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September 1976

UWFDM-175

ANS Topical Meeting Paper.

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

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To be published in the Proceedings of the 2nd ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, September 21-23 1976, Richland, WA

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WHAT IS PAST IS PROLOGUE: FUTURE DIRECTIONS IN TOKAMAK POWER REACTOR DESIGN RESEARCH

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We learn from our past endeavors. Conceptual tokamak power reactor designs over the last five years have provided us with many fundamental insights regarding tokamaks as fusion reactors. This first generation of studies has helped lay the groundwork upon which to build improvements in reactor design and begin a process of optimization. After reviewing the first generation of studies and the primary conclusions they produced, we discuss four current designs that are representative of present trends in this area of research. In particular, we discuss the trends towards reduced reactor size and higher neutron wall loadings. Moving in this direction requires new approaches to many subsystem designs. We describe new approaches and future directions in first wall and blanket designs that can achieve reliable operation and reasonable lifetime, the use of cryogenic but normal aluminum magnets for the pulsed coils in a tokamak, blanket designs that allow elimination of the intermediate loop, and low activity shields and toroidal field magnets. We close with a discussion of the future role of conceptual reactor design research and the need for close interaction with ongoing experiments in fusion technology.

INTRODUCTION

Conceptual tokamak power reactor design studies during the past four to five years have provided many fundamental insights about reactor performance requirements and technological problems of tokamaks as power reactors. The first generation of studies[1-10] helped lay the groundwork for the new and still rapidly developing field of fusion reactor technology. They provided a self-consistent context for judging the impact of new ideas and experimental developments and the basis upon which more near term reactor studies, such as on tokamak experimental power reactors, could be carried out with both quality and relative speed. They have also provided the framework needed to begin important subsidiary studies on environmental impact,[11] safety requirements,[12] resource utilization,[13] and a start at design optimization, some of which is clear in the most recent reactor studies.

In this paper, we will review the primary characteristics and conclusions of the first generation of reactor design studies, the trends represented by the most recent design studies (especially with regard to materials performance, power density, wall loading, and size), and new approaches to the design of first walls and blankets, pulsed coils, the toroidal field magnet, and shielding systems which are consistent with future directions in tokamak reactor research. At the end, a brief discussion is included of the future role of conceptual tokamak reactor studies and puts into perspective their place in the field of fusion reactor technology.

THE PAST IN TOKAMAK POWER REACTOR DESIGNS

The first generation of conceptual tokamak reactor designs[3-10] can generally be classified as pedagogical vehicles for uncovering and analyzing in a self-consistent way the technological problems presented by tokamaks as fusion reactors. By the nature of "first studies," a general characteristic is that each design has several, and sometimes many, conservative assumptions regarding subsystem performance. As such, while many fundamental insights and results were developed (and we shall discuss some of these shortly), one did not expect any of the overall designs to be either optimum or inevitable with respect to questions of economy or reactor size. In Table 1, we compile several important parameters characterizing seven conceptual designs, the European Collaborative Tokamak Reactor Design (ECTRD),[8] the UWMAK-I[3] and UWMAK-II[10] conceptual designs, the Culham-Mark-I design,[7] the Princeton Reference Design (PRD),[4] the Oak Ridge National Laboratory study,[5] and the Japanese Atomic Energy Research Institute study, JAERI-I.[9]

The thermal power rating of the four larger studies is about 5000 MW(th) reflecting the fact that in the time frame envisioned for the application of commercial fusion reactors, the nominal plant size will be in the 1500-2000 MW(e) range. The two physically smaller units of ORNL and JAERI have lower power ratings, 1000 MW(th) for ORNL and 2000 MW(th) for the JAERI system. All systems had circular plasma cross sections and all assumed relatively low total plasma β, ranging from a low of 1.45% in the Culham-Mark-I to 6.5% in UWMAK-II. In addition, all but the Culham design assumed a 14 MeV neutron wall loading less than 2 MW/m^2 with the lowest value being 0.57 MW/m^2. The reason for the low wall loading was the general concern and uncertainty surrounding the question of radiation damage and structural material lifetime in a fusion neutron spectrum. Conservative assumptions were made pending improvements in design approaches and further experimental data.

With four of the designs, a relatively low magnetic field at the plasma axis was used to reflect the judgment that NbTi superconductor is the only conductor with which one could presently, with reasonable confidence, consider building a large bore superconducting magnet. Some progress has been made with Nb_3Sn, which could produce maximum fields in the 12-15T range, but the situation is not much changed today. In the UWMAK studies, the relatively low field was to some extent offset by a low aspect ratio. Nevertheless, the power density in the plasma (thermal power output/plasma volume) is less than 1 W/cm^3 for the low field designs and about 2.5 W/cm^3 for the high field designs.

The seven designs listed in Table 1 cover a range of structural materials, including 316 stainless steel, the nickel

Table 1. Conceptual Tokamak Power Reactor Designs (1972-1975)

	ECTRD[8]	UWMAK-I[3] (UWMAK-II)[10]	CULHAM-MK-I[7]	PRD[4]	ORNL[5]	JAERI[9]
Major Radius, R(m)	16.7	13	12.5	11	10.5	10
Plasma Radius, a(m)	5.75	5	2.5	3.25	2.8	2
Plasma Height to Width, b/a	1	1	1	1	1	1
Aspect Ratio, A	2.9	2.6	5	3.4	3.75	5
Plasma Current, I_p(MA)	22.4	21(15)	9.7	14.6	10	8
Poloidal Beta, β_θ	0.76	1.1(2.3)	2.25	1.83	--	2.1
Total Beta, β	0.023	0.052(0.065)	0.0145	0.04	--	0.036
On-Axis Field B_T^o(T)	4.5	3.82(3.57)	9.5	6.0	2.45	6.0
Maximum Field B_T^{max}(T)	8.0	8.66(8.3)	14.0	16.0	--	12.0
Plasma Power Density, P_p(MW/m^3)	0.46	0.78	3.5	2.3	0.62	2.53
Neutron Wall Loading, P_n(MW/m^2)	0.83	1.25(1.16)	3.9	1.76	0.57	1.42
Thermal Power, P_{Th}(MW)	5000	5000	5380	5305	1000	2000
First Wall Material	V-10Ti-1Cr	316 S.S.	Nb-1Zr	PE-16	Nb-1Zr	Mo Alloy
Max. First Wall Temperature(^{o}C)	500	500	---	630	~1000	680

based alloy, PE-16, and alloys of the refractory metals, V, Nb, and Mo. In addition, aluminum alloys as structural materials for blankets have been considered extensively by the Brookhaven National Laboratory (BNL) group[14] using as a framework the UWMAK-I study by the University of Wisconsin.[3]

Together, these studies have allowed us to carry out an initial assessment of the prospects and problems of tokamak reactors. Many important and fundamental conclusions were generated which are worth reviewing briefly.

a. Materials

The problems of materials in a fusion reactor environment were quantitatively assessed. This area represents perhaps the second most difficult problem facing the fusion community, superceded only by the plasma confinement problem itself. Kulcinski[15] recently reviewed this area in depth so we state only the most important conclusions. In 316 stainless steel, the assessment for operation at 500 to 650°C [3,10] is that the life of the first wall might be as short as 2-4 MW-yr/m^2, with the limiting effect being ductility loss. Further, the effects of high displacement damage (10-20 displacements per atom per MW/m^2) and high helium production rates (~200 appm/MW/m^2) would require extensive attention. Recent measurements[16] tend to confirm this assessment with ductility loss appearing so severe at 650°C that the uniform elongation drops below 1% at fluences equivalent to less than 2 MW-yr/m^2. There may be a saturation in ductility loss at about 1/2% uniform elongation but it is unlikely that this could be utilized. On the other hand, it is also critical to know the effects of temperature on lifetime and measurements[16] at approximately 350°C in stainless shown much better properties after irradiation. We shall return to this point in the next section.

b. Reactor Plasmas

In addition to the fundamental questions of scaling laws, reactors will present several new problems. Four outstanding areas identified were the startup of large, high current plasmas, impurity control, fueling for long burn times, and thermal stability of the plasma burn. These problems were studied in greatest detail in the JAERI-I, PRD, and UWMAK work.

The area now receiving greatest attention in present and upcoming experiments is impurity control. The double null axisymmetric poloidal divertor design developed in the UWMAK research is also the design to be studied on the Poloidal Divertor Experiment (PDX)[17] and the Axisymmetric Divertor Experiment (ASDEX).[18] Also, the use of low Z liners or coatings included in studies at General Atomic[19] and on UWMAK-II[10] is now receiving wide attention and many different approaches have been suggested. The Impurities Studies Experiment[20] as well as most existing devices will add important information.

c. Magnets

Superconducting toroidal field magnets will be essential for tokamak reactors and the problems associated with the design of large bore coils were especially analyzed in the UWMAK and PRD studies. The constant tension "D" shaped design [21,4] is now widely used in both near term and commercial reactor studies and it is especially appropriate with noncircular plasma cross sections. The need to develop a conductor design compatible with the size and stress levels in reactors (horizontal bores from

6 to 20m, stress levels in steel structures less than about 45 ksi, stress levels in copper less than 5-10 ksi) was clearly identified and conceptual design approaches were developed[3,10] which are now being pursued experimentally. The problems with utilizing Nb_3Sn or other very high field superconductors were made clear and substantial work is still required to demonstrate its feasibility.

With regard to ohmic heating (OH) and vertical or equilibrium (VF) field coils, all designs utilized pulsed superconducting magnets and the problem of a/c losses caused by the pulsed fields was clearly identified and analyzed.[10] This problem has been further analyzed and protection methods devised in studies of more near term Experimental Power Reactors (EPR).[22,23]

Another technical area which could involve superconducting magnets relates to plasma startup and peak power requirements. Energy storage systems appear to be required because power demands exceed levels which could be taken from the line even for short times. The type of energy store depends heavily on whether the plasma current rise to the 10-20 MA range is less than or greater than one second. Less than one second rise times would require high peak powers, well in excess or 1000 MW, and demand efficient, fast switching devices like homopolar generators.[24] Greater than one second rise times (e.g., 10s) leads to lower peak power values (less than 1000 MW) and allows use of a relatively straightforward Graatz bridge transfer system. [3,10]. Recent results on the Princeton Large Torus (PLT) indicate that the plasma current rises to about 500 kA in 50 ms filling the chamber bore without instability. Such a plasma could act as the kernel for growing a larger plasma in a controlled way using either an expanding mechanical limiter[25] or an expanding magnetic separatrix. In this case, the rise time should be controllable and the simpliest solution could prevail.

d. <u>Neutronics, Blanket Design and Tritium</u>

A clear conclusion from early studies was that the tritium breeding ratio (BR) in liquid lithium cooled blankets could be much in excess of requirements. BR values of 1.4-1.6 are calculated while the BR values needed for a 1-4 year doubling time is in the 1.01-1.05 range.[26] Solid breeders including intermetallic compounds like LiAl and Li_7Pb_2 and ceramics like $Li_2Al_2O_4$, can theoretically allow low lithium and tritium inventories [27,14,10] while the use of a neutron multiplier like Be appears essential for adequate breeding. It was found that the tritium inventory is highly sensitive to the diffusion coefficient and particle size so that sintering at elevated temperatures is a special concern. Since these effects could negate the potential advantages, clearly required experiments have already begun. [28]

Blanket neutronics methods and analysis advanced considerably as part of the conceptual design efforts and codification of methods for calculating space dependent neutron and gamma heating [3,29] and induced radioactivity and afterheat from radioactive decay [30,31] were achieved. The result is that more realistic and accurate calculations have been done on the total energy released per fusion event (18-24 MeV per fusion depending on blanket design) and that the saturated level of induced radioactivity, upon system shutdown, is about 1 Ci/W(th). Since the designs involved many different structural materials, it became abundantly clear that the residual amount of induced activity is a strong function of materials choice.

One of the most difficult and important problems identified was that of remote maintenance for the repair of activated components, such as the first wall and blanket structure. This has led to efforts at modular blanket design and ready access which continue in all ongoing studies representing a small start on this difficult and long range problem.

Tritium extraction and recovery methods were outlined and discussed in detail in several of these reactor studies with important results. Methods for extraction were outlined, such as the use of Yt beds or Nb windows, and experimental work was suggested. In addition, assessments were made of the most likely paths for tritium release (which appears to be via the power conversion system) and analysis resulted in estimated release rates for tritium of 1 Ci/day or less for a 5000 MW(th) unit.

e. Power Cycle and Plant Design

The power cycle studies have shown the importance of thermal energy storage to provide continuous thermal power to the turbines in an inherently pulsed system like a tokamak. Added costs can be as much as 50-100 $/kWe so that work in this area can provide a substantial payoff. Plant designs indicated the likely arrangement and size of plant buildings and showed the requirements for an on-site hot cell and solid radwaste handling and storage area. The only unusual problem identified relates to leakage magnetic fields and their effect on control systems in the plant. The overall size of the buildings and plant were large reflecting the original choices for the major and minor radii and the neutron wall loading.

g. General Result

A general consequence of the first generation of reactor studies is a solid quantitative basis for the field of fusion technology. Certainly new approaches will solve many problems and further work will add to our understanding, but the foundations have been laid. Indeed, for many researchers in the field, this was the primary purpose of the studies in the first place.

Let us now turn to a discussion of current tokamak reactor studies and the directions they represent. We will then discuss several new approaches to subsystem design and directions in future reactor design research.

PRESENT TRENDS IN CONCEPTUAL POWER REACTOR DESIGNS

Of the conceptual design research ongoing today, four studies are indicative of current thinking and representative of future trends. We have already noted that the first generation of studies summarized in Table 1 were generally based on several or many conservative assumptions that produced low power density, large volume systems. The size of a reactor producing a given amount of power depends on two parameters, the power density in the plasma (which fixes the necessary plasma volume), and the neutron wall loading (which fixed the area of the chamber and which will ultimately be determined by a complex tradeoff between wall life, down time, plant factor, operating temperature, power conversion efficiency, and the volume of solid radwaste generated). The plasma power density is given by

$$P_p = Cn^2 \langle\sigma v\rangle$$

where n is the density, $\langle\sigma v\rangle$ is the Maxwellian averaged reaction rate parameter, and C is a constant. In a β limited plasma such as a tokamak, n^2 can be replaced in favor of β and the toroidal magnetic field so that

$$P_p = \beta^2 B_T^4 f(T)$$

where f(T) is only a function of temperature. The power density, therefore, depends on two factors, β, a parameter determined by plasma physics, and B_T, a parameter controlled by magnet technology. Either or both can be increased to increase P_p.

The neutron wall loading is given by

$$P_n = P_p\left(\frac{14.1}{E_F}\right) \frac{V_p}{A_w}$$

where E_F is the total energy per fusion event, V_p is the plasma volume and A_w is the first wall area. Thus, to reduce the size of a unit generating a given amount of power, it is not sufficient simply to increase P_p. It is also necessary to increase P_n so that A_w can be reduced.

The four design studies referred to above are the UWMAK-III design of the University of Wisconsin,[32,33] the Culham-Mark-II,[34] the JAERI-II design,[35] and the Collisional Tokamak Reactor (COTR) research of MIT and Princeton University.[36]

The UWMAK-III work is essentially complete and detailed documentation is available.[33] The Culham-Mark-II design, which will be used as a vehicle to study the important problem of remote maintenance, and the JAERI-II study are still in progress but preliminary design parameters are available. The first COTR study was scoping in nature and dealt primarily with the extrapolation of collisional tokamak plasma results which show that $n\tau_E$ scales as n^2. Work is in progress on the engineering aspects of COTR devices but details have not yet been reported. A summary of parameters characterizing these designs is given in table 2.

The general trend towards reduced system size is illustrated in Fig. 1 where the size of operating or planned tokamaks is compared with the first generation of reactor studies and with these lastest designs. The most obvious fact is that three of the four current systems are considerably smaller than the first generation of studies (compare Tables 1 and 2) and are not very different in size, except for plasma shape, from the EPR studies.[22,23] Only the JAERI-II design has a lower power density or wall loading than its predecessor. As a general observation, we note that the UWMAK-III, Culham-Mark-II and COTR systems all have noncircular plasma cross sections with elliptical or triangular "D" shapes and a plasma height to width ratio of 1.75 to 2. Thus, the plasma shape factor defined as the plasma circumference/$2\pi a$ is about 1.5 for each design. Since

$\beta = \beta_\theta \left(\frac{S}{qA}\right)^2$, the UWMAK and Culham studies achieved high plasma power density by utilizing plasma physics to increase β. The COTR study, on the other hand, utilizes the technology factor and assumes a maximum B_T of 16 T, which is beyond the useful range of 8-10 T for NbTi. The general trend is clearly to reduce the size of tokamak reactors by utilizing methods to increase β, B_T, or both.

Fig. 2 illustrates size reduction in another way by showing a comparison between the plasma and toroidal field (TF) magnet size and shape in UWMAK-II[10] and UWMAK-III.[32,33] The major radius of the plasma has been reduced from 13 m to 8.1 m and the maximum horizontal bore on the magnet has been reduced from 19.25 m to 13 m. Com-

Table 2

Conceptual Tokamak Power Reactor Designs

(1976)

	UWMAK-III[32,33]	CULHAM-MK-II[34]	JAERI-II[35]	COTR[36]
Major Radius, R(m)	8.1	7.4	10.5	5.0
Plasma Radius, a(m)	2.7	2.1	2.7	1.1
Plasma Height to Width, b/a	2.0	1.75	1.0	1.75-2
Aspect Ratio, A	3.0	3.52	3.9	4.5
Plasma Current, I_p (MA)	15.8	11.6	10.4	8.4
Poloidal Beta, β_θ	2.2	1.87	2.0	2.25
Total Beta, β	0.09	0.093	0.033	0.04
On-Axis Field (T)	4.05	4.10	6.0	7.7
Maximum Field (T)	8.75	8.0	11.0	16.0
Plasma Power Density, P_p(MW/m^3)	2.3	4.75	1.32	4.2
Neutron Wall Loading, P_n(MW/m^2)	2.5	5.7	1.3	1.8
Thermal Power, P_{Th} (MW)	5000	5380	2000	1085
First Wall Material	TZM	UNSPECIFIED	Mo Alloy	UNSPECIFIED
Max. First Wall Temperature (°C)	1000	UNSPECIFIED	680	UNSPECIFIED

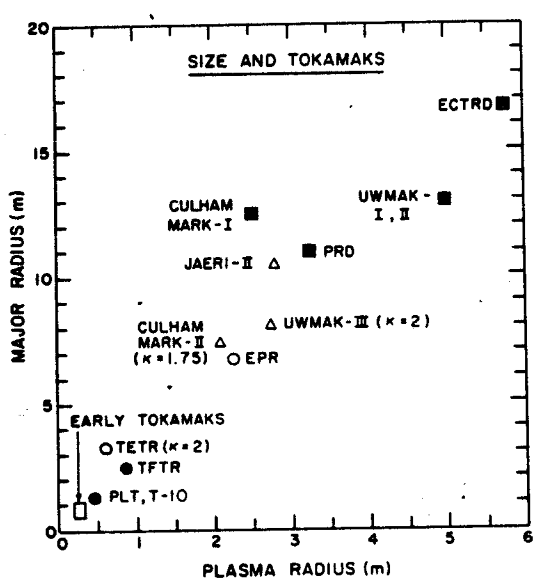

Figure 1 Relationship between the size of operating and approved tokamak experiments, near term reactors, the first generation of commercial reactor studies, and current reactor designs.

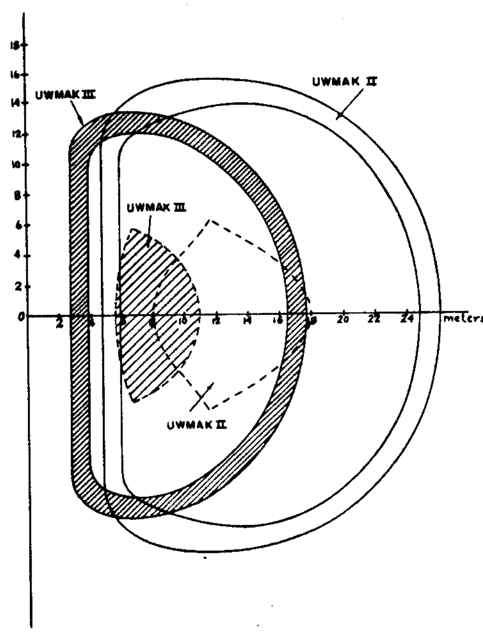

Figure 2 A Comparison of the plasma and toroidal field magnet size and shape between the UWMAK-II[10] and UWMAK-III[32,33] conceptual reactor designs.

pared to a fission reactor core, however, the nuclear island made up of the plasma confinement chamber, the blanket and shield, and the ohmic heating (OH), vertical field (VF), and TF coils is still relatively large. It is therefore necessary to consider the entire plant design to determine how other design aspects effect the overall building and plant size.

This has been done in the UWMAK-III study by the University of Wisconsin in conjunction with the Bechtel Corporation and a plan view of UWMAK-III is shown in Fig. 3-a. The primary containment building is 70 m in diameter and 70 m high. For comparison, we show in Fig. 3-b a plan view of the Clinch River Breeder Reactor Plant (CRBRP), a liquid metal cooled fast breeder reactor representative of future advanced fission reactor systems. The primary containment building for CRBRP is 60 m in diameter and 80 m high. The added height is required to provide space for the vertical removal of fuel assemblies. Interestingly, these two systems are comparable in size. One is not appreciably larger than the other. Only the turbine building of UWMAK-III is larger and this is because it contains four turbines generating 1985 MW(e) compared with one turbine generating 350 MW(e) in CRBRP.

The UWMAK-III and JAERI-II work have utilized a molybdenum alloy (TZM for UWMAK-III) but by contrast this should not be taken as a trend. The primary goal of the UWMAK-III study was to assess the problems and prospects associated with utilizing several advanced technologies in tokamak reactor systems. The Mo alloy, TZM, appears to be the most promising of the advanced high temperature refractory metal alloys.[32,33] Nevertheless, for near term and first generation commercial applications, experience suggests stainless steel or nickel based alloys will be utilized.

Whatever material is used, it will be necessary to develop design solutions that will allow one to achieve reliable operation and reasonable lifetime at elevated neutron wall loadings. In the next section, we discuss this problem together with several other important design ideas regarding magnets, shields, and power cycles which will be important in future considerations of tokamaks as reactors.

NEW APPROACHES AND FUTURE DIRECTIONS

Having laid the basic foundations, it is clear that we are now headed towards reducing the required size of tokamak reactors and developing more optimized solutions to subsystem design. Yet to achieve the goal of size reduction, we must develop approaches that permit first wall and blanket operation at relatively high neutron wall loadings. We must likewise develop improvements and new approaches for many of the other reactor subsystems. In this section, we shall consider four particular problems to illustrate the directions we must follow.

a. First Wall and Blanket Design

A trade-off is involved between neutron wall loading, long first wall life, size, plant factor, power cycle efficiency, and the amount of solid radwaste generated. Clearly, we would like to combine high wall loading with high temperature operation and long wall life. Unfortunately, it may not be possible to achieve each of these simultaneously. What is needed to ultimately decide this trade-off is experimental data on wall life versus

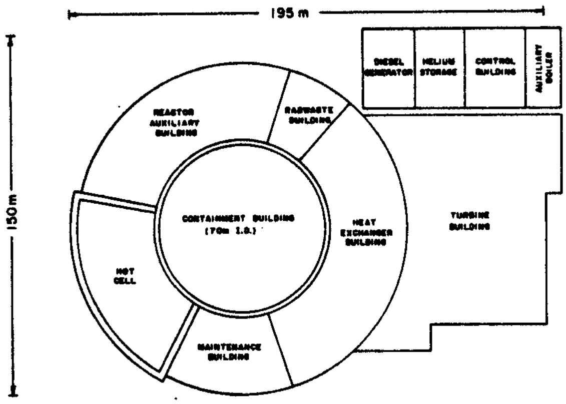

Figure 3a A plan view of the UWMAK-III conceptual reactor plant.

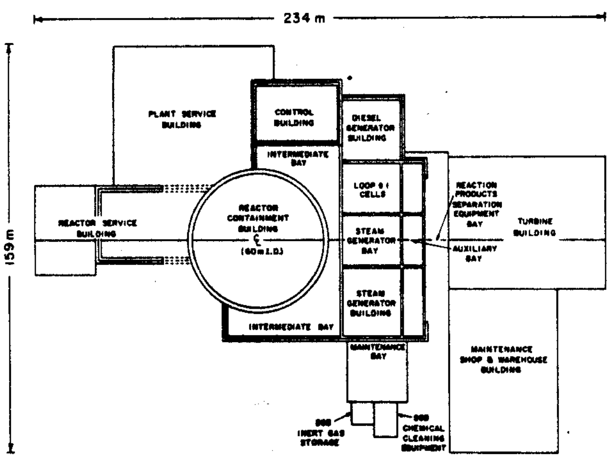

Figure 3b A plan view of CRBRP, the Clinch River Breeder Reactor Project, a demonstration LMFBR.

temperature at realistic operating conditions (e.g., including cyclic fatigue effects).

For most materials, such a curve will take the form shown schematically in Fig. 4 where the lifetime is measured in MW-yr/m^2, a fluence unit based on the number of years one could operate at a given neutron wall loading. There are three distinctive regions of this curve, a low temperature brittle fracture region, an intermediate temperature void swelling region, and an intermediate to high temperature helium embrittlement region. At sufficiently low temperatures, a high density of interstitial dislocation loops occurs which hardens grains and limits lifetime by brittle fracture. Lifetime increases with temperature until void formation occurs and swelling sets in and causes a downturn in lifetime. However, the void swelling phenomena disappears at higher temperature and the lifetime curve would turn back up except that now the mobility of the helium gas becomes high enough to cause accumulation at grain boundaries and loss of ductility. Thus, the final high temperature region shows rapidly decreasing lifetime with increasing temperature. Again, fig. 4 is schematic and in some materials, various regions may

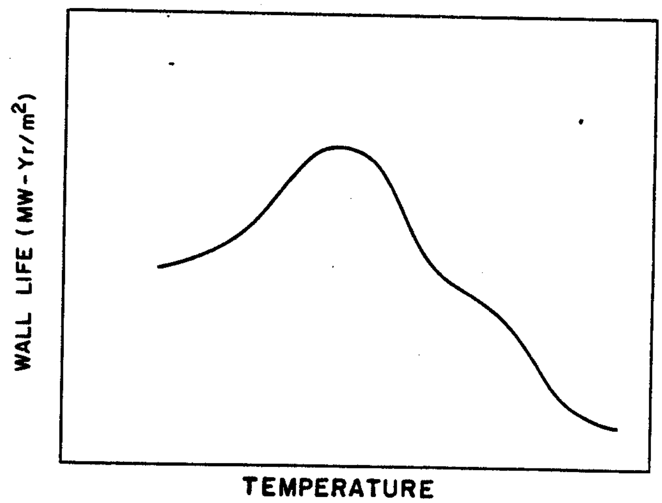

Figure 4 Schematic variation of first wall lifetime as a function of operating temperature.

overlap more or be spread further apart. However, what a designer needs to know is the shape of such curves.

To be specific, consider the case of 20% cold worked 316 stainless steel and the recent measurements of Bloom et al.(16) Samples irradiated at about 350°C showed an increase in ductility with dose and low swelling. Samples irradiated at about 600°C, on the other hand, showed a rapid loss of ductility to less than 1% uniform elongation. The tentative conclusion is that at 350°C, no apparent limit to wall life was found, whereas at 600°C, lifetime estimates can range from 1 to 10 MW-yr/m^2 depending on the design limit employed. The data are probably insufficient to settle the question, but the lifetime could in practice be less than 1 MW-yr/m^2 at 600°C.

A temperature of 350°C or less is a low maximum temperature for a power plant leading to relatively low values for the plant thermal efficiency. On the other hand, plant reliability and availability should be enhanced increasing the plant factor and somewhat counteracting the economic impact of low thermal efficiency. An optimum strategy for blanket design is to couple this information with the fact that the neutron flux decreases exponentially through the blanket so that one could allow the temperature to increase as the flux decreases, keeping the material lifetime approximately independent of position. A less sophisticated strategy is to separately cool the zone in the vicinity of the first wall (and the first wall itself) to keep the maximum temperature to 350°C or less.

As an example, we have considered using boiling water as the first wall coolant to minimize thermal fatigue. With 316 stainless steel structure, the steam for use in a power cycle would be at about 250°C and 650 psi yielding a thermal efficiency of about 25%. The first wall power is however only about 20% of the total. For the remaining 80% of the power, a higher outlet temperature can be achieved leading to a power cycle efficiency of 38-40%. The overall plant efficiency would then range from 35 to 37%. Thus, one would lose 3% on the net plant thermal efficiency but, in exchange, one obtains a long lived first wall and blanket system of high reliability, a higher plant factor, little if any first wall thermal fatigue, much lower levels of solid radwaste generation, and little if any effect on the blanket breeding ratio.

b. Blanket and Power Cycle Integration

In addition to temperature control on the first wall and blanket structures, one can effect a general decrease in primary temperature with no decrease in power cycle efficiency by removing the intermediate loop. In most studies heretofore, an intermediate heat exchanger (IHX) is included for two reasons; to isolate the primary loop from the power cycle to limit tritium leakage and to provide a working fluid for thermal energy sotrage to level the power load to the turbines. The presence of an IHX implies an extra ΔT and thus the need for a higher primarily leg temperature.

To eliminate this loop, one must develop a blanket concept where the working fluid has a high density-heat capacity (ρC_p) product and where tritium is mainly in the form of T_2O or HTO so that diffusion is minimal. Such a design is the Li_2O moving bed blanket concept developed by Sze et al.(37) in Wisconsin.

In this approach, chemical industry experience with beds of particles moving under gravity flow and transported mechanically or pneumatically is utilized. Such beds act as heat transport media with high ρC_p values rather than heat transfer media or coolants. In a fusion reactor with a separately cooled first wall, heat transport is required, not heat transfer.

The advantages of the Li_2O moving bed concept are low pressure, relatively low operating temperature with stainless steel structure ($\lesssim$ 450°C), allowance for separately cooled first walls, control of blanket thermal cycling by control of the bed velocity, the use of solid breeder material without the need for a neutron multiplier to achieve adequate breeding, high ρC_p values for the bed so that energy storage of Li_2O is adequate for load leveling, and tritium in the form of HTO or T_2O so that an IHX can be eliminated. No liquid metals occur anywhere in the system and there are distinctive safety advantages. In particular, a moving bed has great inherent stability unlike fluidized bed systems. Studies are underway to analyze tritium recovery methods,[38] bed transport techniques,[39] steam generator design,[39] and the economic advantages of low pressure, long lifetime blankets and power cycles without an intermediate loop.

c. Normal OH and VF Coils

A difficult problem on tokamaks of the near future may be the use of pulsed superconducting coils. We have, therefore, investigated the power requirements associated with normal OH and VF windings using for illustration cryogenic aluminum at 4.2°K and water cooled copper at room temperature. Three reactors were considered, a small tokamak engineering test reactor (TETR) developed by the University of Wisconsin,[40] the EPR developed by ANL[23] and UWMAK-III. The EPR and UWMAK-III were originally designed with all coils superconducting. As such, the power requirements for normal OH and VF coils might be reduced in these systems by further optimization. In TETR, the OH windings are cryogenic Al at 4.2°K while the VF coils, being quite close to the plasma, are made of water cooled copper.

The results of this analysis are shown in Table 3 and one sees that for either TETR or EPR, the power requirements are modest compared, for example, to the power required for neutral beam injection heating. Clearly, the best case is to utilize Al at cryogenic temperatures rather than water cooled copper. Thus, it may be quite feasible to utilize superconducting, steady state TF coils and normal, pulsed OH and VF coils on the next several generations of tokamak reactors. The ultimate use of normal pulsed coils on commercial sized power reactors will depend on further optimization to minimize cooling power requirements and the ultimate size of such systems.

d. Low Activity Shields and Magnet Systems

A design advantage on both near term experimental reactors and commercial power reactors would be to have a low activity shield and magnet system that would allow ready access and maintenance to systems outside the main vacuum chamber and blanket regions. Such an approach can be accomplished by constructing the last 20-40 cm of shielding with aluminum alloy structure, lead tungsten, and boron carbide. The Pb, W and B_4C would be used in readily available nonstructural forms. The advantages of aluminum, carbon, and boron in low activity applications has been widely studied.[41, 42, 20] Tungsten does not lead to low radioactivity levels, but the primary long lived activity, ^{183}W, decays by β-decay so that dose effects are minimal with any surrounding structure.

Such a shield can be combined with the use of superconducting TF coils in which Al is used as the stabilizer and an Al alloy, such as 2219, is used as the magnet structure. This approach was studied extensively for UWMAK-III.[32, 33] The advantages are that Al is a better conductor than Cu at 4.2°K, the magnet system is less expensive, and the conductor and structure have a common coefficient of thermal expansion. The experimental observation[43] which appears to make this approach possible is that high purity aluminum stabilizers reinforced with aluminum alloy structure can be cyclically strained to a strain level of 0.0038 with little degradation in the resistivity ratio of the Al.

One disadvantage of an aluminum alloy structure is that it must be operated at somewhat lower stress levels compared to stainless steel (30 ksi versus 45 ksi). However, the potential advantages appear to be overriding. Combined with a low activity outer shield, the time delay before one can gain access would be determined by the decay time of the activated aluminum which is known to be very short.[41]

CLOSING COMMENTS

Tokamak fusion power reactor design is in a high state of flux, as it should be for a field so young and open to innovation and imagination. New experimental results spur design innovation while design results based on conservative assumptions spur the experimental search for more information and better future designs. This is in keeping with the past and future role of conceptual reactor design, namely, to interact with experimental research in fusion technology and to act as a vehicle for testing the impact of new information and new approaches to design. Only in this way can the improvements required to convert tokamaks into a viable future power source be realized.

ACKNOWLEDGEMENTS

I thank Dr. D. K. Sze and Dr. S. Hong for help with some of the calculations. This research was supported by grants from the Division of Magnetic Fusion Energy, ERDA, the Electric Power Research Institute, and the Wisconsin Electric Utilities Research Foundation.

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TABLE 3

Power Requirements for Normal Pulsed Ohmic Heating and Vertical Field Coils (MW)

OH Coils

Conductor	TETR(40)	EPR(23)	UWMAK-III(32)
Al at 4.2°K	10	2	152
Cu at 300°K	21	4.3	322

VF Coils

Conductor	TETR(40)	EPR(23)	UWMAK-III(32)
Al at 4.2°K	32	42	800
Cu at 300°K	70	92	1400