overall efficiency of pulsed Tokamak reactors as envisaged at present depends to a greater extent on improvements in reactor technology, than in plasma physics.

TABLE III

600 MW_e REACTOR REFERENCE PARAMETERS		
Major radius, R	7.8	m
Minor radius, a	2.0	m
Safety factor at edge, q_a	2.2	
Poloidal beta, β_p	2.5	
Current beta, β_I	1.6	
Total beta, β	0.077	
Fusion power beta, β*	0.13	
Burn time, τ_b	27	s
Heating time, τ_h	6.0	s
Neutral injection power	86	MW
Cycle time, τ_c	41	s
Ratio of mean to peak power during cycle	0.43	
Plasma current, I_p	11	MA
Centreline vacuum toroidal flux density $B_{\phi o}$	3.9	T
Peak fuel line density	3.1×10^{21}	ions/m
Energy storage capacity	8.2	GJ
Gross thermal output	75	GJ
Circulating energy	5.7	GJ
Energy multiplication factor, Q	13	
Recirculating power fraction, ε	0.19	
Nominal capital cost per unit of generating capacity	960	£/kW_e

TABLE IV

COMPONENTS OF CIRCULATING ENERGY		
Primary energy transfer loss	0.55	GJ
Vertical field energy transfer loss	0.42	GJ
Energy lost in plasma startup	0.15	GJ
Energy lost in plasma rundown	0.12	GJ
Neutral injection energy	1.28	GJ
Refrigeration energy	1.64	GJ
Auxiliary energy	1.50	GJ

TABLE V

MAIN COMPONENTS OF REACTOR COST	%
Toroidal field coils	20
Poloidal field coils	11
First wall and blanket (with renewal)	6
Shield	8
Energy storage and transfer system	10
Neutral injection heating system	8
Conventional plant	26

Reactor costs can be allocated to the various reactor components and the main items are shown in Table V. The magnetic field coil systems and their associated power supplies and cooling represent approximately 40% of the total station cost. It is therefore highly important for the Tokamak, and the pulsed Tokamak in particular, that the magnetic field systems be studied in detail. As a first step in this process, the poloidal and toroidal field systems of this reference design will be described below. The coil layout is shown in Figure 3.

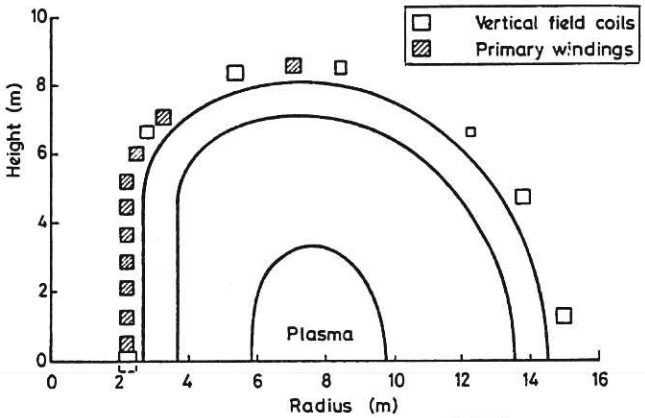

Fig.3 Position of toroidal and poloidal field coils in reference design.

4.2 Toroidal Field Coil Design

Details of the toroidal field coil design are summarised in Table VI. The coil dimensions are a result of the coil current density, number of coils, space for segment removal and segment dimensions. The low value of toroidal field ripple indicates that fewer toroidal field coils might have been used

TABLE VI

TOROIDAL FIELD COIL DESIGN		
Total ampere turns	153	MAT
Peak toroidal flux density	8.27	T
Major radius of inner limb centreline	3.2	m
Major radius of outer limb centreline	14.1	m
Major radius of peak height	7.4	m
Peak height of centreline	7.6	m
Height of straight section	4.7	m
Coil radial thickness	1.0	m
Coil toroidal thickness	0.53	m
Shield door clearance at height of 5.5 m	0.57	m
Toroidal field ripple	0.3	%
Number of turns per coil	257	
Amps/turn	29.7	kA
Superconductor thickness	8.8	μm
Insulation thickness	0.12	mm
Coolant channel thickness (2 per turn)	0.3	mm
Mean transverse pulsed flux density	0.48	T
Mean longitudinal pulsed flux density	0.56	T
Superconductor pulsed field loss	6.4	kW*
Matrix pulsed field loss	1.2	kW*
Support loss	46.1	kW*
Current lead loss	4.8	kW*
Nuclear heating	8.0	kW*
Thermal radiation	1.8	kW*

*average energy deposition rate at 4K over the cycle and there will obviously be room for some optimisation when physics tolerances become better known. The detailed coil structure implies some minor advances in the design of coolant channels and insulation beyond those presently under study.[17] The major sources of heat in the system are also shown. Approximately 65% of the support loss arises from the centralising force on the coils due to their own field. Toppling forces caused by the poloidal field coils contributes much of the remainder of the support loss. The heat leakage due to the centralising force can be removed, using the "self supporting arch" concept or it can be minimised by reducing the peak toroidal field. It is likely that only minor improvements in the toppling force support heat leakage can occur through careful coil positioning, particularly for vertical field coils outside the toroidal field coils, since all the return flux of the vertical field system must cross the toroidal field coils. This is discussed in more detail in section 5. Since support losses are the major heat source in the toroidal field coil system, more attention should be paid in future designs to minimising the heat leakage.

TABLE VII

POLOIDAL FIELD COIL DESIGN		
Peak primary current (symmetric swing)	67.8	MAT
Peak vertical field coil current (both directions)	21.8	MAT
Startup energy multiplier	1.76	
Inductive voltseconds required	177	Wb
Resistive voltseconds (rise time only)	28	Wb
Equilibrium voltseconds contribution	45	Wb
Core flux during off time	-88	Wb
Core flux during burn time	118	Wb
Core radius	2.2	m
Peak flux density in primary	7.8	T
Initial energy stored in primary W_{PR}	3.0	GJ
Transferred to homopolars during rise time, W_{R1}	2.7	GJ
Transferred from homopolars during rise time, W_{R2}	2.7	GJ
Transfer loss during rise time, W_{LR}	0.27	GJ
Transferred to homopolars during rundown, W_{Q1}	2.6	GJ
Transferred from homopolars during rundown, W_{Q2}	0.48	GJ
Transfer loss during rundown, W_{LQ}	0.16	GJ
Transferred to primary during off time	2.4	GJ
Transfer loss during off time, W_{LOFF}	0.12	GJ
Primary homopolar size, W_{STO}	2.9	GJ
Vertical field system homopolar size, W_{VF}	4.4	GJ
Vertical field system transfer loss, W_{LVF}	0.42	GJ
Total ampere metres	3.4	MAm
Filament loss	15.4	kW*
Coupling loss	7.6	kW*
Eddy current loss	13.0	kW*
Conduit loss	3.3	kW*
Reinforcement loss	2.6	kW*
Support loss	6.9	kW*
Current lead loss	7.9	kW*
Nuclear heating	2.0	kW*
Thermal radiation	1.2	kW*
Helium pumping	8.7	kW*

*average energy deposition rate at 4K over the cycle

4.3 Poloidal Field Coil Design

The major parameters of the primary and vertical field coil systems of the reference reactor design are shown in Table VII. As was stated in section 3, the primary winding operates with a symmetric swing of current over the plasma current rise time, in order to minimise energy storage. Furthermore Figure 3 shows that the primary is concentrated at low major radius, easing external access problems. The net vertical field coil current is zero and there is negligible mutual inductance between the primary and vertical field coil systems, thus permitting the two systems to be operated independently. The startup multiplier mentioned in section 2, is the factor by which plasma internal magnetic energy at the end of the current rise must be multiplied to indicate the energy required for current establishment. From this factor the resistive flux swing can be determined, and this together with the inductive flux swing and plasma inductance specify the peak core flux density, which in this design occurs during the plasma burn. The various energy flows to and from the vertical field and primary coil systems and their associated energy storage and transfer systems are also listed in the Table, and these can be more easily understood with reference to Appendix B and reference 5.

The dominant losses in the poloidal field coils occur in the superconducting filaments and the copper-stabiliser. Reduction in the core flux density would have a significant effect on pulsed field losses since most losses scale with $(\Delta B)^2$. Support losses are much lower than for the toroidal field coil system and helium pumping losses might also be reduced by a reduction in core flux density.

5. PARAMETER SENSITIVITY

Since the parameters derived in the previous chapter represent a point design which has not been fully optimised, it is not useful to carry out a full sensitivity analysis such as that reported for the RFP[5]. However, the effect of modifications to some of the basic assumptions of chapters 2 and 3 on the energy multiplication and cost of the reference design will be discussed, in order to determine the sensitivity of parameters to these choices.

5.1 Plasma Pressure Ratio

The reactor parameters which change with mean toroidal beta, β_ϕ between 5 and 10% and with poloidal beta, β_p between 2 and 3 are shown respectively in Tables VIII and IX. Plasma dimensions and the mean neutron wall loading have been kept fixed to maintain the gross reactor output at a constant level. Changes in the fusion reaction rate with the different density profiles associated with each beta value are accommodated by

TABLE VIII

TOROIDAL BETA MODIFICATION			
Toroidal beta, β_ϕ (%)	5.8	8.0	10.2
Poloidal beta, β_p	2.5	2.5	2.5
Plasma current (MA)	10.3	11.0	11.3
Peak toroidal flux density (T)	9.5	8.3	7.2
Safety factor at plasma edge	2.7	2.2	1.8
Vertical field energy (GJ)	4.1	4.2	4.3
$\Delta Q/Q$	-0.02		+0.02
$\Delta C/C$	+0.03		-0.03

TABLE IX

POLOIDAL BETA MODIFICATION			
Poloidal beta, β_p	2.0	2.5	2.9
Toroidal beta β_ϕ (%)	8.0	8.0	8.0
Plasma current (MA)	12.2	11.0	10.2
Peak poloidal flux density (T)	8.5	7.8	7.2
Primary current (MA)	75	68	63
Safety factor at plasma edge	1.9	2.2	2.4
Vertical field energy	4.8	4.2	3.9
$\Delta Q/Q$	-0.075		+0.038
$\Delta C/C$	+0.057		-0.030

extending or reducing the burn time while maintaining the same fractional burnup. The difference between burn and cycle time remains fixed at 14 seconds. The reference reactor design is shown in the centre column of each table. In Table VIII the poloidal beta is fixed, since it is considered to be a function of the plasma aspect ratio. Thus the only parameters that vary significantly (see Appendix A) are toroidal field and the safety factor at the plasma edge. For fixed power output and wall loading there is a small range of toroidal beta values that can be considered. Reduction much below a level of 5% increases peak toroidal flux density beyond 10 Tesla, and refrigeration requirements and increased coil forces raise costs and reduce energy multiplication. There is little opportunity for raising toroidal beta since the safety factor at the plasma edge soon falls below 2, making the plasma unstable to kink modes. These difficulties might be reduced if poloidal beta can be raised, that is, if its relationship with aspect ratio can be improved. The variations in Table IX were carried out at constant toroidal beta (i.e. constant toroidal field) and show that raising poloidal beta raises the safety factor at the plasma edge. This change also has the beneficial effect of lowering plasma current, vertical field energy, primary current, and peak poloidal field and hence increasing energy multiplication and reducing costs.

The energy multiplication factor can therefore be increased and reactor costs reduced by using plasmas with the highest possible values of poloidal beta and toroidal beta within the constraints of stability. However, the reference design has already assumed relatively optimistic values, and thus further significant improvements in energy multiplication and costs beyond the values quoted should not be expected.

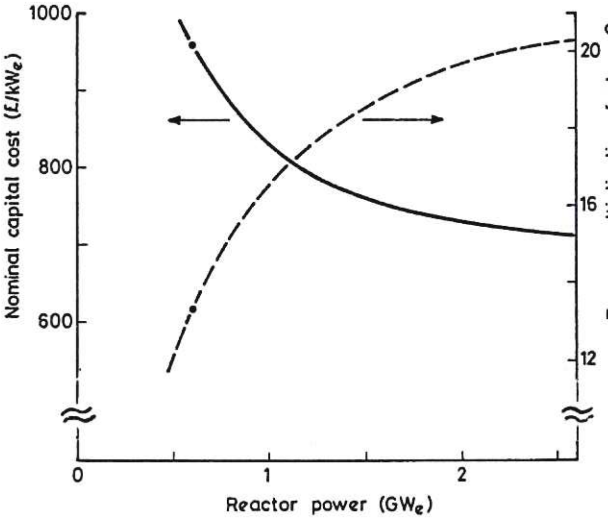

Fig.4 Variation of reactor nominal capital cost per unit output and energy multiplication factor with net reactor power.

5.2 Reactor Power

An arbitrary assumption made in this study has been to assume a reactor output of 600 MW_e. It is therefore useful to examine how reactor economics and efficiency change with the required reactor output.

The variation of capital costs and energy multiplication with reactor power output is shown in Figure 4. The plasma aspect ratio, the mean neutron wall loading and the various plasma pressure ratios have been kept constant and hence the reactor dimensions increase with power output, and the ratio of plasma surface to volume decreases. The burn time must therefore be extended and the plasma density scaled down to achieve the required wall loading, and so the levels of peak toroidal and poloidal field fall below the values assumed for the reference design. Consequently many of the energy and equipment requirements of the design do not increase linearly with the output power, and the energy multiplication and the cost per unit output improve accordingly.

The benefits of building power stations of larger output become less significant at high power ratings. With the assumptions used here there is a limit to the energy multiplication factor, since the circulating energy contains components proportional to reactor output, as well as components which are reduced with decreasing field strength. The capital cost per unit of output is also limited by costs which are proportional to power output.

5.3 Poloidal Field Coil Location

Several reasons for siting the poloidal (primary and vertical) field coils outside the toroidal field coil system were given in section 3.3, but the impact of this decision should be described in more detail.

The major parameters of three designs are shown in Table X. The first is the reference design, the second has vertical field coils only inside the toroidal field coil system and the third has vertical field and primary windings inside. To simplify the calculations the toroidal field coil system has been assumed to be fixed and thus the internal coil systems are using space which would more realistically be used by blanket, shielding and ducting. No account has been taken of the additional costs incurred by internal coils, both in increasing the difficulty of remote maintenance and in increasing the problems of shielding the coils from a heavier neutron flux. Also the reduction in the critical current density for

TABLE X

ALTERNATIVE POLOIDAL FIELD COIL LOCATIONS			
Position of primary relative to toroidal field coil	Outside	Outside	Inside
Position of VF coil relative to toroidal field coil	Outside	Inside	Inside
Primary current (MA)	68	73	23
VF coil current (MA)	±20	±11	±11
Peak primary energy (GJ)	3.0	3.5	1.1
VF energy (GJ)	4.2	1.7	1.7
Refrigeration requirements at 4K (kW/cycle):			
Toroidal field coils	68	59	62
Poloidal field coils	69	55	31
$\Delta Q/Q$		0.02	0.26
$\Delta C/C$		-0.06	-0.11

internal poloidal field coils, caused by the presence of the toroidal field, has been ignored. Thus the designs examined here with internal coils represent the best results that can be achieved. Plasma parameters and dimensions remain fixed throughout, so there are small variations in the net reactor output power due to changes in circulating energy.

When only the vertical field coils are situated inside the toroidal field system some small improvement in energy multiplication occurs. The vertical field energy is decreased because the coils are closer to the plasma and the pulsed field losses in the toroidal field coils are reduced because some of the vertical field flux can return inside the toroidal field coils. However, the primary current must be increased because the contribution of the vertical field coil system to the core flux swing is reduced. The reduction in the cost of vertical field coils and of poloidal field energy storage has a more significant effect on overall reactor cost.

Larger changes occur if the primary windings are also situated inside the toroidal field system. Since the mutual inductance of the primary with the plasma is increased, the primary field energy is significantly reduced. The primary flux crosses the toroidal field coil, increasing the pulsed field losses, but this is more than compensated for by the reduction in pulsed field losses in the poloidal field coils due to the reduced currents. However, these significant changes in losses and energies occur partly because of the assumption of fixing the toroidal field coil dimensions. In a more realistic comparison the increase in energy multiplication and reduction in cost would be less significant.

Thus, a major conclusion of this study is that the reduction in energy multiplication and increased costs of siting poloidal field coils outside the toroidal field coils appear to be tolerable unless the cost of constructing, maintaining and shielding internal coils can be reduced below ∿ 10% of the overall reactor cost. Furthermore there is little increase in energy multiplication if only the vertical field coils are internally sited.

If water cooled copper coils were to be used inside the toroidal field coil system, reactor output would be required to overcome the resistive losses. In the designs considered here this loss would amount to $60MW_e$ for the vertical field coil system and $40MW_e$ for the primary coil system, even if sufficient room existed to permit coils with as low an overall current density as $2.5MW/m^2$. This would result in a reduction of energy multiplication to $Q = 9.3$ and $Q = 7.8$ respectively at the reduced power level. Copper poloidal field coils are therefore not a realistic option for reactors.

5.4 Access Between Toroidal Field Coils

The toroidal field coil dimensions of the reference design were chosen to allow segments consisting of blanket, shield and ducting to be removed between coils, with a 0.5 m clearance for the shield door. While the space on the reactor midplane allowed for the inboard portion of a segment has been minimised, the space allowed for the outboard portion depends on a detailed engineering design of the segment itself and segment removal and maintenance and in particular on the shield door clearance. With the "Princeton-D" model for the toroidal field coil shape, any increase in coil height required for segment removal and maintenance will result in an increase in this "outboard space". The effect of changing the shield door clearance is shown in Table XI. A

TABLE XI

ACCESS BETWEEN TOROIDAL FIELD COILS				
Outboard space	3.3	3.8	4.3	m
Door clearance	0.34	0.57	0.79	m
Vertical field coil current	±20.0	±21.8	±23.5	MA
Vertical field energy	3.51	4.21	5.02	GJ
Toroidal field ripple	0.55	0.32	0.19	%
$\Delta Q/Q$	0.015		-0.017	
$\Delta C/C$	-0.016		0.019	

40% increase in the vertical field energy with a 0.5 m increase in door clearance (i.e. a 1 m increase in the outboard space) causes a 3-4% decrease in the energy multiplication factor and increase in cost.

In the above analysis 20 toroidal field coils have been used. However, in the design shown in column 1 of the table the toroidal field ripple is still low, suggesting that fewer toroidal field coils could be used. This would have the beneficial effect of increasing the intercoil clearance for the shield door but would also increase the door and segment weight. Calculations indicate that for 14 coils the door clearance can be raised above 0.5 m with a resulting toroidal field ripple of 3.5%. A better estimate of the tolerable level of toroidal field ripple is required before the outboard space and door clearance can be minimised.

6. PARAMETER COMPARISON WITH THE REVERSED FIELD PINCH REACTOR

At this stage in the development of the pulsed Tokamak and the Reversed Field Pinch reactor, comparisons can only be tentative since detailed designs of reactor components have not yet been made and costing is at a very preliminary level. Furthermore, the physics assumptions of the two designs are quite different, and experimental

TABLE XII

COMPARISON OF REACTOR PARAMETERS	Pulsed Tokamak	Supercon. RFPR
Major radius (m)	7.8	14.5
Minor radius (m)	2.0	1.5
Net power output (MW_e)	600	600
Gross thermal output (MW_{th})	1825	1900
Burn time (s)	27	25
Heating time (s)	6	4.5
Cycle time (s)	41	37
Plasma current (MA)	11	17
Primary current	68	28
Vacuum vertical field on axis (T)	0.54	0.35
Toroidal magnetic field on axis (T)	3.9	3.8
Toroidal magnetic field at coil (T)	8.3	-1.0
Number of toroidal field coils	20	28
Toroidal beta	0.08	0.50
Poloidal beta	2.5	0.31
Total beta	0.08	0.19
Startup and rundown energy loss (GJ)	0.27	2.92
Energy storage requirement (GJ)	8.2	8.5
Energy multiplication	13.3	11.8
Capital cost (£/KW_e)	960	750

confirmation of Tokamak physics has reached a more advanced state than that of the RFP. However, it is possible to reach some general conclusions, and these are listed below. A comparative table of reactor parameters is shown in Table XII.

The Tokamak requires a tight aspect ratio for stability and this has been achieved here by using an elongated plasma cross section and a larger minor radius than the RFP. Reversed Field Pinch physics places little restriction on aspect ratio, but a small minor radius should permit rapid ohmic heating to ignition. In the Tokamak ohmic heating is insufficient for ignition and supplimentary heating has been provided here in the form of neutral injection.

From the point of view of reactor maintenance the RFP suffers from the presence of a copper stabilising shell immediately behind the first wall. The difficulties in maintaining such a shell are not yet fully understood and its presence was ignored in the overall capital cost. It has been estimated[4] that the minimum continuous length of shell must be four times the plasma minor radius, and this fixes the minimum segment size that would be removed. As in the Tokamak, the number of toroidal field coils is specified by the tolerable level of field ripple at the outer plasma edge, and these two facts mean that the toroidal field coils of the RFP must be moved aside for segment removal. The tight aspect ratio of the Tokamak helps in this respect, producing a more open structure allowing segment replacement between static toroidal field coils.

The RFPR plasma current is $\sim$ 50% higher than that in the pulsed Tokamak. However this current is produced by a lower level of primary current than in the pulsed Tokamak because the mutual inductance of primary and plasma is much greater for the RFPR. The vertical field that must be provided for the non-circular Tokamak is also higher than for the circular RFP plasma. The Tokamak requires a central toroidal field value similar to the RFPR. In the Tokamak this must be supplied by the toroidal field coil system, resulting in high field on the internal limb of the coil. For the RFP, toroidal field energy is provided through the poloidal field coil system in the process of self-reversal of the toroidal field at the plasma edge, and only this low field has to be maintained by the external toroidal field coils. Thus the peak field at the toroidal field coil is much lower in the RFP. The net effect of these differences is that even with the simpler toroidal field coil construction in the Tokamak, the magnetic containment system for the RFP is cheaper. This is a direct consequence of the different value of total beta attainable.

The energy multiplication of the two designs seem to be very similar in spite of the differences mentioned above, with less than 20% difference between the circulating power in the two systems. In the RFP no external energy is used to heat the plasma, but this is more than compensated for by the increased plasma losses of magnetic and kinetic energy, in startup and rundown of the plasma current. These losses are more than ten times larger in the RFP than in the Tokamak, but again as physics understanding of these processes is gained, the difference may be reduced. For both designs, the use of copper coils is undesirable, producing high levels of circulating power.

The apparent difference in cost of the two devices may not be very significant in the light of the physics uncertainties, particularly of the RFP. However, not only are the coil systems of the Tokamak more expensive than those of the RFP, but also the neutral injection requirement of the Tokamak places a further burden on costs. There is little difference in the energy storage costs of the two devices, the requirements for pulsed poloidal and toroidal field supplies in the RFP balancing the requirement for pulsed poloidal field and neutral injection supplies in the Tokamak.

For the Tokamak, power increase at constant aspect ratio leads to a reduction in cost per unit output. For the RFP, the requirement of ohmic heating to ignition means that plasma aspect ratio must be increased with output with the result that costs per unit output fall more slowly.

In summary, at this stage in the evolution of both pulsed Tokamak and Reversed Field Pinch reactors it seems that the higher magnetic fields and the neutral injection requirements of the Tokamak are always likely to make it more expensive than the RFPR. However, the energy multiplication factor will generally be higher in the Tokamak due to the steady state nature of the toroidal field.

7. CONCLUSIONS

Design parameters have been deduced for a 600 MW_e pulsed Tokamak reactor. With the assumptions and constraints taken for this study, the energy multiplication factor Q is 13 and the circulating energy fraction 19% with a thermal conversion efficiency of 40%. The energy multiplication factor can be increased and the cost reduced by increasing the reactor output.

For a Tokamak with fixed power output and mean wall loading the physics and engineering constraints confine poloidal and toroidal beta to small ranges. The major limitations are the maximum toroidal magnetic field which can be supported by the reactor structure, the minimum safety factor q_a at the plasma edge, and the assumption that poloidal beta cannot exceed some power of the plasma aspect ratio. Operation at the maximum poloidal and toroidal beta minimises the reactor cost and maximises the energy multiplication factor, but the values used in this study are thought to be close to the realistic limits.

The major components of recirculating energy are the energy required to heat the plasma to ignition by neutral beam heating and the refrigeration to maintain the temperature of the superconducting magnets. Thus the energy multiplication factor is constrained as much by technological limitations as by necessary improvements in plasma physics.

The siting of poloidal field coils outside the toroidal field coils increases the primary current required to induce the plasma current and the currents in the vertical field windings which maintain equilibrium, but does not have too serious an effect on the energy multiplication factor or costs. Internal vertical field coils alone offer little improvement. The apparent benefit of having all poloidal field windings inside the toroidal field windings is not thought to outweigh the increased maintenance problems associated with interlinked windings.

The overall capital cost and energy multiplication factor of the pulsed Tokamak reactor have been found to be similar to those of a superconducting Reversed Field Pinch reactor with the same net output when similar assumptions have been taken. The magnetic field system of the Tokamak is more expensive than in the RFP and the neutral beam injection system for plasma heating is an additional cost. Against this must be balanced the additional problems introduced in the RFP by the need for a stabilising shell close to the first wall. It must be remembered, however, that the level of understanding of the physics of the two systems is not the same, so that detailed comparisons are premature and a continuous assessment of their relative merits will be necessary as further development proceeds.

Acknowledgements

This study has been built upon the knowledge gained in previous reactor design studies of the Reversed Field Pinch and Culham Mk II A and Mk II B Tokamaks.

The authors would therefore like to thank their colleagues at Culham Laboratory, Rutherford Laboratory, Harwell and the Universities of Padua, Calabria and Naples who have contributed to these studies. In particular for the work reported here they would like to thank C. Walters for his calculations on heat generation in superconductors in the presence of pulsed fields, and A. Sykes, J. Wesson, M. Turner and C. Thomas for the use of their equilibrium codes.

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APPENDIX A

Choice of Parameters for a Reactor of Fixed Output

The mean neutron wall loading during the cycle $\overline{P}_n$ can be written in terms of the net reactor output P_{out} as;

$$\overline{P}_n = \frac{P_{out}}{\eta_{TH}(1-\varepsilon_c)} \frac{Q_n}{Q_R} \frac{1}{S} \tag{A1}$$

where $S = 4\pi^2 Ra\sqrt{(1+\varepsilon^2)/2}$ is the wall area, η_{TH} is the thermal conversion efficiency, ε_c is the circulating power fraction, Q_n is the neutron energy/reaction, Q_R is the fusion output/reaction, R is the plasma major radius, a the plasma minor radius and ε the plasma vertical elongation. From the definitions of β_p, $\overline{\beta_\phi}$ and q_a, viz.,

$$\beta_p = \left(\frac{R}{a}\right)^x = \frac{2\mu_o \int p\, dV}{\int B_\theta^2\, dV} \tag{A2}$$

$$\overline{\beta_\phi} = \frac{2\mu_o \int p\, dV}{\int B_\phi^2\, dV} \tag{A3}$$

$$q_a = \frac{a}{R}\sqrt{\frac{1+\varepsilon^2}{2}}\;\frac{B_\phi}{B_\theta}\; f \tag{A4}$$

where $x \sim 0.5 - 1.0$ and $f \simeq 1.1$, the following equation can be obtained: (A5)

$$R^{2(2-x)} = \left[\frac{P_{out}}{\overline{P}_n}\frac{Q_n}{Q_R}\frac{1}{2\sqrt{2}\,\pi^2\eta_{TH}\,(1-\varepsilon_c)}\right]^{(2-x)} \cdot \frac{(1+\varepsilon^2)^{x/2}}{2\overline{\beta}_\phi q_a^2} f^2$$

and equation A1 can be rearranged to read:

$$Ra = \frac{P_{out}}{\overline{P}_n}\frac{Q_n}{Q_R}\;\frac{1}{2\sqrt{2}\,\pi^2\eta_{TH}\,(1-\varepsilon_c)(1+\varepsilon^2)^{1/2}} \tag{A6}$$

Equation (A5) and (A6) can now be used to determine R and a respectively for fixed values of $\overline{\beta}_\phi$, q_a, ε, x, P_{out} and assumed values of $\overline{P}_n$, Q_n, Q_R, η_{TH}, ε_c and f. However, the level of toroidal field in the plasma must now be calculated and the value of toroidal field at the TF coil determined in order to check whether such dimensions are reasonable from an engineering standpoint.

From the definition of power output and $\beta^* = 2\mu_o(\int p^2 dV/\int dV)^{1/2}/B_{\phi o}^2$:

$$\beta^{*2} B_{\phi o}^{\;4} = \frac{P_{out}\; 64\,\mu_o^{\,2}(kT)^2}{Ra^2\; 2\pi^2\varepsilon\; \langle\sigma v\rangle\; Q_R\; \eta_{TH}\;(1-\varepsilon_c)\; G_b} \tag{A7}$$

where $\langle\sigma v\rangle$ is the reaction rate parameter (which has been derived at constant temperature but would more realistically be a volume averaged value) and G_b is the ratio of peak to mean power during the cycle. A rough rule of thumb is that $\beta^* \sim 3\bar{\beta}_\phi/2$, which enables an estimate of $B_{\phi o}$ to be made. For typical burn, heating and off times, $G_b \sim 0.4$. Then, if Δ_{bs} is the blanket and shield thickness on the median plane between plasma and reactor centreline, the peak field is given by;

$$B_{\phi max} = \frac{R\ B_{\phi o}}{R - a - \Delta_{bs}} \tag{A8}$$

The peak toroidal field value may place a constraint on the permissible range of values for $\bar{\beta}_\phi$, q_a, ε, x, and P_{out}, but another possible engineering constraint is on the peak core field. A first estimate of this value can be determined from the plasma current i.e.

$$I_p = \left(\frac{2\pi}{\mu_o}\right) \left(\frac{a^2}{R}\right) \left(\frac{B_{\phi o}}{q_a}\right) \left(\frac{1 + \varepsilon^2}{2}\right) f^2 \tag{A9}$$

and the plasma "external" inductance:

$$L_p = \mu_o R \left[\ln\left(\frac{8R}{a\sqrt{\varepsilon}}\right) - 1.75\right] \tag{A10}$$

by assuming that;

$$B_{cmax} = \frac{L_p\ I_p}{2\pi\ R_c^2} \tag{A11}$$

where R_c is the core radius.

Plasma physics places constraints on the obtainable values of aspect ratio, R/a and elongation ε, but the permissible ranges of these parameters are not at present well defined. However it is felt that low aspect ratio devices are more likely to be stable at higher values of $\bar{\beta}_\phi$ and thus reactor design should aim to maximise B_{cmax} and $B_{\phi max}$ to minimise aspect ratio. In order to do this here it has been the policy to specify $\bar{\beta}_\phi$, q_a, ε and P_{out} while permitting x to vary.

APPENDIX B

Energy Flow Diagram

The basis for the energy multiplication factor calculation is the energy flow diagram shown in Figure B1. At the beginning of the cycle, energy is exchanged between the primary coil system and homopolar generators in order to provide the flux swing in the central core to drive the plasma current (W_{R1} and W_{R2}). This energy must include a fraction that is assumed to be lost in startup (W_{st}) in addition to the energy required in the primary field during the burn.

After initial establishment of plasma current, vertical field coils must control plasma shape and position and this requires energy (W_{vf}). It has been assumed that no energy is transferred from the vertical field system either to the plasma or primary field system and hence this energy is recoverable after the current rundown phase. The plasma current must be driven to zero at the end of the burn and this is achieved by an energy exchange between primary windings and homopolars (W_{Q1} and W_{Q2}). A fraction W_{INT} of the plasma internal energy remains unavailable for direct recovery and is removed as heat. The off time is used to replenish the primary coil system ready for the cycle to restart. In order to heat the plasma to the required temperature at the start of the burn neutral injection has been employed (W_{NI}) and this is the major contribution to plasma kinetic energy. Once burn temperature is reached, thermonuclear reactions in the plasma produce 14.1 MeV neutrons which are multiplied in the blanket to produce an additional 2.4 MeV in reactions with solid lithium. An as yet unknown loss mechanism is assumed to transfer most of the alpha-particle energy(3.5 MeV) to the first wall during the burn. All the energy produced by neutrons inside the plasma, blanket and shield by the means described above is removed as high grade heat by helium coolant and exchanged with standard steam-raising plant for conversion to electricity with an efficiency of $\eta_{th} = 0.4$. A fraction (5%) of this electrical output (W_E) must be used to supply auxiliaries, and further components of recirculating energy (W_c) include refrigeration power for the coil systems (W_{REF}) and losses (W_{LR}, W_{LQ}, W_{LVF}, W_{LOFF}) in energy transfer (95% each way), current startup (W_{ST}), rundown (W_{INT}) and in neutral injection (W_{LNI}).

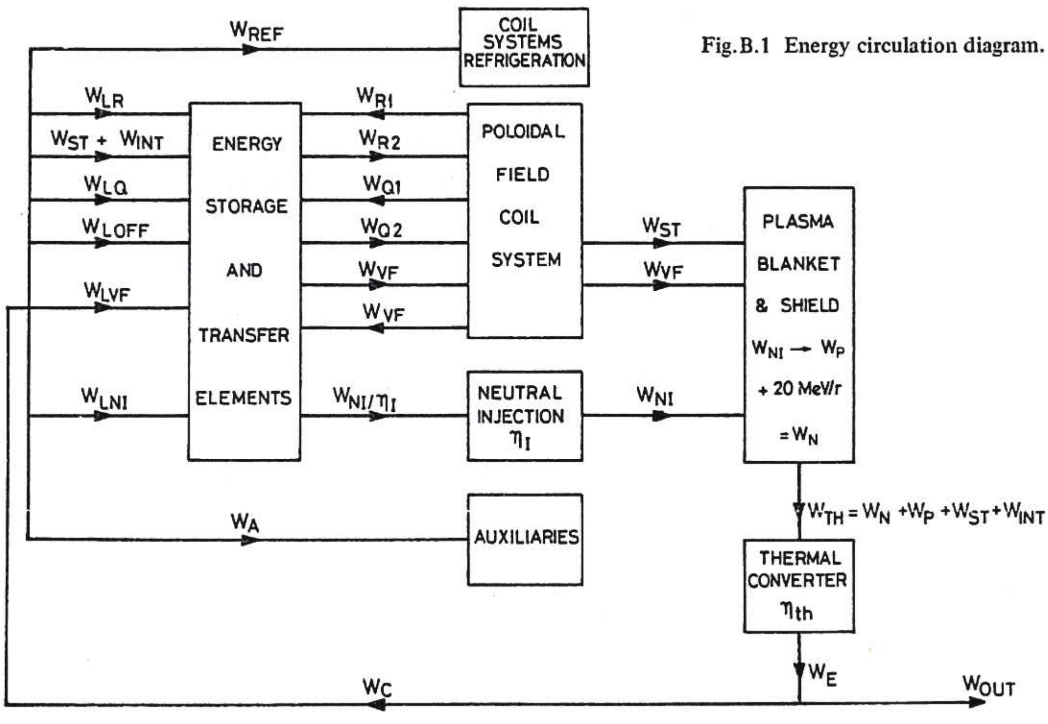

Fig.B.1 Energy circulation diagram.

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