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MAGNETIC FUSION COMMERCIAL POWER PLANTS

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Magnetic Fusion Commercial Power Plants

Presentation to Fusion Power Associates Annual Meeting and Symposium

John Sheffield, Oak Ridge National Laboratory

1. Introduction

Toroidal magnetic systems present the best opportunity to make a commercial fusion power plant. They offer potential solutions to the main requirements that confront a power plant designer. These requirements are:

- having a low transport rate of heat and of the deuterium-tritium (D-T) fuel;
- having a high enough transport rate of helium ash to minimize dilution of the fuel;
- providing a divertor system to remove helium efficiently and minimize the generation and input of wall-generated impurities;
- operating in steady state with no plasma disruptions, and with a low level of recirculating power to sustain the plasma;
- having a cost effective "simple" magnetic coil system with, ideally, no trapped coils and good access for maintenance and repair; and
- having a high enough beta that with the simple coils the system has a high enough power density to approach the ideal cost of electricity (COE) limit.

An ideal system may be postulated^{1,2,3} in which the coils are a very small part of the cost, and the cost stems primarily from the inescapable components: minimal plasma heating (and sustaining system), tritium breeding blanket, shield, particle input, removal and treatment system, heat transfer system, generators, buildings, and balance of plant.

No present system meets the ideal standards; however, toroidal systems contain among them the elements required. Consequently, a logical program may be based upon an evolutionary development, building on the contributions of the tokamak, which has been the mainline of research for a number of years.

2. Tokamaks

The tokamak has received the highest funding; consequently, it is the most studied, the most developed, and the most credible approach, at present, to making a fusion power plant. Success in the tokamak program has led to the initiation of the International Thermonuclear Experimental Reactor (ITER) project.⁴ The ITER engineering design activity is under way as a four-party venture — European Economic Community, Japan, Russia, and the United States — under the auspices of the International Atomic Energy Agency (Figure 1). The ITER program forces fusion research to face up to the realities of magnetic fusion with an extended pulse burning plasma. The studies highlight the problem areas and show clearly where the tokamak and toroidal systems must improve in regard to the requirements listed above.

The average COE of present fission power plants is ~ 60 mills/kWh for an ~ 1000-MW(e) plant.⁵ In contrast, an ideal fusion power plant might have a COE of $\lesssim 40$ mills/kWh. This value is not strongly dependent on plant size because the COE of the blanket and fuel cycle and shield are nearly independent of plant size and power density, and other basic components show a weak dependence on power ($\sim P^{0.5}$). However, such an ideal system has not been conceived yet.

A tokamak power plant based upon ITER technology and physics, but with advanced reactor-relevant first wall and blanket, and costed on the same basis as fission systems, would have a

noncompetitive COE ~ 130 mills/kWh for an ~ 1000 -MW(e) power plant, and COE ~ 110 mills/kWh for an ~ 2000 -MW(e) power plant.

Fortunately, there are avenues to improve the technology, reduce unit costs, and improve the physics, as discussed in the ARIES reactor studies.⁶ The advanced, neutron-resistant, low-activation materials for the first wall and blanket are being developed in parallel with ITER and will be tested in later phases of ITER. Improvements in the technologies may be expected as research continues and experience is gained in ITER and other facilities. Advances in tokamak physics will be tested in the Toroidal Physics Experiment (TPX),⁷ which will be built and operated as a national facility at the Princeton Plasma Physics Laboratory.

Based on the expected TPX operating scenarios, improved technology and moderately lower unit costs, the parameters of advanced tokamak power plants from evolving the ITER design and technology approach and from the ARIES studies, are given in Table 1.

Table 1. Parameters expected of tokamak power plants derived from ITER and from the ARIES study

Major radius (m)	5.5 - 6.5
Minor radius (m)	1.5 - 2
Plasma elongation	~ 2
Toroidal field (T)	7.5 - 4.5
Maximum coil field (T)	16 - 11
Plasma current (MA)	6.5 - 13
Net electric power [MW(e)]	1000 - 2000
COE (mills/kWh)	75 - 65

It is hoped that effective cost savings relative to fission will accrue from the better safety and waste characteristics of fusion, thus improving the attractiveness of fusion. A key assumption made in calculating potential COEs for fusion power plants is that a good capacity factor, $\sim 75\%$, will be obtained. This is probably the greatest uncertainty in projecting fusion costs because it implies a level of component and operational reliability and ease of maintenance that is not demonstrated. Operation of ITER and TPX will help bring reality to this area.

3. Complementary Systems

A number of approaches exist to improve the toroidal system, including further variants of the tokamak, such as the spherical torus, and complementary systems:⁸ the stellarator, reversed-field pinch (RFP), field-reversed configuration (FRC), and spheromak (Figure 2). Each offers potential advantages, but some aspects are not demonstrated to be as good as in the tokamak.

One approach to lowering the COE, which may be possible in some of the complementary systems, is to operate at higher power density; that is, a higher neutron and heat flux to the first wall blanket and divertor. How far it will be possible to move in this direction depends critically on developing an effective divertor system and first wall and blanket materials. To put this issue in perspective, note that ITER will operate with an average neutron flux to the first wall $\bar{p}_{wn} \sim 1 \text{ MW}\cdot\text{m}^{-2}$; most recent fusion plant designs have $\bar{p}_{wn} \sim 2.5$ to $5 \text{ MW}\cdot\text{m}^{-2}$; and analysis of inherent safety in the face of loss-of-coolant accidents⁹ suggests that safe operation may require a limit of 6 to $10 \text{ MW}\cdot\text{m}^{-2}$.

It should also be remembered that there is a simple connection between the average neutron flux $\bar{p}_{wn}$, the beta and field required $\langle \beta \rangle B^2$, the power per unit length P_e/R , and the required thermal diffusivity χ_E ,⁸ as illustrated in Figure 3 for a simple model toroidal D-T power plant operated at $100 \text{ kW(e)}/\text{tonne}$ of the fusion island.

The complementary toroidal configurations offer potential advantages over the advanced tokamak and hence, incremental cost advantages:

- inherently steady-state — stellarator,
- lower recirculating power — stellarator,
- simpler divertor — FRC (spheromak),
- better access — spherical torus, FRC, spheromak, and
- lighter coils (higher power density) — spherical torus, RFP, FRC, spheromak.

More detail about reactor features is given in Table 2.

Table 2. Reactor features

AREA	SPHERICAL TOKAMAK	ADVANCED STELLARATOR	REVERSED-FIELD PINCH	FIELD-REVERSED CONFIGURATION	SPHEROMAK
Transport	OK $H \sim 2$	OK at large size. Need E-field in torsatron	Connor-Taylor or Carreras-Diamond scaling?	Need 10X better than present scaling?	Need 10X better than present scaling. May be possible?
Helium Removal	Consistency with H-mode?	E-field effects. Direct losses can help.	?	OK?	OK?
Divertor	ITER-like. Large radial distance	ITER-like needed?	?	Axial good	Axial good?
Steady State	Efficiency? Disruptions?	Good	$F - \Theta$	Proton - driven in $D-^3He$?	Helicity injection
Beta	Good	$\lesssim 5\%$ OK experiments needed	$\beta\Theta \approx 0.2$	Good tilt mode	OK tilt mode
Access	(a)Very good	Issue for design optimization	(a)Fairly good. Conducting shell!	(a)Very good	(a)Very good

(a) For all systems with plasma currents, there is an issue for blanket design and need for good electrical connections, particularly if major disruptions are an issue.

4. Summary

The ideal toroidal system has not been realized in conceptual toroidal power plants. Yet, the potential for its realization seems to exist among the capabilities of various toroidal configurations. A sensible goal for magnetic fusion should be to have a plan that allows evolution from the tokamak, as we see it today, toward an advanced toroidal system that incrementally approaches the ideal system and has:

- improved transport, coupled with rapid helium ash removal;
- improved heat distribution and a simplified divertor;
- a reduction of the recirculating power fraction;
- higher beta - lighter, simpler coils;
- neutron-resistant, low-activation materials with good thermal conductivity;
- moderately higher power density consistent with the improved materials; and
- improved access, maintainability, and reliability.

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Figure 1. ITER Configuration 1994

Figure 2. Representative toroidal systems with nested flux surfaces; stellarator, tokamak, reversed-field pinch, spheromak, field-reversed configuration, and dense Z-pinch.

Figure 3. Coupled values of $\langle \beta \rangle B^2$, average neutron flux to the first wall $\bar{p}_{wn}$, P_e/R , and χ_E , shown as a function of minor radius (a), for a self-sustaining D-T plasma in the simplified toroidal reactor model, operated at 100 kW(e) per tonne.