PHILOSOPHICAL TRANSACTIONS OF THE ROYAL SOCIETY A
MATHEMATICAL, PHYSICAL & ENGINEERING SCIENCES

Concepts and Designs of D-T Fusion Electricity Generating Plant

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Concepts and designs of D–T fusion electricity generating plant

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Although the plasma temperature and confinement time required for a fusion reactor have not yet been simultaneously obtained in laboratory experiments, current progress in fusion research encourages the expectation that they will be achieved during the coming decade. It is, therefore, now desirable to develop conceptual designs of deuterium–tritium fuelled commercial fusion reactors for the generation of electricity. The additional physical and technological developments required for the construction of fusion reactors are reviewed and two conceptual designs are described. Based on such designs, the anticipated advantages and costs of fusion can be assessed, and possible time-scales for the development of fusion power proposed.

1. Introduction

Substantial progress has been achieved recently in containing and heating hydrogen plasmas with the ultimate objective of obtaining conditions under which thermonuclear reactions in a deuterium–tritium plasma would give useful nuclear power. This progress has been sufficient to justify the approval of large experiments such as JET, T.F.T.R. and J.T.-60 which will become operational in 1982–4 and are expected to contain plasmas with temperatures of 10 keV and energy confinement times approaching 1 s. These parameters will be close to those required for a self-sustained reaction, so it is reasonable to expect that the following generation of fusion devices will be experimental reactors in which the emphasis in design and operation will be mostly on engineering aspects rather than plasma physics. Preliminary studies of such devices have already begun.

It is thus important to study the many technologies required in a fusion reactor and to undertake conceptual designs of possible commercial fusion reactors. Such conceptual designs allow the form of a reactor to be visualized, the interaction of physical and technological constraints to be examined, the relative merits of alternative plasma confinement geometries to be compared and the anticipated advantages of fusion to be evaluated. These studies can, however, only lead to preliminary conclusions, since the plasma physics extrapolations on which they are based are not fully understood, and the necessary engineering developments are not complete.

The main technological features of a fusion reactor are discussed in §2, and their integration in a reactor is illustrated in §3 by describing two conceptual designs. These sections deal primarily with toroidal magnetic confinement systems to limit the range of discussion. Fusion research within the U.K.A.E.A. is limited to these systems, and the tokamak has for several years been the system closest to obtaining the plasma parameters required for a reactor. For completeness, reference is also made in §3 to parallel studies in open-ended magnetic confinement systems and inertial confinement systems. Some conclusions from conceptual reactor studies are given in §4, and possible time-scales for the development of fusion reactors are proposed in §5.

2. Reactor requirements

Current plasma physics experiments are undertaken with hydrogen plasmas. The use in reactors of deuterium–tritium plasmas, and the resultant production of fusion neutrons and considerable thermal power, will introduce new engineering requirements; the main areas are reviewed in the following section. The additional plasma physics requirements are also summarized.

2.1. *Physics*

The temperature of a reacting deuterium–tritium plasma will be maintained by the high energy alpha particles formed in the fusion reactions. Since the reaction rate increases rapidly with temperature whereas the energy losses only increase slowly, the plasma temperature will be unstable in the range 10–30 keV preferred for a reactor, and a means of stabilizing the temperature is required. Possible control mechanisms include adiabatic expansion and compression of the plasma about the ignition point, increased particle losses by ripple diffusion, or increased cooling of the plasma by the recycling of particles at the wall or a divertor. The effectiveness of these approaches has only been simulated, however, in simplified computer calculations based on current understanding of particle and energy confinement and remains to be demonstrated in future experiments.

To sustain a reacting plasma it must be supplied with new fuel, and reaction products and impurities must be removed. Refuelling may be achievable by recycling cold gas from the wall, or by the fast injection of frozen pellets of deuterium and tritium. Exhaust of reaction products and impurities will probably require a magnetic divertor into which plasma escaping from the hot reacting core is guided before it reaches the reactor wall. Various magnetic configurations have been considered and are under investigation in the current generation of experiments. If either refuelling the plasma or exhausting the impurities appears impossible, it would be possible to operate the reactor in a pulsed mode in which the vessel is evacuated and refilled with new fuel between burns of 20–40 s.

A final physical requirement for any reactor based on magnetic confinement is that the plasma be maintained at a sufficiently high value of the parameter β, the ratio between the plasma pressure and the pressure of the confining magnetic field. This requirement is related to the economic production of power, and implies an increase in β beyond the levels obtained in current tokamak experiments.

2.2. *Tritium breeding*

Tritium has a half-life of 12.6 years and therefore is not abundant in nature, but must be produced within the reactor. This is achieved by the use of a breeding blanket surrounding the plasma, which contains lithium and captures the fusion neutrons. The lithium may be in many forms – liquid lithium metal, molten salts, or solids such as lithium oxide or lithium aluminate, and is used in combination with a neutron reflector such as graphite. The characteristic thickness of the blanket is about 0.5 m.

Simple neutronic calculations show that adequate breeding ratios can be obtained in a range of designs for blankets with natural lithium and that designs of blankets with a low lithium content are feasible if lithium enriched with ^{6}Li is used. More complex calculations take account of the heterogeneous nature of the breeding blanket and of the need for holes, but still show sufficient breeding in most cases. Uncertainties exist owing to a lack of nuclear data for some materials, especially for neutrons in the energy range 2–14 MeV.

The choice of breeding material depends not only on its neutronic performance, but also on several engineering and economic factors such as the operating temperature and compatibility with the chosen coolant and structural materials. Another important aspect is the recovery of the tritium from the breeding material and its safe containment within the reactor. Liquid lithium, which provides one of the highest breeding ratios, has a high affinity for tritium so that recovery is difficult and a high inventory (*ca.* 10^8 Ci) remains in the reactor. Solid breeders allow much lower inventories (*ca.* 10^6 Ci) to be achieved, but often give marginal breeding ratios.

2.3. *Blanket structure and heat transfer*

Three-quarters of the energy released in the reacting plasma is associated with the neutrons, and is deposited in the blanket as heat. Thus the blanket structure must contain both the breeding material and the blanket coolant as well as acting as a vacuum boundary between the plasma region and the blanket. Structural material in the first wall region, closest to the plasma, is highly stressed because economic power generation requires a high power density, but it also suffers the greatest radiation damage since the neutron flux from the reacting plasma is high. The blanket region is not easily accessible for maintenance, and therefore the blanket structure must be very reliable.

If liquid breeding materials are used, the breeder may also be used as the reactor coolant. Liquid lithium has properties similar to liquid sodium and may be used, as in the fast breeder reactor, with output temperatures around 550 °C. The presence of a high magnetic field, however, leads to magnetohydrodynamic pumping losses and high coolant pressures. For solid breeders, helium is the preferred coolant although high pressure water has been considered. Pressures of 5–10 MPa are required in both cases. Heat fluxes are generally lower than in fission reactors, except at the first wall, where direct radiation from the plasma is absorbed.

Radiation damage of the structural materials is more serious than in fission reactors owing to the higher neutron energies and the resultant enhanced rates of transmutation reactions leading to the formation of hydrogen and helium in the structure. Void formation can occur as in fast breeder reactors, and at higher temperatures helium bubbles form. Many envisaged fusion reactors will operate in a cyclic rather than a truly steady-state manner and thus fatigue and crack growth will be more prevalent, especially for magnetic systems operating in the pulsed mode or inertial confinement systems, and wall power loadings must be reduced accordingly. The development of structural materials is one of the most serious technological problems in the development of fusion reactors.

2.4. *Magnetic field system*

Essential components of fusion reactors based on magnetic confinement are the windings that create the necessary magnetic fields. These must be superconducting since the ohmic dissipation in copper conductors would be comparable with the reactor output power. Major problems are the forces involved in large and complex windings, the need in many systems of pulsed magnetic fields, and the protection of superconducting systems from unexpected transitions to the normal state. The windings must be shielded from the blanket by a shield which reduces the neutron flux by a factor of 10^6 and is typically 1.0 m thick.

Two superconducting materials are commonly considered for fusion magnets, the alloy niobium–titanium and the compound niobium–tin. The former is limited to magnetic fields below 8 T but has the advantage that it is ductile and may be coprocessed with copper to form

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a stabilized conductor. Niobium–tin can be used at higher magnetic fields, but it is brittle and must be formed by the reaction of its two constituents after processing. In both cases it is necessary to use the superconductor in the form of many filaments in a matrix of a normal conductor, such as copper, which provides stabilization against magnetic instabilities in the superconductor and an alternative conducting path if a transition to the normal state occurs. For conductors subjected to pulsed magnetic fields, finer filaments will be needed to withstand magnetic instabilities and therefore the conductor is more expensive. In all cases cooling by liquid helium is essential, and forced-flow cooling is preferred.

A substantial structure is required to withstand both the tensile stresses in each winding and the forces due to mutual interactions between coils. The bulk structure required to withstand these forces, particularly in faults, generally determines the limit to the maximum magnetic field that may be used rather than the intrinsic properties of the superconductors.

2.5. *Reactor maintenance*

Two important differences between current plasma physics experiments and fusion reactors are that the reactor must operate reliably with high availability for long periods, and that its structure becomes radioactive, so maintenance operations must be undertaken by remote means. The development of reactor concepts that allow simple and rapid access of remotely operated machines for maintenance or repair is essential. In particular, since it is not expected that the first wall or blanket structural material will have a life comparable with that of the rest of the reactor, it is necessary to plan means of obtaining access to the centre of the reactor to replace the blanket quickly as a routine maintenance operation.

Three general approaches to maintenance of the blanket structure have been considered. The use of internal machines which replace small units sequentially, the removal of larger units of blanket and shielding between fixed magnetic windings, and the removal of complete segments of a reactor, including blanket, shield and magnets. The first of these is relatively slow and cannot deal with unforeseen failures in ducts behind the blanket; the third results in the unnecessary movement of extremely large masses and the dismantling of the magnet structure which is normally maintained at cryogenic temperatures. The choice will also be affected by the location of the primary vacuum vessel, and this is still a question to which there is no agreed answer.

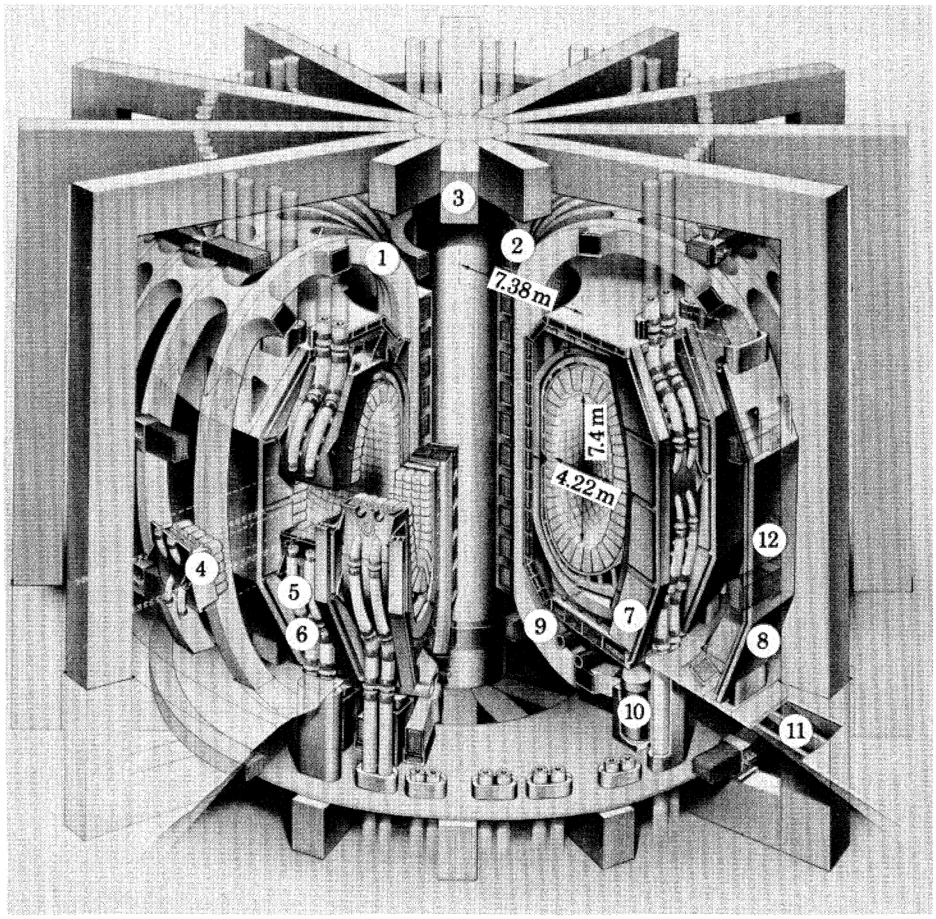
Whichever maintenance system is preferred it is necessary to demonstrate that it is possible to break, replace and test coolant lines and vacuum joints by remotely operated machines in reasonable times and with a high degree of reliability. The problem is considerably more complex than in fission reactors because of the geometry, the interaction of vacuum and high pressure coolant requirements, and the need to limit tritium leakage to a very low level.

3. Reactor designs

Many conceptual designs have been undertaken, covering most plasma confinement geometries and a wide variety of approaches. In many cases the studies have concentrated on particular aspects, such as the blanket or magnet design, but several have been sufficiently complete to allow preliminary cost or safety assessments to be made. Two designs undertaken at Culham are described below, and both are toroidal magnetic systems.

3.1. *Tokamak reactor*

The Culham Conceptual Tokamak Reactor Mk II A (Hancox & Mitchell 1976) shown in figure 1 was designed to give a net electrical output of 2500 MW. The geometric form was chosen to be similar to the JET experiment, with the objective of investigating the implications of the choices of a shaped minor cross section and tight aspect ratio. It was assumed that suitable mechanisms for controlling the plasma temperature and thermonuclear output were available and that refuelling and plasma exhaust would allow quasi-steady-state operation, although the means of achieving any of these essential functions were not specifically incorporated in the design.



(1) toroidal field coils
(2) poloidal field coils
(3) core
(4) blanket module
(5) cooling ducts
(6) duct joints
(7) shield structure and vacuum wall
(8) shield door
(9) shield cooling
(10) shield support
(11) servicing floor
(12) injector, refuel and control access

FIGURE 1. Culham Conceptual Tokamak Reactor Mk II A.

A relatively high first wall power loading was chosen at 4.6 MW/m² with high pressure helium cooling of a liquid lithium metal blanket. The blanket is constructed of many individual cells, allowing mass production and testing of components. Each cell is connected to the coolant ducts within the shield region. The choice of blanket structural material was left open.

A major feature of the Mk II A design is the approach to remote maintenance. The shield region of the reactor is divided into two layers, the outer layer being a permanent structure which also acts as the primary vacuum vessel. This structure has openings, located between toroidal field coils, closed by rigid doors which may be removed horizontally to allow access to the blanket and inner shield. After cutting the coolant ducts, a complete segment of blanket and inner shield can be removed for repair or maintenance in shielded workshops. An independent survey of alternative maintenance procedures concluded that this arrangement offered the highest reactor availability.

Several divertors have been considered for this reactor to exhaust both the plasma diffusing to the wall and impurities. A bundle divertor is an attractive proposition, although a detailed design is not possible until the allowable field ripple has been established. If a relatively high magnetic field perturbation is possible, superconducting windings could be used since sufficient space can be allowed for shielding. If magnetic perturbations must be reduced a copper winding could be placed closer to the plasma, but at the expense of using 10–15% of the reactor output power to energize the coils. If a bundle divertor is not acceptable it would be necessary to use a poloidal divertor, and some of the mechanical implications of this have been investigated in a Mk IIB design which contains a single null in the poloidal magnetic field. Several complications are introduced in the blanket structure and maintenance procedures, but solutions appear possible.

A more detailed analysis of the poloidal field system has been undertaken in another similar study, Mk IIC (Spears & Hancox 1979) which assumes pulsed operation of the reactor without divertor or refuelling of the plasma during the burn. The severe thermal cycling of the first wall due to pulsed operation requires the wall power loading to be reduced to 1.5 MW/m^2 so that a reactor of similar dimensions to the Mk IIA generates a net electrical power of 600 MW. A detailed analysis of the recirculating power required to operate auxiliary reactor plant such as neutral injectors, the energy storage and transfer system, and cryogenic cooling for the magnetic field windings indicates that 20% of the gross electrical output will be required.

3.2. *Reversed field pinch reactor*

The reversed field pinch is an axisymmetric toroidal magnetic configuration in which plasma can be confined at a much higher value of β than is possible in a tokamak. The r.f.p. reactor, therefore, has the advantage of requiring lower levels of magnetic field. Other potential advantages include the possibility of obtaining ignition by means of ohmic heating alone, and the freedom to choose any convenient aspect ratio. A study has been undertaken of an r.f.p. reactor on the same basis as the Mk IIC Tokamak, a net electrical power of 600 MW with pulsed operation being assumed.

As a preliminary study (Hancox *et al.* 1977), the possibility of using copper rather than superconducting windings was investigated, since this should be most advantageous in a pulsed high-β system. It was found, however, that under optimum conditions the recirculating power fraction could not be reduced below 40%, which is unacceptably high. The use of superconducting windings for magnetically confined fusion reactors therefore appears essential.

A conceptual reactor design with superconducting windings has been developed (Hancox & Walters 1978), and is shown in figure 2. The blanket was similar to that used in the Tokamak, but to maintain plasma stability it is necessary to include a conducting shell close to the first wall, and feedback stabilization coils behind the blanket. The shell introduces serious problems, since for good electrical conductivity it must be constructed from copper or aluminium and maintained at a lower temperature than the adjacent first wall and blanket. It must also have a length that is large compared to its radius to remain effective in stabilizing the plasma, and therefore remote maintenance becomes more difficult, since long sections of the blanket and shell must be withdrawn for servicing, this requires toroidal field coils to be moved around the torus.

Operation in the pulsed mode, without refuelling during the burn, allows a plasma burn of 25 s and a full cycle time of 37 s. Between burns the magnetic energy of 8.5 GJ is stored in the reactor windings by currents which flow in the opposite direction to that required during the burn. At start-up and run-down the energy is transferred by means of a homopolar motor-generator which acts as a capacitive transfer element. The windings act as an air-cored transformer to induce the required currents and magnetic fields in the plasma, and owing to the poor coupling a primary current change of 28 MA-turns is required to induce the plasma current of 17 MA. Using a subdivided conductor with nearly 400000 superconducting filaments and 243 copper strands in a 30 kA conductor reduces the power loss in the pulsed superconducting windings to 110 kW at 4 °K.

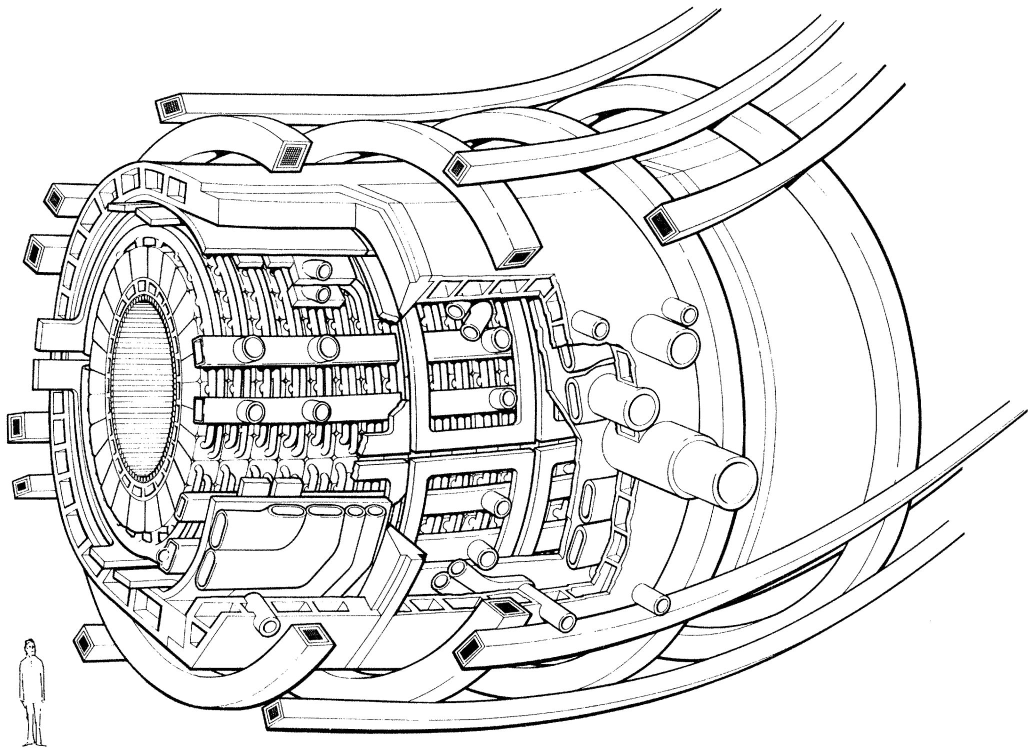


FIGURE 2. Reversed field pinch reactor with superconducting windings (part section, without support structure for magnetic field windings).

An evaluation of the recirculating power fraction and cost of this conceptual r.f.p. reactor indicate that they are close to the figures obtained for the pulsed tokamak reactor. The lower level of magnetic field required in the r.f.p. reactor is roughly compensated by the fact that all windings must, it appears, be pulsed whereas the main toroidal field system in a tokamak is operated at a steady level. The need for a conducting shell in the r.f.p. reactor has serious consequences for access and maintenance, whereas the auxiliary heating system required to obtain ignition in the Tokamak is very expensive. At the current lev el of physics understanding it is not possible to choose between the reactor potential of these two systems.

3.3. *Other reactor studies*

Toroidal magnetic confinement systems capable of steady-state operation include the stellarator and the E.L.M.O. bumpy torus. The stellarator appears unsuitable as a reactor because of its low β and complex magnetic field windings. The E.L.M.O. bumpy torus (Uckan *et al.* 1978) merits further study since it has a high β, high aspect ratio, and modular construction leading to easier maintenance. However, its confinement is poorly understood, and the development of the necessary high power gyrotrons for high frequency heating of the plasma may be technically complex. Other toroidal systems relying on fast heating or compression and pulsed operation, such as the belt pinch (Bustraan *et al.* 1979) or toroidal theta pinch (Ribe *et al.* 1974), do not appear to be attractive propositions because of the cost of rapid energy transfer, high power losses in both normal and superconducting windings, and the difficulty of handling the associated high voltages.

Open magnetic confinement systems include magnetic mirrors, linear pinches, and laser heated solenoids. In general, confinement in these systems is poor, unless linear systems several kilometres long can be contemplated. Magnetic mirror systems are relatively well understood, but have high recirculating power fractions, so very high efficiencies of injection and energy recovery are essential unless energy multiplication in the blanket by fission reactions is considered. One exception might be the tandem mirror (Moir *et al.* 1977) in which a low field central mirror is situated between high field mirrors at each end. This system requires injection energies of 1.2 MeV and peak magnetic field of 17 T, however, presenting extreme technological problems. The potential of the tandem mirror reactor depends primarily on the development of new and more effective methods of preventing the loss of plasma energy from the end of the mirror.

A variety of inertial confinement systems exist with laser, relativistic electron beam, light ion and heavy ion drivers to compress pellets of deuterium–tritium fuel. All depend on the development of high power, high efficiency drivers and involve the use of complex multi-layer pellets whose design involves classified information. Laser drivers have been intensively investigated (Conn *et al.* 1977; Maniscalco *et al.* 1977), but with a shift to shorter wavelengths to improve pellet compression, the possibility of obtaining sufficiently high laser efficiencies is reduced. Beam drivers have demonstrated high efficiencies, but doubts exist in most cases about the ability to transport the beams at high current densities into the target chamber and focus them onto the pellets. In all cases the target chambers are much simpler than those required for magnetic confinement, but must withstand micro-explosions releasing energies of the order of 100 MJ at repetition rates of several shots per second. Currently, it is not possible to judge whether target pellets can be manufactured at the required low cost.

4. Prospects for fusion power

Assessments of the prospects of developing fusion as a future energy source must cover the possible operational, economic, safety and environmental aspects of fusion reactors. While many of the characteristics of fusion power are apparent from its general principles, it is essential to base such assessments on consistent conceptual reactor designs since these often represent a compromise between desirable advantages and engineering or economic necessities. In general, comparisons should be made with all available energy sources, but in practice it is usual to consider fusion as an alternative to fast breeder fission reactors since their characteristics of high capital cost, low fuel cost and extensive fuel reserves are similar.

The primary case for the development of fusion power is that it opens new reserves of energy, since the deuterium–tritium reaction uses lithium as its primary fuel. Lithium is relatively abundant in nature and not widely used for other purposes. Identified high grade reserves are about seven million tonnes, whereas present usage, mainly for lubricants and ceramics, is currently five thousand tonnes per annum. These reserves are equivalent to 9×10^9 MW_{th}-years if the two isotopes could be consumed in equal quantities, leaving 70% of the reserves for other purposes. Other strong arguments for fusion are related to safety and environmental advantages. For example, an uncontrolled nuclear run-away is not expected to occur in a magnetically confined reactor, since the energy transfer rate from α particles to the plasma ions is orders of magnitude slower than the growth rate of instabilities driven by any excessive plasma pressure. The total fuel inventory in the reactor at any time does not exceed 0.5 g, equivalent in quasi-steady systems to about 100 s full power operation. In inertially confined systems there is no possibility of a run-away reaction. The biological hazard potential of the radioactive structure of a fusion reactor immediately after shut-down is expected to be an order of magnitude less than that of the contents of a fully fuelled fission reactor, and there is no serious after-heat problem. The simple fuel cycle requires the separation of untreated deuterium and tritium from the reaction product helium and small quantities of impurities (there are also small quantities of protons produced in the reaction D(D,p)T to be separated), and the separation of tritium from the breeding blanket, all of which can be accomplished in compact and local processing plant. While the choice of structural material is still open, some possible structural elements such as titanium and vanadium decay to very low levels of activity in times of the order of 100 years so that although active storage of waste is necessary the time-scales involved can be short.

The main uncertainties with regard to the safety of fusion are the extent to which tritium may leak from the reactor and the danger which it represents to workers and the general public. Tritium is a β emitter with 12.6 year half-life, and its control is important since it readily passes through most materials at high temperature and is absorbed by all organisms in the form of tritiated water. At the currently accepted radiations levels, the routine escape of tritium from a 2000 MW_{th} station must be kept below 15 mg per day.

A major difficulty for fusion is its potentially high capital cost, based on current conceptual design studies. Several studies, mainly involving tokamaks, suggest that the capital cost of the nuclear island of a fusion reactor power station will be three to six times greater than the equivalent fast reactor, so that the total station costs may be 50–100% higher. This high capital cost is due to both the complex technologies involved, and the low power density. The latter requires that the first wall loading in future conceptual design studies be increased, which places greater emphasis on the development of materials capable of withstanding the radiation and thermal environment. While considering the high capital cost of fusion, it must be borne in mind that the fuel supply and processing costs for fusion are very low, making a significant contribution to the reduction of overall generating costs.

5. Time-scale for the development of fusion

Possible time-scales for the development of fusion have been drawn up in most countries with a major fusion programme. The plans start from the very large experiments now being constructed, which will become operational during 1982–4 and should obtain plasma parameters close to those required in a reactor. These experiments will give results that will allow scaling to reactor size and will demonstrate the effectiveness of heating and impurity control methods. Some will have provision for operating with D–T fuel and give preliminary information on the behaviour of high energy α particles produced by nuclear reactions in the plasma.

The objectives of the following generation of devices could range from a further plasma experiment to an electricity generating reactor, depending on the success of the experiments now under construction and the boldness with which the final objective of fusion power is pursued. A cautious approach would lead to an experiment with reacting plasma, but only the minimum necessary reactor technology, and would require a further stage of development leading to a prototype reactor operating between 2010 and 2020. A more vigorous approach would envisage the next step as being an experimental reactor, with all the necessary technologies being included during operation. Taking this route would lead to a demonstration of all the essential physics and technology by the end of this century.

An important initiative has been taken recently by the International Atomic Energy Agency, which has established a study group to consider the objectives and characteristics of a next-generation Tokamak known as Intor (International tokamak reactor). In their first report (Intor Group, 1980) the study group have presented an extensive data base assessment, listed areas in which further research and development is required, and indicated the possible parameters of a future Tokamak reactor. This device would be an engineering test facility, generating thermal power of 600 MW during burn pulses of 100 s with a duty cycle of 70% and maximum availability of 50%. It was concluded that it was scientifically and technologically feasible to undertake the construction of Intor to start operation in about 1990.

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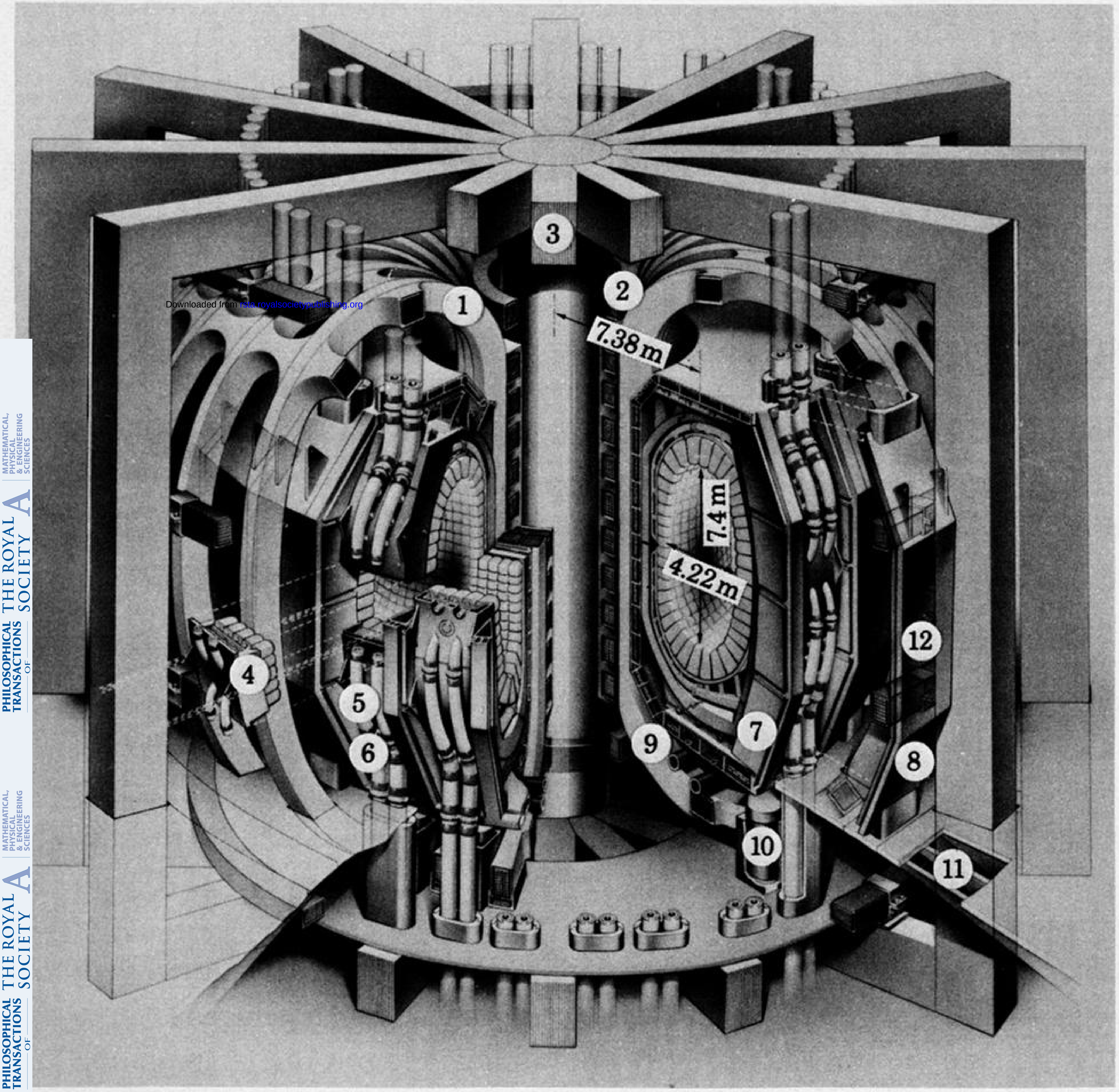

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FIGURE 1. Culham Conceptual Tokamak Reactor Mk IIA.

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