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SYSTEMS STUDIES OF PROPOSED FUSION REACTORS

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ABSTRACT

Systems studies of fusion reactors include the parametric analysis of reactor concepts, the development and visualisation of specific reactor designs, and the overall study of fusion power systems. Their objective is to combine knowledge of plasma physics and nuclear technology to give a balanced view of fusion and its development. To illustrate the present state of systems studies, three reactors are described of which two are tokamaks and the other a laser-fusion system. Such studies allow a comparison between fusion and other power generation systems, and indicate that capital costs for fusion may be high but that this is compensated by safety and environmental advantages.

Les études de systèmes pour les réacteurs à fusion comprennent l'analyse des paramètres des réacteurs des différentes conceptions, l'élaboration et le représentation de projets spécifiques de réacteurs et l'étude générale des centrales de puissance thermonucléaire. L'objectif est d'intégrer les connaissances dans la physique du plasma et dans la technologie nucléaire pour donner une perspective équilibrée de la fusion et de son développement. Pour illustrer l'état présent des études de systèmes, on donne la description de trois réacteurs, deux desquels sont des tokamak et l'autre un système de fusion par laser. De tels études permettent de comparer la fusion avec d'autres systèmes de production d'énergie et indiquent que le capital investi dans le cas de la fusion peut être élevé mais que cela trouve une compensation dans les avantages pour la securite et l'environnement.

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1. INTRODUCTION

Fusion research has been a growing field of activity for more than twenty-five years. Although the final objective has always been the generation of useful and economic power by nuclear reactions between light elements, it is only during the past five years that studies of reactor systems have been undertaken in detail. The reason for this is that it is only in recent years that understanding of the behaviour of high temperature plasmas has developed to the stage at which it is possible to extrapolate to reactor conditions with any certainty. There is now confidence that the necessary plasma temperatures and confinement times to demonstrate the physical feasibility of fusion will eventually be obtained so that it is both reasonable and desirable to envisage the final form of a reactor system. Such activities are covered by systems studies, and are the object of this review.

2. TYPES OF SYSTEMS STUDIES

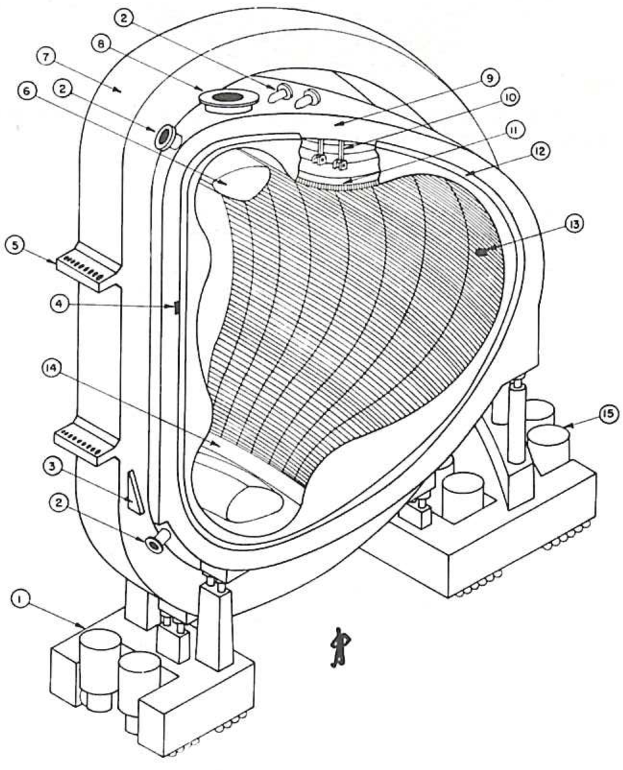
Systems studies may be conveniently divided into three main areas of activity; parametric studies, reactor designs, and power system designs and evaluations.

Parametric studies allow the outline conception and optimisation of any plasma confinement system as a reactor by allowing a wide range of physical parameters to vary, and thus give an overall view of many alternative possibilities. These studies presuppose that there is sufficient knowledge of plasma physics to allow scaling laws to be expressed mathematically, and in so far as this is still not entirely true the resulting conclusions cannot be considered as final. Nevertheless, they are valuable because they high-light critical parameters or assumptions, show which are the important physical or technological constraints, and indicate the broad directions in which future development should proceed. The addition of unit costs allows an estimate to be made of the extent to which fusion could be economically competitive with other energy systems. For example, such studies have indicated the critical importance of superconductor costs for a low-beta stellarator reactor, the necessity of low re-circulating powers for mirror and laser-fusion reactors, and the importance of efficient shock heating in high-beta pinch reactors.

The second major area of activity is the development and visualisation of specific reactor designs. Having identified interesting possibilities by parametric studies, a more detailed design can be undertaken of specific cases. Such designs are conceptual in the sense that it is not intended that they should be built, and because they still contain many assumptions in the plasma physics and unproven technologies. Their importance is that they more clearly identify these uncertainties thus indicating areas where further work is required, as well as bringing together the many results of previous work. Another feature of these studies is that each problem is seen in the context of the whole reactor, thus avoiding the danger of finding apparent solutions to problems which simply transfer the difficulties elsewhere. It is also of value to see the interaction of plasma physics and technological problems, thus helping to establish the correct balance between them. Several such conceptual reactor designs have been completed, and some of them will be discussed in more detail. A tokamak reactor design developed at the University of Wisconsin is illustrated in figure 1.

The third major aspect of this subject is the study of overall power plant designs and their evaluation. In this sense the conceptual reactor designs mentioned above are considered as part of a power producing system which includes heat transfer circuits, electrical generators, and connections to an electrical grid. The way in which the reactor fulfils the needs of the power system and responds to external perturbations is studied. Important aspects of the operation of the reactor are its fuel cycle, load factor, transient response, reliability, capital and generating costs, and sensitivity to resource limitations. Features which may be critical for the continuing development of fusion are estimates of the normal and accidental rates of leakage of radio-activity from the reactor and its associated fuel processing plant, and the build-up of radio-active materials in its structure and their eventual disposal. A complete power station design with which such a study can be undertaken is illistrated in figure 2.

In addition to the three main lines of activity described above other aspects of systems studies should be mentioned. These include studies of sub-systems such as reactor blankets or heat transfer systems, and new concepts studies including alternative power generation or transfer systems such as fissile blankets or advanced energy convertors. It will not be possible to deal with these specifically in this review, which will concentrate on reactor designs and their evaluation.



Isometric view of one module on its motorised vehicle.

1: Front motorised caterpillar	9: Shield
2: Lithium inlet or outlet	10: Rear blanket support rods
3: Front magnet dewar support	11: Heat-removal cells
4: Front blanket support bar	12: Blanket seal flange
5: Magnet support shear beam	13: Neutral beam injection port
6: Vacuum port shield	14: Particle collection plate
7: Toroidal magnet in its dewar	15: Rear motorised caterpillar
8: Vacuum connection	

Figure 1.

Module of the University of Wisconsin 1.5 GW(e) tokamak reactor design UWMAK - I. (Badger et al (1974)). The mean wall diameter is 11m.

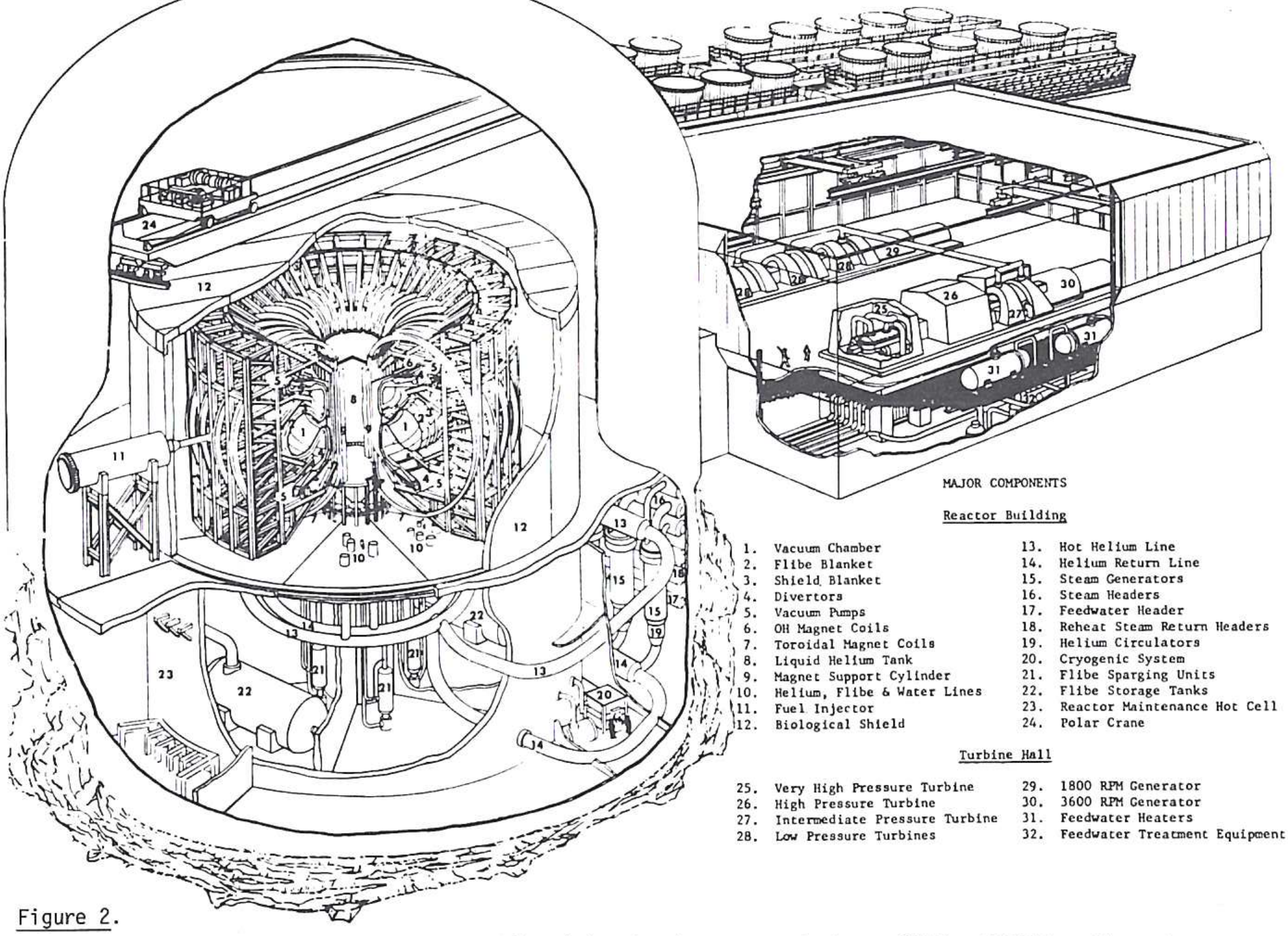


Figure 2.

Overall view of the Princeton University 2.1 GW(e) tokamak reactor design. (Mills (1974)). The major diameter is 25m.

3. THE OBJECTIVES OF SYSTEMS STUDIES

The general objective of systems studies is to combine the present knowledge of plasma physics and nuclear technology in order to obtain the best possible overall view of a fusion power system, both with regard to the balance between its many components and also its possible development in time. More specifically these studies can be of value in the three following ways.

Firstly, to evaluate alternative confinement geometries as potential reactor systems, as an aid to planning the most effective plasma physics research programme. The natural tendency in planning such a programme is to base it on past progress and present understanding in plasma physics. Since, however, the ultimate objective of fusion research is the development of a fusion reactor, this tendency may not give the most effective research programme. The confinement system giving the best plasma parameters will not necessarily form the basis of the best reactor system, and the prospects for the solution of both the plasma physics and the technological problems of each confinement system must be kept in mind. In this respect the present emphasis on the tokamak system may be justified, but there are still major problems to be overcome and it is not clear that other systems should be discounted yet.

Secondly, to evaluate the technological and engineering problems involved in fusion development, as an aid to establishing a balanced reactor technology programme. Here also there is a natural tendency to work in certain areas, usually those that are well established in laboratories with experience in fission reactor technology. Whilst this is partly justified by the fact that these problems are common to many reactor designs based on different plasma confinement systems, it is equally true that many areas are neglected despite their importance to one reactor concept or another. The planning of a balanced technology programme requires the recognition of all the outstanding problems and an assessment of their importance in relation to the full reactor design.

Thirdly, systems studies can be of value in evaluating the operational, economic, and environmental advantages of fusion, as a basis for a possible justification of the growing fusion research and development programme. The development of a fusion reactor will certainly be as expensive as, and will probably take longer than, the development of a fission reactor (Hancox (1974)). Even if a more efficient programme is possible by concentration on only one type of reactor and by extensive international collaboration the cost in money and man-power will be very large. This expenditure can only be justified if a fusion reactor will eventually show substantial advantages over existing or future alternative power sources. An important objective of systems studies is therefore to assess these advantages realistically and quantitatively. Only when they have been convincingly demonstrated will adequate resources be made available for fusion research.

It is clearly recognised that at the present time systems studies are very incomplete and that none of these objectives can be fully realised. There are gaps in the present physics knowledge which will not be filled until much larger plasma physics experiments have been constructed, and technological uncertainties which will not be fully resolved until a prototype reactor is operating. In these respects present systems studies are far from complete and the conclusions which are drawn from them not yet conclusive and must be continuously revised as knowledge increases. However, as fusion passes from the research phase to the approaching development phase the uncertainties should diminish to the point at which reactor designs and systems evaluations represent a reasonably balanced and accurate view of the subject and an effective means of fulfilling the objectives stated above.

4. DESCRIPTION OF REACTOR DESIGNS

To illustrate the present state of systems studies of proposed fusion reactors, three reactor designs will be described. Examples originating in Europe have been chosen, but this is not to deny the thoroughness or value of similar designs undertaken elsewhere, expecially in the United States of America. The majority of designs proposed so far are based on plasma confinement in the tokamak geometry, since this is currently considered to be the system in which physical feasibility will first be demonstrated. Thus two tokamak reactor studies are described which were undertaken with rather different assumptions or objectives, and one laser-fusion reactor is also described, representing an approach which does not rely on magnetic confinement of the plasma.

A fuller list of conceptual reactor design studies is given in table 1, from which it is seen that designs have been completed for reactors based on most of the well known plasma confinement systems with the exception of high-beta toroidal systems such as the reversed field pinch (Bodin et al (1974)) or the steady state high-beta stellarator. All existing designs are based on the use of deuterium-tritium fuel with tritium breeding in a blanket containing lithium since this reaction has the most favourable reaction rate parameter $\overline{\sigma v}/T$, although alternative reactions have been considered for mirror reactors. Many designs are for reactors with a large unit size, of the order of 5000 MW(th) in several cases, on the grounds that such sizes will be required by the time fusion power is introduced.

TABLE 1

Conceptual Reactor Designs

Type	Size MW(th)	Design Group	Reference
Tokamak	5830	Culham	Hancox (1972)
	170	Fintor	Bertolini et al (1974)
	2000	J.A.E.R.I.	Sako et al (1974)
	1000	Oak Ridge	Fraas (1973)
	5300	Princeton Univ.	Mills (1974)
	5000	Univ. Wisconsin	Badger et al (1974)
Mirror	620	Livermore	Werner et al (1974)
Theta-pinch	3600	Los Alamos	Ribe et al (1974)
Laser-fusion		Los Alamos	Booth (1972)
	5000	Jülich	Förster et al (1974)
	-	Oak Ridge	Fraas (1974)

In the following descriptions the emphasis has been placed on the structural aspects of the reactors since several other technological aspects, such as tritium breeding and recovery, radiation damage, plasma heating, and superconducting magnet design, are dealt with in another review at this Conference.

4.1 The Culham Tokamak Reactor

The first reactor design in Europe which attempted to cover all technologival aspects was undertaken at Culham Laboratory (Hancox (1972)). The basis of the design was a 2500 MW(e) tokamak reactor in which it was assumed for simplicity that a steady state could be maintained indefinitely by diffusion driven currents, and is illustrated in figure 3. The dimensions of the reactor were determined from a parametric study and rough economic optimisation which gave a major toroidal radius of 12.5m and a minor first wall radius of 2.5m. The profiles of plasma temperature and density were assumed, together with the limiting values of the safety factor ($q = 2.5$) and the ratio of plasma pressure to magnetic pressure ($\beta = 5.0$).

Considerations of the various neutronic, chemical, thermal, vacuum and other requirements of the plasma containment vessel were related to a practical structural concept, and the scheme adopted is illustrated in figure 4. The breeding region of the nuclear blanket consists of many individual cells, each containing liquid lithium and independently connected to the cooling system. The advantages of this construction are that thermal expansion and radiation swelling can be accommodated, and material thicknesses, stresses, and temperature gradients can be kept tolerably small. Since advanced welding techniques can be more readily applied to small rather than large components, the quantity production of cells in refractory alloys becomes feasible.

The main reactor structure and shield is situated outside the breeding blanket. This structure is a double walled vessel, analogous to a submarine hull, with iron, borated water, and lead in the interspace for shielding. It is divided into thirty-two equal segments, each of which can be removed with its corresponding magnet and modular blanket structure for maintenance of the first wall. The magnet is superconducting, with a niobium-tin conductor, to generate a mean toroidal magnetic field of 9.5 tesla.

Of the engineering problems which have been considered in detail for this design, particular attention was given to the questions of the maintenance and replacement of damaged parts and the implications of alternative coolant fluids. The ability to undertake maintenance of the reactor is important because the first wall of the blanket experiences a neutron flux of 1×10^{15} neutrons/cm^2.sec with neutron energies up to 14 MeV, so that the dose after 25 years' operation at 65% load factor would be 5×10^{23} neutrons/cm^2 resulting in about one thousand displacements per atom. It is not known whether it will be possible to develop materials capable of retaining their high temperature structural strength at such levels of damage, so that provision must be made for the blanket structure to be replaced several times during the life of the reactor. In this design it is necessary to remove a complete segment for maintenance and then replace individual cells.

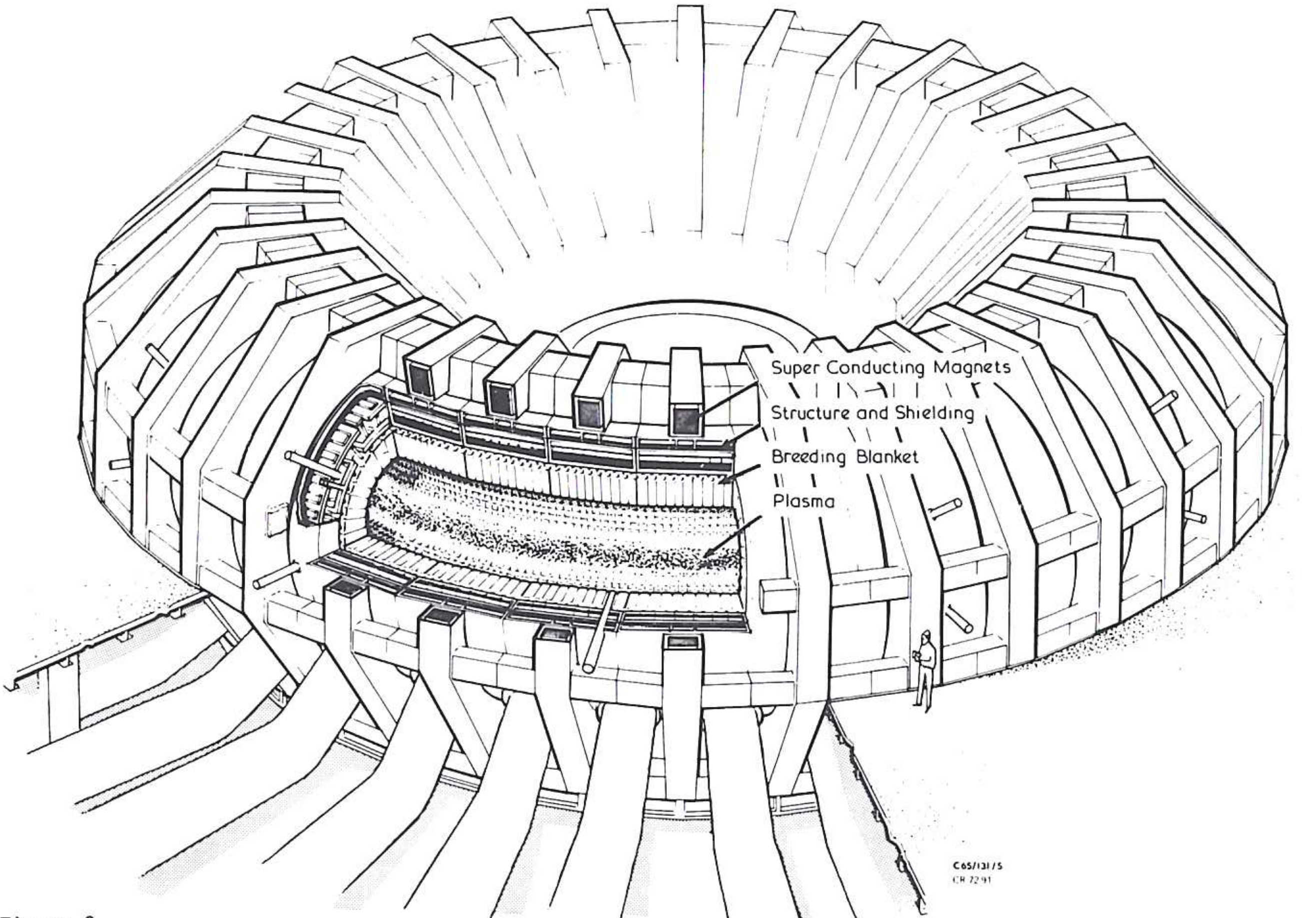


Figure 3.

General view of the Culham 2.5 GW(e) tokamak reactor design. (Hancox (1972)). The major diameter is 25m.

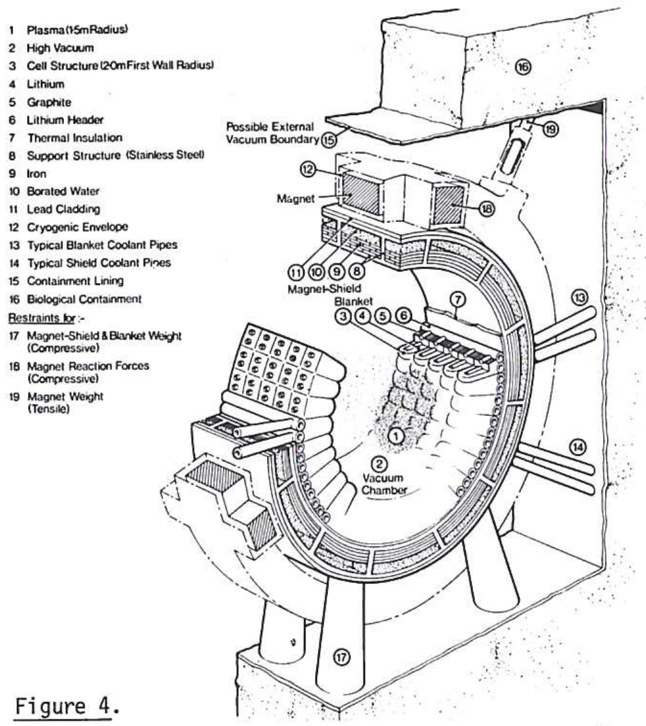


Figure 4.

Constructional arrangement of the components of the Culham tokamak reactor. The first wall diameter is 5m. (Mitchell and George (1972)).

Alternative cooling systems have been considered in detail, with designs for direct lithium cooling and indirect helium cooling (Mitchell and Hancox (1972), Mitchell and Booth (1973)). Lithium cooling is limited by the magneto-hydrodynamic drag on the lithium due to the magnetic field, but is effective when the average power loading on the first wall of the breeding blanket is not higher than 4 MW/m^2. Its advantages are its simplicity and high rate of heat transfer without high pressures. An alternative is helium cooling of static lithium, which may remove the need for a secondary coolant circuit, reduce corrision in the primary heat exchanger, and at high output temperatures may allow the use of gas turbines. The structure of a cell based on the latter possibility is illustrated in figure 5.

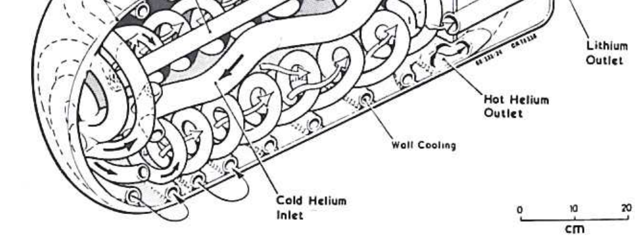


Figure 5.

A helium cooled blanket cell from the Culham reactor.

Amongst the limitations of this design were the omission of a divertor for extracting plasma and reaction products escaping to the wall, or a detailed consideration of structural materials for the breeding blanket. The former was in process of being corrected in a Collaborative Tokamak Reactor Design (CTRD) which was a coordinated study involving Culham Laboratory, K.F.A. Jülich, I.P.P. Garching, and C.N.E.N. Frascati, and was aiming at a design which could incorporate either a bundle divertor or a double-null poloidal divertor.

4.2 The Fintor Tokamak Reactor

This study derives its name from the three groups involved: C.N.E.N. Frascati, J.R.C. Ispra, and the University of Naples. Its objective was to study an experimental reactor, whose construction would precede a full-scale prototype reactor, yet which would include all essential technologies (Bertolini et al (1974).

The size of the reactor was determined by solving in a self-consistent manner the steady state power balance equations for the plasma. The energy confinement time was taken to be proportional to the neoclassical confinement time in the collisionless regime, with a factor of ten to allow for uncertainties. By specifying additional technological limits for the blanket thickness, the size of the transformer core, and the maximim magnetic field, a minimum reactor power of 170MW (th) was obtained. The physical dimensions of the reactor are similar to those proposed for full-size power reactors, major toroidal radius 11.25m, minor first wall radius 2.55m, with the result that the average power loading and neutron flux at the first wall are an order of magnitude lower than in a power reactor.*

The toroidal magnetic field is produced by thirty-six D-shaped niobium-titanium superconducting magnets giving a mean field of 4.5 tesla. The poloidal magnetic field is generated by currents in the plasma and in superconducting windings designed to produce both the vertical field required for plasma equilibrium and the field for a divertor. Two possible axisymmetric poloidal divertors were considered, with one or two null points. The divertor with a single stagnation point was preferred, and is included in the cross-section of the reactor shown in figure 6. Its advantages are that the collector plates are well removed from the plasma, and that during reactor start-up the divertor might still be effective even with constant current in the windings. The main disadvantages of the single null divertor are the space used on the inside surface of the torus, which necessitates a larger aspect ratio torus, and the difficulty of arranging the windings so that they can be divided into sections to allow maintenance of the reactor by removing sectors.

The nuclear blanket contains liquid lithium as the breeding material, and is cooled by helium The coolant outlet temperature is limited to 375°C since the reactor is not intended to produce power efficiently, and this allows the use of type 316 stainless steel for the structure without corrosion problems. The blanket is constructed from 216 rings of two alternating cross-sections which allow the construction of a toroidal system which can be dismantled (Biggio et al (1973)). This concept has been extended by sub-dividing each ring into six modules for easier removal. Alternatively, in a more recent design, the shielding section is

*An early version of this design had the parameters 300MW (th), 10.0m, and 2.5m respectively (Bertolini et al (1973)).

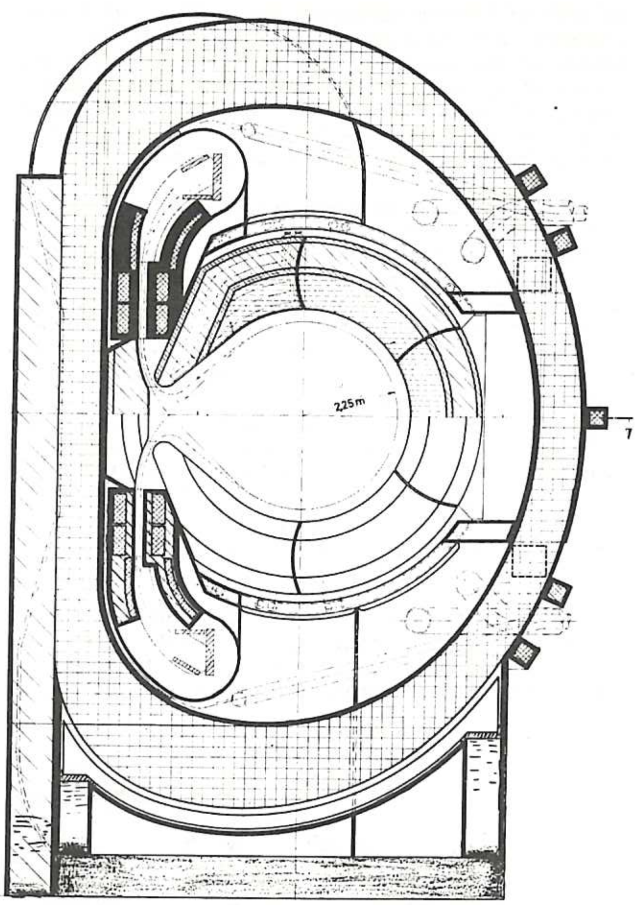


Figure 6.

Cross-section of the Fintor tokamak reactor design with single-null divertor. (Bertolini et al (1974)). The mean first wall diameter is 5.1m.

retained as a modular structure but the breeding blanket is constructed as half rings which can be removed through access holes in the mid-plane of the reactor. This minimises the number of coolant lines which have to be disconnected and the amount of shielding which has to be moved during maintenance, and may allow repairs to be made without moving a toroidal magnet.

For simplicity and flexibility the whole reactor is contained in a vacuum vessel. This vessel is cylindrical in shape with a diamater of 60m and a height of 25m. It is made of concrete and provides the biological shielding and the mechanical support for an internal steel lining which also acts as a tritium barrier.

The Fintor concept, as described above, is still in the process of development. A reduction of the physical size, alternative divertor and blanket structures, and different blanket cooling configurations are under consideration. This illustrates an important aspect of such conceptual reactor design studies, which is that they cannot be complete designs representing a final conclusion but a means of advancing existing knowledge of fusion reactors.

4.3 The Saturn Laser-fusion Reactor

This design, prepared at K.F.A. Jülich, is a 1540 MW(e) reactor with an inertially confined plasma heated by lasers (Förster et al (1974), Bohn et al (1973)). The single spherical unit is supported by two vertical columns, and a horizontal ring; an arrangement which suggests the name Saturn. It is illustrated in figure 7. Fuel pellets of lithium deuteride and tritide, of 1.2mm diameter, are injected pneumatically from the bottom of the vessel at a velocity of 300 m/sec and a rate of up to 100 per second, The pulsed laser light, introduced through 54 beam ports is focussed at a fixed point and the pellets are tracked and steered to this focus by secondary lasers.

The main structure was designed to last the full life of the reactor but allows for complete replacement of the breeding blanket. It consists of a 10m radius spherical grid with 1145 holes, and is water cooled. Each hole is occupied by a module, which may be either a power module (1075) or a vacuum module (70), both of which are illustrated in figure 8. In either case the module is an integral system comprising both breeding blanket and shield, which is of convenient size and weight (8 tonnes, maximum), for easy handling. The modules are arranged so that their inward facing ends shield the main structure from direct exposure to reaction products, and all necessary connections are located on the outer surface of the grid structure.

A novel feature of this reactor design is that instead of a conventional heat transfer system from the blanket supplying a turbo-alternator, a large number of turbo-alternators are incorporated into the reactor. Thus each of the power modules mentioned above also includes its own neon cooling system and Brayton cycle turbine with an inlet temperature of 850ºC; giving an output 1.65 MW (e). The main advantage claimed for this arrangement is a high availability of the whole system, since any component can be easily and quickly replaced by a spare standard unit. The vacuum modules are similar to the power modules, except that the heat generated in the blanket is rejected at low temperature.

Two solutions have been considered for the composition of the breeding blanket. The first uses natural lithium, and the second a much smaller volume of enriched lithium. The second alternative has the advantage that the cell walls and other blanket materials can be kept at substantially lower temperatures due to the larger surface area available for heat transfer to the coolant gas. In either case it is proposed to retain the tritium within the modules, and then sequentially exchange modules after about 100 days' service. This exchange allows the tritium to be removed, the structure of the modules and supports to be regularly inspected, and the moving parts to be serviced. The mean tritium inventory in the reactor due to this method of operation is nearly 60 kG.

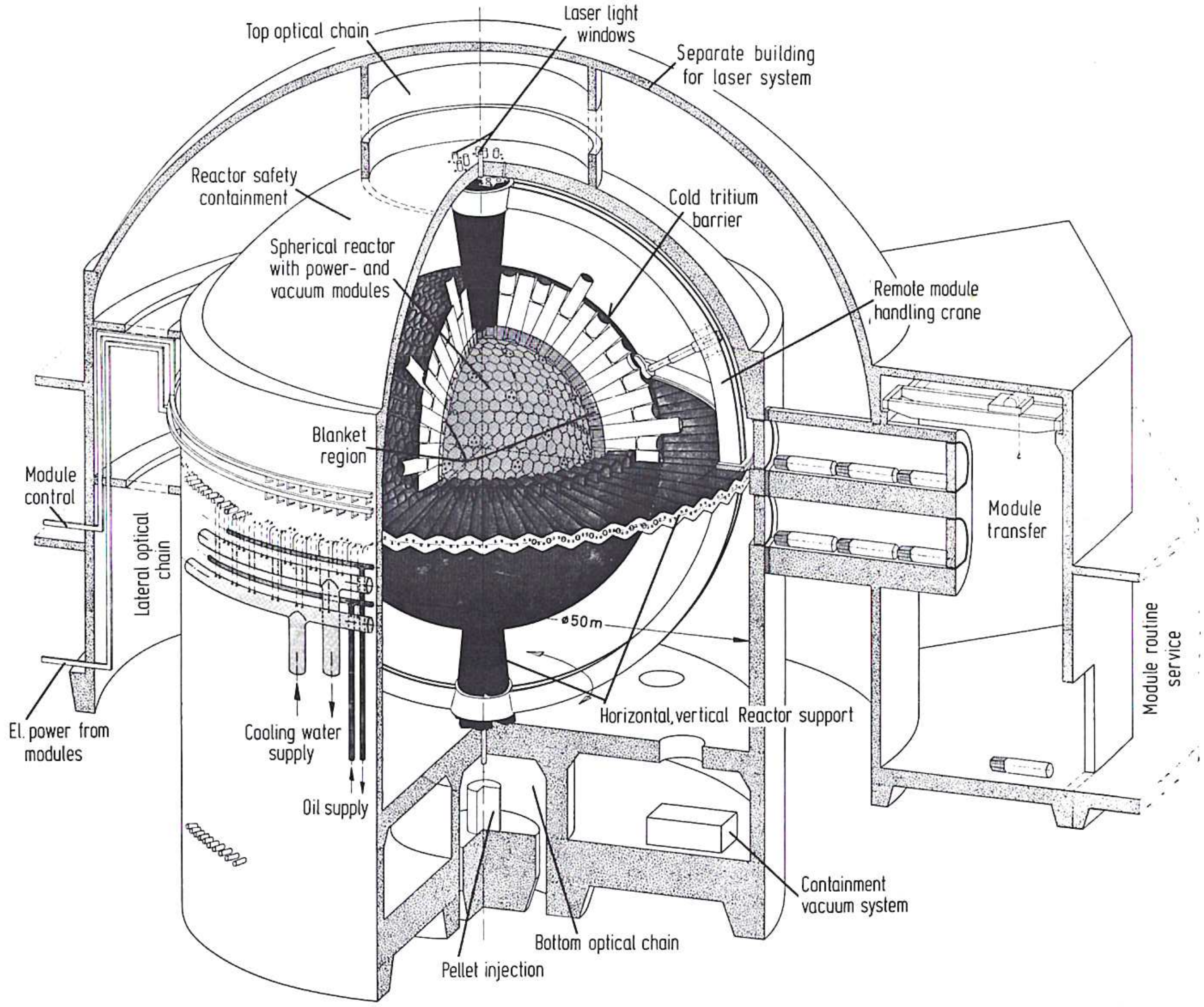


Figure 7.

Overall view of the K.F.A. 1.58 GW(e) laser-fusion reactor design "Saturn".) (Forster et al (1974)). The internal diameter of the reactor cavity is 20m.

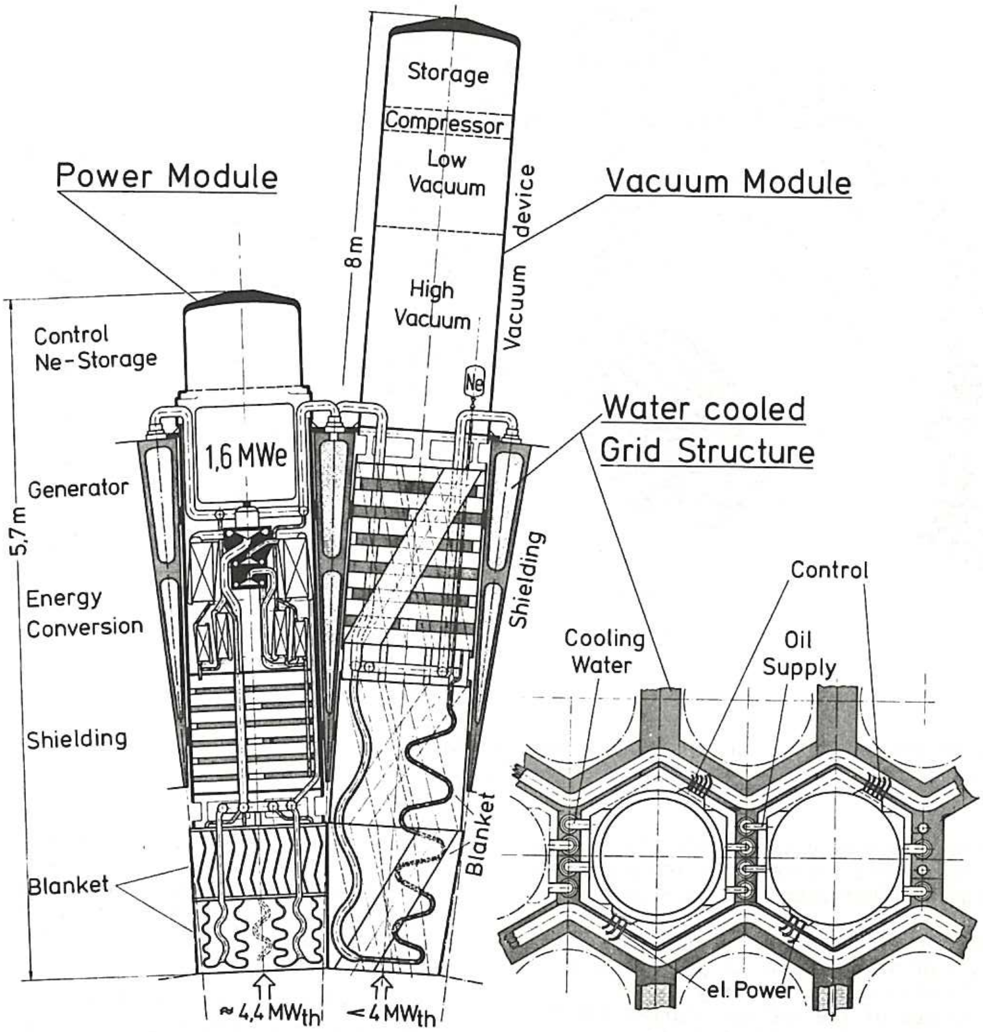


Figure 8.

Constructional details of the power and vacuum models from the Saturn reactor.

The guiding principle for the engineering of this reactor design was the application of convincing performance characteristics and technique. Thus the emphasis was on feasibility, reliability and safety aspects. In all of these aspects the spherical geometry and absence of large magnet windings is an advantage compared with the tokamak designs considered previously. Whether lasers of sufficient power, efficiency, repetition rate and economic life, will be available is still unknown.

4.4 Special Features of other Designs

The three reactors described in the previous sections illustrate many aspects of conceptual designs, but cannot cover all the features found in present studies. Some of these other aspects are mentioned briefly to show the variety of existing designs. A comparison of the parameters of six tokamak reactor designs is given in table 2.

Alternative breeding materials have been considered. The Princeton University (PPPL) tokamak reactor illustrated in figure 2 (Mills (1974)), uses the molten salt lithium-beryllium-fluoride (flibe) which largely overcomes magneto-hydrodynamic losses if the breeding material is circulated through the system as a coolant. Corrosion problems are also reduced, and experience with this material already exists from the molten salt reactor at Oak Ridge. The major disadvantage of flibe is its poor breeding ratio due to neutron capture in the fluorine, but with a suitable choice of structural material a satisfactory breeding ratio is possible.

Other alternative breeding materials are the solids lithium-aluminium alloy or lithium oxide. The latter is proposed in the JAERI tokamak reactor, illustrated in figure 9, in the form of pebbles which might be removed once or twice a year for tritium recovery (Seko et al (1974)). Lithium-aluminium alloy has been considered in conjunction with an aluminium structure to give a reactor structure of very low activity (Powell et al (1974)). If helium coolant is used to extract the heat directly from the breeding material it should be possible to achieve a coolant outlet temperature

TABLE 2

Comparison of Tokamak Reactor Designs

	UWMAK	PPPL	CULHAM	CTRD	JAERI	FINTOR
Power GW (e)	1.5	2.0	2.5	2.0	0.8	(0.05)
Major radius m	13.0	10.5	12.5	16.8	10.0	11.2
Minor radius m	5.5	3.6	2.5	5.7	2.5	2.5
Aspect ratio	2.6	3.2	5.0	2.9	5.0	5.0
Ion Temperature keV	11	56	20	20	25	23
Safety factor q	1.7	2.1	2.5	2.0	1.5	2.0
Beta (poloidal)	1.1	1.6	2.2	0.8	2.0	2.0
Beta (toroidal) %	5.6	3.5	1.4	2.4	3.6	2.0
Plasma current MA	21	15	10	22	8	5
Magnetic field T	3.8	6.0	9.5	4.5	6.0	4.5
Wall loading MW/m^2	1.8	3.4	3.8	0.5	1.6	0.1
Coolant	Li	He	Li	-	He	He
Structure	SS	PE16	-	-	Inc	SS
Breeder	Li	Flibe	Li	-	Li_2O	Li

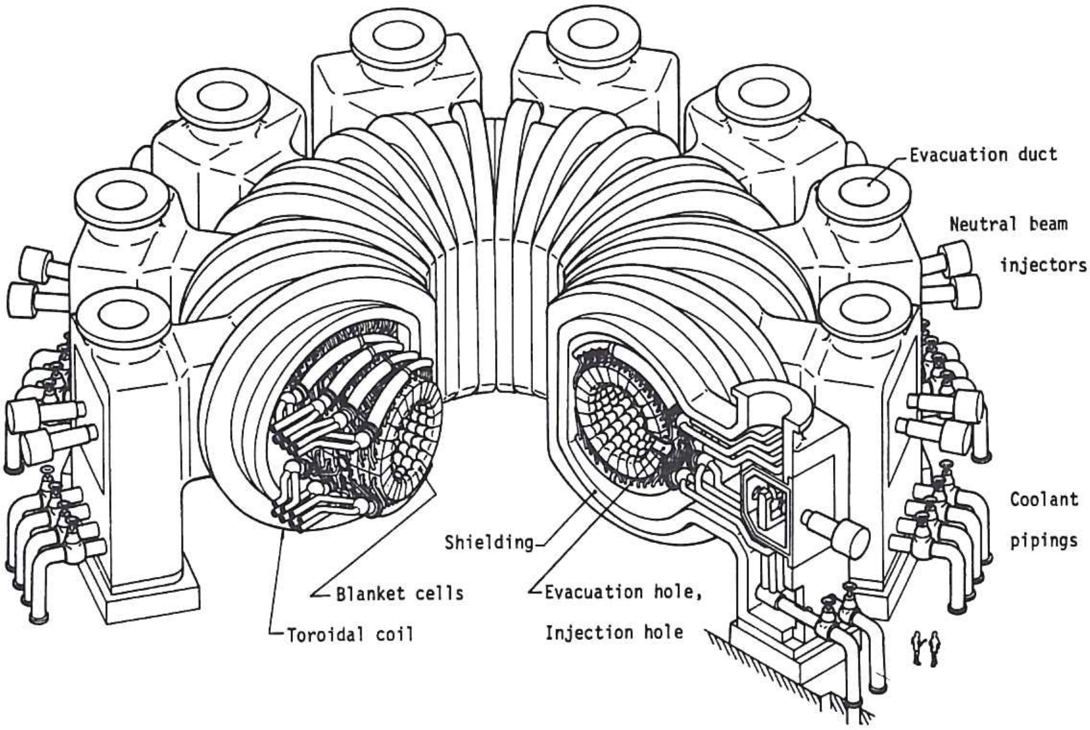


Figure 9.

Overall view of the 800 MW(e) tokamak reactor design developed by the Japan Atomic Energy Research Institute (Sako et al (1974)). The major diameter is 20m.

well above the temperature of the structure, thus achieving a reasonable thermal efficiency despite the low operating temperature of the aluminium.

The best choice of structural material for the first wall of a reactor is at present very uncertain. It will depend not only on the mechanical concept of the nuclear blanket and the ability of the material to withstand intense radiation damage from 14 MeV neutrons but also on possible interactions between the plasma and the wall. A novel arrangement considered in connection with the University of Wisconsin UWMAK II tokamak reactor (Conn et al (1974a)) is the use of a graphite curtain in front of the first wall. The graphite serves two purposes; firstly it is a material of low atomic number so that the effects of plasma contamination are minimised, and secondly

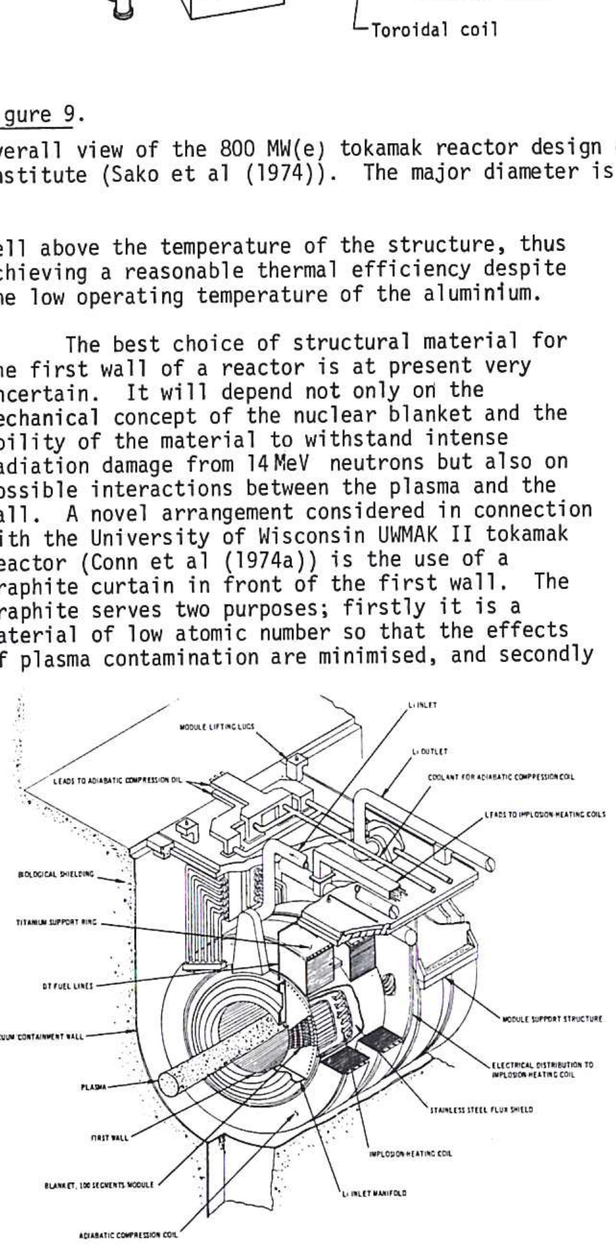


Figure 10.

Module of the Los Alamos Reference Theta Pinch Reactor design (Ribe et al (1974)). The module length is 2m.

it may moderate the energy of neutrons reaching the structure behind it so that radiation damage effects are reduced. Such a curtain could be replaced relatively easily since it is not itself a structural material.

Turning to other confinement systems, the theta-pinch reactor proposed by Los Alamos (Ribe et al (1974)) includes many different features. It is a pulsed device with a burn-time of 0.08 seconds, and therefore uses copper magnetic field coils supplied from an energy storage system. For efficient implosion heating of the plasma part of the coil system must be within the nuclear blanket structure. Since the implosion heating magnetic field changes very rapidly ($\sim 0.1\ \mu$ sec) the sections of the first wall structure must be electrically insulated from each other, and an insulator is required which is capable of withstanding the radiation environment. The theta-pinch reactor has a large toroidal aspect ratio, so that it can very conveniently be divided into a large number of easily handled modules for reactor maintenance and repair. One such module is shown in figure 10.

Quite different features occur in reactors based on an open magnetic confinement geometry, such as the mirror reactor proposed by the Lawrence Livermore Laboratory (Werner et al (1974)). Because of the high circulating power, large neutral injection and direct conversion systems are the most noticeable features of this design. The direct conversion system is physically much larger than the reactor itself since the escaping plasma must be given a directed motion and reduced in density before the ions can be separated from the electrons and decelerated in a multiple grid structure to give a conversion efficiency in the range 60 to 90%. The use of an energy multiplying fissile blanket would be most advantageous and most practical in a mirror reactor.

For reactors with inertially confined plasmas there are many alternative concepts, especially with regard to the first wall. The thermonuclear burn of each fuel pellet is completed in 10 picoseconds, the energy carried by X-rays is deposited on the first wall of the cavity in about 10 nanoseconds, and the energy carried by charged particles in about 1 micro-second, and therefore the protection of the first wall is very important. One concept is a wetted-wall, which relies on a thin protective coating of lithium which forms on the inside surface due to porosity in the wall material and is replaced after each pulse. A dry-wall might be possible with an ablative layer of carbon. Alternatively, the effects due to the charged particles can be transferred to more remote parts of the reactor cavity by the use of magnetic fields. Another novel concept is the Blascon in which the micro-explosion is contained entirely within a vortex formed in rotating liquid lithium, with the advantage that there is no structural material to be replaced or to suffer radiation damage from 14 MeV neutrons (Fraas (1974)).

5. The Limitations of present Systems Studies

Much thought and ingenuity has already gone into systems studies, and many details of reactor and systems designs have been clarified. Nevertheless, it is recognised that there are still many uncertainties in both the assumptions that have been made and in the data that has been used. This does not invalidate the work, but engenders caution in applying the conclusions. Having described what has been achieved, it is also important to examine some of the areas of uncertainty.

A major assumption is the adequacy of plasma confinement, which is the main question being answered in present day fusion research. There is little doubt that adequate confinement times could be obtained in a power reactor based on toroidal magnetic systems, although there may be limitations on the smallest reactor which could contain a self sustained reacting plasma. What is less clear at present is whether in these systems a reacting plasma can be maintained in a thermally stable condition at an economically acceptable plasma pressure and with suitable profiles of density and temperature across the reactor. If neo-classical confinement is obtained the plasma might be thermally unstable so that a control mechanism would be required for steady state operation. If, on the other hand, trapped ion instabilities predominate the temperature may be stable, but even then it might be necessary to find a means of adjusting the operating temperature and the plasma profiles to optimum values. The need for an adequate plasma pressure will be considered later in relation to the economics of fusion power.

For steady-state operation of a reactor it will also be necessary to inject new fuel to replace plasma lost by diffusion or reactions. In toroidal magnetic systems this fuel cannot be injected at high energy because the energy gain per injected particle from reactions is too low. Methods of injection at low energies in the form of either clusters or pellets have been suggested, but there is insufficient understanding of the physical processes involved to be able to judge their chance of success. Such clusters or pellets will be eroded by the plasma as they enter, and the critical question is whether their life will be long enough for a proportion of the fuel to reach the central regions of the plasma. Secondary questions are whether the distribution across the reactor of incoming fuel will be suitable for maintaining the required plasma density profile, and whether the injection process will introduce further instabilities or losses which will affect plasma confinement.

Steady-state operation also assumes that reaction products formed in the plasma can be extracted, and that plasma diffusion away from the central regions can be removed before it interacts with the first wall of the reactor and introduces impurities into the plasma. Unfortunately, present understanding suggests that in toroidal magnetic systems reaction products and impurities will not diffuse out of the reactor as quickly as the fuel, and that divertors cannot be entirely effective in isolating the plasma from the wall. Thus the build-up of unwanted species in the plasma, giving increased radiation losses and lower reaction rates, will eventually limit the time for which the plasma can be maintained. This time has been estimated to be of the order of ten to one hundred confinement times (equivalent to 10 to 1000 seconds), so that a tokamak reactor would have to operate on a pulsed basis with periods of a few seconds between pulses to purge the confinement region and refill with clean fuel. This mode of operation will place a heavier burden on some reactor components such as the auxiliary heating system for ignition, will introduce serious cyclic stress problems, and if the pulse length is too short will reduce the mean power output for a given capital investment.

The problems of refuelling and impurity accumulation apply mainly to the low-beta toroidal confinement systems or the high-beta stellarator operated as a steady-state reactor. Other high-beta toroidal reactors such as those based on the theta-pinch or reversed field pinch will operate as short-pulsed systems without refuelling, but may suffer from inadequate plasma heating or lack of plasma stability. Open configurations can be refuelled at higher energies provided efficient direct conversion is possible, and do not suffer from impurity accumulation, but have only marginal plasma confinement. Thus none of the present reactor concepts can yet claim to have an adequate physics basis, although continuing progress in plasma physics research encourages the expectation that the necessary conditions will be achieved in at least one system.

Turning to the technological aspects of the reactor studies, one problem which has been investigated in several studies but has not yet been satisfactorily solved is how to repair and maintain the reactor. This is particularly difficult in low-beta toroidal magnetic systems because of their geometry and because of the massive structure required for the magnetic field windings. Thus the conventional stellarator geometry with both toroidal and helical field windings is unsatisfactory for a reactor, and alternatives must be found which allow better access.

In the same way the use of axisymmetric divertors in tokamaks may create severe difficulties. Early proposals for maintaining such systems by removing complete segments of the reactor appear rather impractical, and more recently there have been attempts to devise methods which only require the removal of smaller parts of the nuclear blanket and shielding, or even systems for working remotely inside the reactor so that the shielding remains undisturbed.

Another important technological uncertainty is the useful life of structural materials under irradiation with 14 MeV neutrons. The neutron fluxes involved are comparable with those experienced in fast breeder reactors ($\sim 10^{15}$ neutrons/cm^{2}sec), but due to the high initial energy of the neutrons the rate of (n,p), (n,2n), and (n,α) reactions in the structural and first wall materials will be about two orders of magnitude higher than in a fission reactor. Thus the effects of transmutations and embrittlement will be much more serious. The difficulty of access to the first wall of the reactor, as well as the intrinsic value of the materials which may be involved, suggest that the wall material should not be changed more frequently than the fuel elements in a fission reactor, so that the development of improved materials will be critical for fusion. However, since the highest neutron fluxes at the first wall of the reactor will not be encountered until the operation of a power reactor, neither the practical problem nor the success of possible solutions will become immediately apparent.

Many other uncertainties exist associated with the properties and corrosion resistance of materials, tritium breeding rates, the feasibility of extracting tritium from the nuclear blanket, the reliability of large superconducting magnets, and specific aspects of different types of reactor. Their nature and extent is revealed by reactor studies so that although the studies must be incomplete and limited by a lack of information, it is equally true that the studies are useful in assessing the need for future work in fusion reactor technology.

6. Assessments of Fusion Reactors

An important objective of systems studies is to allow an assessment of the prospects of fusion as a future energy source. Such assessments can cover the possible operational, economic, safety, and environmental advantages of fusion. In general comparisons should be made with all alternative energy sources, but in practice it is sufficient to consider fusion in relation to the fast breeder fission reactor since their characteristics of high capital cost, low fuel cost, and extensive fuel reserves are similar. Since the two systems are at different stages of development it is difficult to make precise comparisons, but the outcome will affect decisions on the resources which should be made available for fusion research or, if the outcome were particularly favourable for fusion, could delay the introduction of fast reactors.

6.1 Economic assessments

Capital cost estimates have only been undertaken in detail for tokamak reactor designs. Whilst the different degrees of detail available for fusion and fast breeder reactors makes comparisons difficult, and inflation makes absolute cost figures meaningless, it seems that a fusion reactor based on present plasma physics knowledge will be more expensive than a fast breeder reactor.

A breakdown of the main items of capital cost in the Princeton reactor design are shown in table 3 (Mills (1974)). As with most reactors based on the magnetic confinement of a plasma, the most expensive item is the magnetic field system. The cost of the reactor, together with its associated fuel handling and control equipment and buildings, was 420 Mua or 206 ua/kW(e) at 1974 costs.* A similar study based on the Culham reactor design gave 178 ua/kW(e) adjusted to 1974 costs, compared with an estimate of 70 ua/kW(e) for a 2.5 GW(e) fast breeder reactor. Using cost data derived from the Culham study, the costs of other tokamak reactor designs have been estimated and the results, which are summarised in table 4, fall in the range 176 to 340 ua/kW(e) when adjusted to a common output power. These costs are far higher than

TABLE 3

Estimated cost of the Princeton Reactor Design in thousand of dollars, from Mills (1974)

Reactor Supports and Foundations		
Main Supports and Foundations	5000	
Coil Supports	27000	
Vacuum Vessel Supports	1000	
Shipping Equipment	1000	
		34000
Magnet Systems		
Toroidal Field Magnet	61800	
Divertor Magnets	2630	
Vertical Field Magnets	6600	
Control Field Magnets	2140	
Magnet Protection System	4500	
Power Supplies for Magnets and Protection Devices	33600	
		111270
Dewar and Refrigeration System		
Toroidal Field Dewars	13750	
Divertor Field Dewars	1170	
Vertical Field Dewars	2990	
Control Field Dewars	1720	
Refrigeration Equipment and Piping and Insulation	63000	
		82630
Helium Inventory		
Cryogenic Helium	1900	
Reactor Cooling Helium	430	
		2330
Divertor and Vacuum Vessel		
First Wall	4500	
Vacuum Wall and Supports	8940	
Pumps	2100	
Divertor	14500	
		30040
Shields, Blanket and Cooling		
Breeding Blanket	28100	
Shielding Blanket	18630	
Other Biological Shields	750	
Blanket Cooling Systems	85300	
		132780
Fuel Injection System		10000
Total		403050

TABLE 4

Capital cost Comparison of Tokamak Reactor Designs

(Estimated costs are on the basis of reactor dimensions and cost data for the Culham reactor)

		UWMAK	PPPL	CULHAM	CTRD	JAERI	ORNL
Power	GW(e)	1.5	2.0	2.5	2.0	0.8	0.5
Outlet temperature	^{o}C	480	640	600	-	600	1052
Capital cost (quoted)	Mua	-	420	445	-	-	150
Capital cost (est).	Mua	520	380		790	210	265
Unit cost (quoted)	ua/Kw(e)	-	206	178	-	-	290
Unit cost (est).	Ua/Kw(e)	360	190	-	370	260	510
Unit cost (est. 2.5 Gw(e))	ua/Kw(e)	304	176	178	340	178	298

earlier estimates made on the basis of simplified reactor designs with mugh higher plasma pressure ratios and wall power loadings (Carruthers et al (1967), Rose (1969)).

It has been argued in the past that higher capital costs for fusion were acceptable because low fuel costs resulted in comparable total generating costs. This advantage could be as high as 35 ua/kW(e) equivalent capital cost if natural lithium is used for tritium breeding, but would be less if enriched lithium was required. Such comparisons omit the probable necessity of changing

*Costs in units of account (ua) are calculated on the basis of 1 ua = 50 Belgian francs and the exchange rates effective in 1974.

the first wall structure several times during the life of a fusion reactor, which adds to the operating costs. In the Princeton reactor the cost of operation and maintenance was estimated to be 3.10 mills/kWh, of which first wall replacement each five years contributes 1.03 mills/kWh A survey of operating nuclear plants in the U.S.A. showed an average combined operating and fuel cost of 4.8 mills/kWh, or 4.1mills/kWh, for the ten most recent plants, so that the advantage for fusion is substantially reduced. A further factor which may have to be taken into account is that generating costs quoted for fusion reactors have assumed quasi steady-state operation, whereas present indications are that discontinuous operation may be necessary because of the accumulation of impurities in the plasma.

From these arguments it appears that the cost of fusion power, judged on the basis of present reactor designs, will be high. Although it is not clear that this will prevent the eventual introduction of fusion, since other alternative energy sources may also be expensive and fusion may have substantial safety and environmental advantages, it is certain that fusion would be much more attractive if it were cheaper. On the basis of the figures given above the capital cost of a tokamak reactor should be reduced by a factor of three or four. This could only be accomplished by a reduction of the linear dimensions of the reactor by a factor of about two, whilst maintaining the same levels of power output and magnetic field. This is equivalent to requiring an increase in the power density in the plasma, which in the present designs is in the range 0.5 to 3.5 MW/m^3 compared with 80 MW/m^3 in the core of a PWR or 600 MW/m^3 in the core of a fast breeder reactor.

This reduction in size of a fusion reactor at constant power output imposes more stringent requirements on the plasma confinement. At an optimised plasma temperature and aspect ratio it implies that the ratio of the mean plasma pressure to the toroidal magnetic pressure, β, must increase from currently assumed values between 1.4% and 5.6% to values around 10%. Present tokamak experiments, on the other hand, operate at values of β below 0.1%. Whether such an increase is theoretically or experimentally possible, by the use of shaped cross-sections or other means, must be the subject of careful study.

Although detailed cost estimates for reactors based on other plasma confinement systems are not yet available, it is likely that improvements in expected plasma parameters of the order of those necessary for the tokamak will strongly affect the relative advantages of the different systems. For this reason fusion research must be continued on a broad front until it becomes clearer which system will lead to a commercially viable reactor.

6.2 Safety and Environmental assessments

A preliminary comparison of the safety and environmental aspects of fusion and fission reactors shows the possibility of substantial advantages for fusion. Since fusion is a nuclear reaction there will always be a health and safety risk for people associated with, or living near, a fusion reactor but it appears that this risk could be reduced to a very low level.

One immediate advantage of fusion is that the consequences of an uncontrolled run-away reaction would be small, and could certainly be contained within the reactor vessel. This is due to the fact that the fuel contained in the reacting zone per 1000 MW(e) of output power is only about 0.25 grams of deuterium-tritium mixture, compared with 3 tonnes of plutonium and 10 tonnes of uranium in the core of a fast breeder reactor.

Another advantage of the fusion reactor is that the primary fuels, deuterium and lithium, are non-radioactive and that the primary reaction product helium is also non-radioactive. Furthermore, the secondary fuel, tritium, which must be regenerated by nuclear reactions in the blanket, can be processed simply and on-site with a delay of only a few hours. In this way the tritium inventory is minimised and transportation and storage problems avoided. The diversion of nuclear fuel for anti-social purposes will not be a serious problem.

The two major problems associated with fusion reactors are the leakage of tritium and the build-up of radioactive isotopes in the structure of the reactor. Two estimates of the rate of leakage of tritium suggest rates of about 1 mg/day from a 1000MW (e) station (Draley & Greenberg (1972), Badger et al (1974)). This is about 1 ppm of the total tritium inventory. If it is released in oxide form it could give a dose to critical groups of the population of 6 mrem/year at 500m from the point of release, which is about 1% of the ICRP recommended maximum. If release rates as low as this can be achieved the hazard will be less than is expected from the release of tritium from fission reactors and their associated processing plants. The biological hazard potential would only be comparable to that due to radioactive isotopes released from fossil fuelled power stations. Considerably more detailed study of possible fusion reactors and their tritium processing plants is required, however, before these claims can be substantiated.

The second major problem in fusion reactors is the build-up of radioactive isotopes in the structure due to reactions with 14 MeV neutrons. The activity will rise rapidly to around 1 Ci/w in a period of a few months, and only decay over a period of many years. The calculated activities for several reactor designs are shown in table 5. Although not affecting the general public this activity will be a hazard during maintenance operations and will create a waste disposal problem. If stainless steel or niobium alloys are used for the structure the total biological hazard potential will be of the same order as the plutonium in a fast breeder reactor, although the fact that this material does not have to be recycled greatly reduces the associated danger. Other structural materials could be considered, however, which would improve the situation. The use of vanadium-titanium alloys would reduce the biological hazard potential by about three orders of magnitude. The use of aluminium would substantially reduce both the activity and the decay times. Both of these

TABLE 5

Comparison of Radioactivity of Several Blanket Designs

After two years of operation, from Conn et al (1974b)

Design	Structural Material	Activity (Ci/w_t) after shut down of:				
		1 day	1 week	1 year	10 years	100 years
Princeton	P.E. 16	1.1	0.8	0.2	0.01	8×10^{-5}
Oak Ridge	Niobium	0.8	0.4	2×10^{-3}	6×10^{-4}	8×10^{-5}
Wisconsin	St. Steel	0.7	0.4	0.2	0.01	3×10^{-5}
Livermore	St. Steel	0.6	0.4	0.2	0.01	4×10^{-5}
Brookhaven	Aluminium	0.4	2×10^{-4}	1×10^{-7}	1×10^{-7}	1×10^{-7}

materials would be technically possible but would introduce economic penalties because of the need for lower working temperatures, and again considerably more detailed study of alternative reactor designs is required to quantify the relative merits and difficulties of the many options available.

Recently the study of the environmental impact of fusion has been extended to a consideration of the resources available for the construction of fusion reactors. Although the primary fusion fuels are widely available, some of the other materials which may be required are less abundant, with the implication that these materials might have to be diverted from alternative uses or be obtained in extensive new mining operations. Table 6 shows estimates of the quantities required of several materials, and the ratio of known resources to the resources of the primary fuel lithium adjusted by the relative quantities of each required to build a reactor (Kulcinski (1974)). Some difficulty is foreseen in obtaining niobium for superconducting alloys, helium for cooling the superconducting magnets, lead for shielding, and beryllium if it is required to improve the breeding.

TABLE 6

Materials requirements for Fusion Reactors

based on data from Kulcinski (1974)

Element	Reserves M. tonnes	Required tonne/MW(e)	Resources ratio
Iron	180000	10.0	100.0
Aluminium	3000	0.6	26.0
Chromium	370	2.0	1.0
Lithium	180	0.9	1.0
Titanium	134	0.8	0.85
Copper	310	2.0	0.80
Boron	66	0.8	0.43
Vanadium	26	0.5	0.27
Nickel	24	1.5	0.08
Niobium	8.0	0.8	0.05
Lead	85.0	11.0	0.04
Helium	1.2	0.3	0.02
Beryllium	0.4	0.1	0.02

7. Conclusion

In the previous sections several conceptual reactor designs and reactor assessments have been reviewed. Nearly all of this work has been undertaken during the past five years, and in this period the understanding of what a reactor will look like and how it will work has grown rapidly. On the other hand, much remains to be done before the studies described here can develop into realistic designs of practical reactors.

If alternative confinement systems are to be compared, there is a need for new studies based on geometries such as the stellarator, magnetic mirrors, and several high-beta configurations. These should be developed at least to the depth of present tokamak studies. For comparisons between systems to be of value the underlying physics assumptions must be similar, and because of uncertainties in these assumptions a variety of possibilities should be explored. In this way the objectives of the present phase of plasma physics research will also become more clearly defined.

If progress is to be made on the assessments of fusion as an energy source, existing and future systems studies also need to be continued to greater depth. As would be expected assessments of the capital cost of building fusion reactors and of the operating and maintenance costs have increased as more detail of the designs has been included, and the full extent of this trend must be known. Similarly, the safety and environmental assessment of fusion reactors are at a very early stage, and it is impossible to give quantitative information until much more design information is known. These are important areas for the continuing justification of fusion research.

Finally, although the various aspects of systems studies are an important part of fusion research, it is clear that they cannot be separated from the other activities involved. Their credibility depends on continuing progress in plasma physics, and on developments in reactor technology. Together with advances in these other fields, systems studies can guide and consolidate progress towards a commercial fusion reactor.

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