LA-UR- 85-1035

CONF-850808--3

LA-UR--85-1035

DE85 009622

TITLE: ALTERNATIVE FUSION CONCEPTS AND THE PROSPECTS FOR IMPROVED REACTORS

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SUBMITTED TO 20th Intersociety Energy Conversion Engineering Conference (IECEC)
Miami Beach, Florida
August 18-23, 1985

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Alternative Fusion Concepts and The Prospects for Improved Reactors*

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ABSTRACT

Past trends, present status, and future directions in the search for an improved fusion reactor are reviewed, and promising options available to both the principle tokamak and other supporting concept are summarized.

IN TRACING THE EVOLUTION of conceptual fusion reactor designs, the reduction of the dominance of the fusion power core (FPC, i.e., plasma chamber, first wall, blanket, shield, coil, and related structure) and associated reactor plant equipment emerges as a central theme. Reduction of the FPC size and increase in the power generated per unit FPC mass are the most visible but not sole elements of an improved fusion reactor, however. A summary of past progress, present status, and future directions for reactor improvements is first given. An overview of major options for improved confinement systems is then presented, with poloidal-field-dominated (PFD) systems appearing particularly well suited in meeting goals for improved fusion reactors. The experimental status and reactor prognoses for major PFD systems--the reversed-field pinch (RFP) and the spheromak compact toroid (CT)--is then given. The spherical torus (ST) tokamak is also included as a PFD-like variant of the tokamak. Brief conclusions note the rich and inter-related ensemble of fusion options that promise a decreased FPC role in the overall cost equation while simultaneously allowing a more flexible and cost-effective development path. Systems that are amenable to factory-constructed FPCs of moderate size and capacity while operating within a reduced nuclear envelope emerge as an attractive goal for fusion power.

*Work performed under the auspices of US Department of Energy, Office of Fusion Energy.

DIRECTIONS FOR IMPROVEMENT

The general perception that magnetically confined fusion projects a central electric power station that may be too large, complex, and costly, while requiring a development path that is too intensive of time and budget, has led to a reexamination of goals and the charting of new direction for both the principal tokamak(1)* as well as other fusion concepts.(2,3) Continued progress towards an optimal end-product for fusion is most directly illustrated in Fig. 1., which for a range of conceptual fusion reactor designs gives the FPC mass utilization (useful thermal power divided by the FPC mass) and the FPC power density, lines of average FPC density and the engineering power density for a range of fissile and fossil energy sources are also given. Progress towards an optimal fusion energy source is reflected by the relative positions of conceptual design for the early UWMAK-I tokamak,(4) the later STARFIRE tokamak,(5) and the GENEROMAK, which is an economic target recently suggested(20) for fusion in general. Ongoing reexamination(1) of the tokamak option generally projects optimal designs that fall within the GENEROMAK range, whereas recent studies of other promising approaches based on compact spheromak (CSR)(19) and reversed-field pinch (CRFPR)(9-12) reactors project FPC performance, as measured on Fig. 1., not unlike that for light-water fission reactors (PWR).

Although the Fig. 1 design-point summary is useful for monitoring progress and projecting goals, a more detailed analysis of physics and technology constraints and the associated tradeoffs related to development cost and time, end-product operational and cost issues, and general safety and resource concerns is required to define both the attractiveness and competitiveness of fusion power. Hence, in

*Numbers in parentheses designate references at end of paper.

addition to increased FPC mass utilization, the goals for improved fusion reactors to varying degrees are determined by the following items:

- Potential for reduced total power output and associated capital investment, with the possibility of multiplexing a number of smaller FPCs to drive a larger total site capacity.
- Emphasis and/or enhancement of passive safety (against a loss of coolant) through inherent FPC design characteristics.(30)
- Stress long-pulsed or steady-state plasma operation while addressing related issues of plasma current drive, heating, fueling, and impurity/ash control.
- Simplify the FPC design in terms of reduced fields, stresses, and stored (magnetic) energy while using advanced materials and/or fabrication techniques only where clear-cut advantages are perceived.
- Maintain a high overall plant efficiency by utilizing direct energy conversion (when possible), high coolant fluid temperatures, and minimum power recirculated to the FPC and associated support systems (i.e., coils, current drive, plasma heaters, coolant pumps).
- Emphasize physically small modular FPCs that assure a flexible development path and ultimately factory (off-site) fabrication, full non-nuclear FPC pre-testing, and single- or few-piece FPC maintenance and repair.

Although many of the technologies that determine the implementation of these goals, and, hence, the achievement of the GENEROMAK threshold target, are generic to magnetic fusion, details of the plasma confinement physics play an important role in determining both the interaction and required advancement of these technologies. Central to this coupling of technology with physics is the plasma pressure relative to the magnetic-field pressure evaluated either at the plasma (physics beta, β) or at the magnet coil (engineering beta, β_E). Although most costing models used to evaluate fusion prospects predict a weak cost-of-electricity (COE) dependence on beta for $\beta > 0.05$, the subtle but important impact of this parameter on magnet technology (stress, superconducting versus resistive conductors), structural requirements (stored energy, stress, forces), and blanket choice, (high-power-density liquid-metal breeder/coolants versus other separate-function combinations) is significant. Higher beta values, therefore, can open design windows and options that increase the end-product credibility, while not necessarily strongly impacting the COE, as presently computed.

The means by which β and β_E can be increased and the associated impact on FPC geometry and technology, therefore, is central to the goals for an improved fusion reactor. For a tokamak of minor plasma radius r_p, major radius R_T, toroidal current I_ϕ, and toroidal field B_ϕ, stability-related beta limits generally scale as $\beta \sim \varepsilon/q \sim I_\phi/r_p B_\phi$, with increased beta for a given safety factor, $q = \varepsilon B_\phi/B_\theta$, implying small aspect-ratio, $1/\varepsilon = R_T/r_p$, and high-current plasmas.(31) Increased elongation of the plasma minor cross-

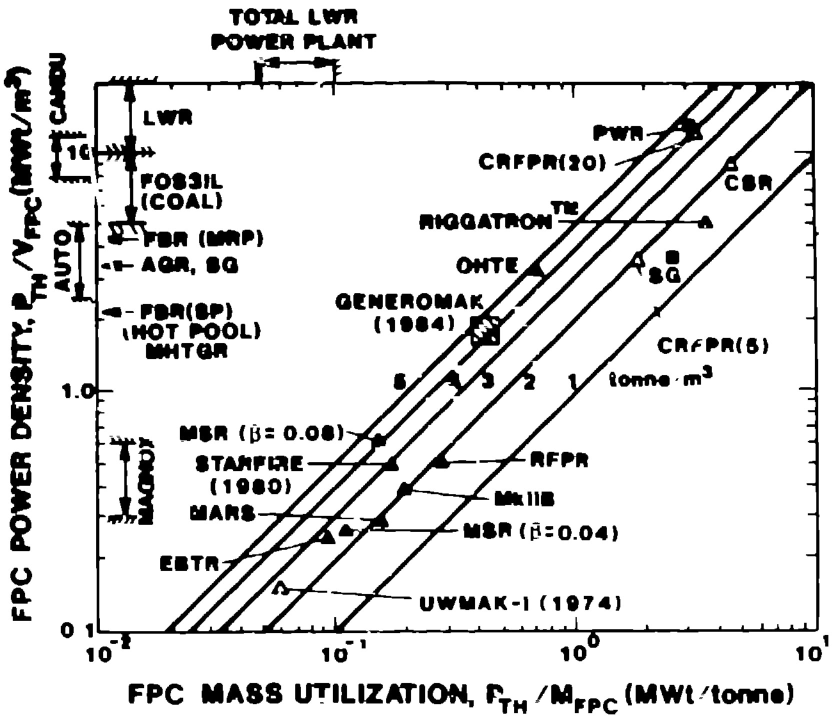

Fig. 1. Summary of FPC on engineering power densities projected for recent fusion reactor designs: The early University of Wisconsin UWMAK-I tokamak(4); STARFIRE tokamak(5); Culham MkIIB tokamak(6); superconducting Reversed-Field Pinch Reactor, RFPR(7,8), Compact Reversed-Field Pinch Reactor, CRFPR(20)(9,10) and CRFPR(5)(11,12), Los Alamos Modular Stellarator Reactor, MSR(13); University of Wisconsin Modular Stellarator Reactor, UWTOR-M(14), ELMO Bumpy Torus Reactor, EBTR(15); Mirror Advanced Reactor Study, MARS(16); reactor based on Ohmically-Heated Toroidal Experiment, OHTE(17); Riggatron™ tokamak(18), compact spheromak reactor, CSR(19); generic fusion reactor, GENEROMAK(20). The engineering power density for a number of fusion reactor systems are shown, where the volumes are defined by the primary pressure vessel: Pressurized-Water Reactor, PWR(21,22); steam generator for a PWR, SG(22); Superphenix Fast Breeder Reactor, FBR(SP)(23), Advanced Gas-Cooled Reactor, AGR(24); Magnox Gas-Cooled Reactor, MAGNOX(25), Modular Fast Breeder Reactor, FBR(MRP)(26), Modular High-Temperature Gas Reactor, MHTGR(27), Pressurized Heavy-Water Reactor, CANDU(28). The cyclone furnace used in a modern coal-fired plant is also shown(29). Except for the modular fission plants, most systems are in the ~1000–1200 MWe(net) class.

section also increases(32) I_ϕ and β. The increased plasma current, however, may make more difficult plasma startup, current-drive, and/or disruption control,(1) and beta-enhancement schemes based on plasma-shape (inboard indentation) and current-profile control in larger-aspect-ratio, lower-I_ϕ tokamaks are also receiving attention(33). By plasma indenting and profile control, ballooning instabilities that now limit the tokamak beta (through plasma disruption) can theoretically be removed, and a second stability region (SSR) of high-beta (> 0.15) operation is predicted. The indenting or "pusher" coil as well as the feasibility of and degree to which current profiles must be controlled, however, present concerns for the SSR tokamaks.

While SSR tokamaks may allow stability against ideal ballooning and internal kink modes, possibly requiring an electrically conducting first-wall shell to stabilize external or surface kinks, the RFP admits a large number of internal kink modes into the plasma by decreasing the safety factor q to below unity on the plasma axis and allowing even lower values near the plasma edge, where q reverses sign. The resulting high edge-plasma magnetic shear possibly coupled with some form of short-term first-wall stabilization of external (surface) kink modes gives a high-beta plasma configuration that resides stably in a near-minimum-energy state.

The spheromak in terms of q-profile and number of modes admitted into the torus resides between the RFP and conventional tokamak, with typically the on-axis q being ~ 0.7 and decreasing to below ~ 0.5 at the geometric axis. This q profile is steepened when current is driven on open poloidal flux surfaces, but nevertheless only relatively long wavelength modes are admitted to the spheromak compared to the RFP. Unlike the RFP or tokamak, the high-beta spheromak has no external conductor or other structure linking the torus, and a uniquely attractive plasma geometry results.

The poloidal-field-dominated (PFD) plasmas offer unique routes to achieving the aforementioned goals for significantly improved fusion reactors. While the RFP and spheromak are truly PFD systems, the tokamak in the form of a low-aspect-ratio (ϵ = 0.6-0.7) spherical torus(31) also has a strong poloidal field at the outboard equitorial plane, and to some degree may share some of the attributes of these high-beta PFD systems while operating within the q > 2-3 tokamak constraints.

SPECIFIC APPROACHES

OVERVIEW OF OPTIONS – Figure 2. depicts key classes of magnetic confinement systems presently under study. This chart is organized to emphasize approximate relationships between concepts, with systems supporting large plasma currents positioned on the left, while those containing little or no plasma current being positioned on the right. The latter systems are dominated by externally imposed axial or toroidal magnetic fields (including the "conventional" tokamak and the tandem mirror) and, therefore, generally require large superconducting coils. Confinement systems located on the left side of Fig. 2 support more of the plasma pressure by internal plasma currents, have reduced requirements for externally imposed magnetic fields, and to varying degrees can operate with efficient resistive (copper or aluminum alloys) coils; these PFD concepts require minimal blanket/shell requirements compared to superconducting systems, and a considerable reduction in the FPC mass, size, and complexity is envisaged.(2,3)

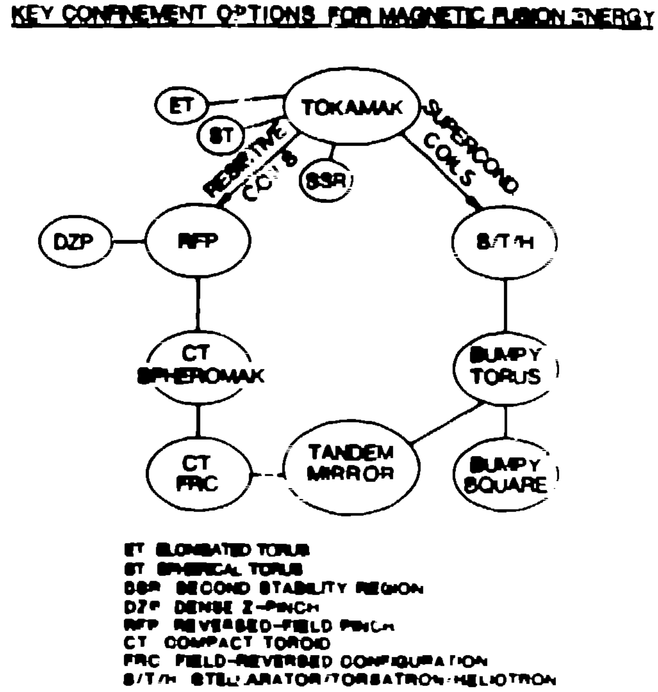

Fig. 2. Options for magnetic fusion. The higher-beta options for the tokamak include the spherical torus, ST(31); the elongated torus, ET(32); and operation in the second stability region, SSR(33). The stellarator/torsatron/heliotron is grouped as S/T/H(13,34). As for the S/T/H, the bumpy torus(15) can be viewed in terms of plasma confinement on drift surfaces, this usually large system projecting compactness when formed into a square or high-order polyhedron(36). The reversed-field pinch, RFP(9,12), is the first significant step away from the "conventional" tokamak as a PFD system. The Dense Z-Pinch, DZP,(40) and compact toroid (CT) spheromak(19) have no toroidal field outside the plasma. The field-reversed configuration, FRC(38), is a CT with no toroidal field, either inside or outside the plasma. The tandem mirror(16,37) embodies characteristics of both FRCs, S/T/Hs, and bumpy torus squares, including the use of high-field superconducting and resistive coils, drift surfaces, energetic electron rings, and linear central geometry.

To varying degrees, the advanced tokamaks (i.e., ST, BT, SSR) attempt to enter this PFD region, with the efficient use of resistive copper coils to confine higher-beta, higher-power-density plasmas also promising dramatic reductions in FPC size, mass, complexity, and cost. Although the tokamak physics data base far outstrips many of the other approaches listed on Fig. 2., the advanced tokamak embodiments must project significantly from present understanding into regions where other concepts have equal if not stronger data bases. This situation coupled with: a) the ability to transfer physics understanding across concept boundaries, b) strong experimental successes, particularly for RFPs, spheromak CTs, and FRC CTs, and c) the possibility to extend these latter concepts to viable commercial end-products without ever-increasing plasma sizes all combine to point to significant improvements in systems located on the PFD (left) side of Fig. 2. For these reasons the prognoses for improved reactors is based on the RFP, spheromak, and the ST tokamak.

Improvements in the toroidal-field-dominated systems have also been suggested,(35-37) although these systems would exploit advantages that do not necessarily include reduced FPC cost and size (i.e., inherent safety accompanying low-power-density plasma operation, intrinsically steady-state plasmas, high-Q ignition, natural divertor and separation of DT neutron and charged-particle fluxes, and linear geometry).

It is noted that the FRC(38,39) and DZP(40) are fully contained by poloidal field (i.e., toroidal or axial plasma currents) and present reactor extrapolations that are unique and interesting in their own right. Considerations of stability, however, generally dictate pulsed operation for these systems, with moderately comprehensive reactor studies being available only for stationary(39) or translating(40) FRCs.

POLOIDAL-FIELD-DOMINATED (PFD) SYSTEMS - The physics and reactor status of both the RFP(41,12) and the stationary (gun-produced and sustained) spheromak(42,19) have been recently reviewed and are summarized here as being prototypical PFD systems. The ST tokamak,(31) while not a PFD system, is nevertheless summarized here as a tokamak option that may also share some reactor benefits of operating in this regime.

Reversed-Field Pinch (RFP) - The RFP is emerging as an attractive reactor concept because of encouraging physics results from a number of experiments and because of inherent properties that promise compact, high-power-density reactors.(9-12) As for the tokamak, the poloidal field, B_θ, in the RFP is generated by toroidal plasma currents, I_ϕ, but the toroidal field, B_ϕ within the plasma is comparable to B_θ and decreases through zero to a small negative value (hence, the name RFP) outside the plasma. The RFP engineering features, therefore, are dominated by the poloidal field, which decreases from the plasma inversely with distance to the

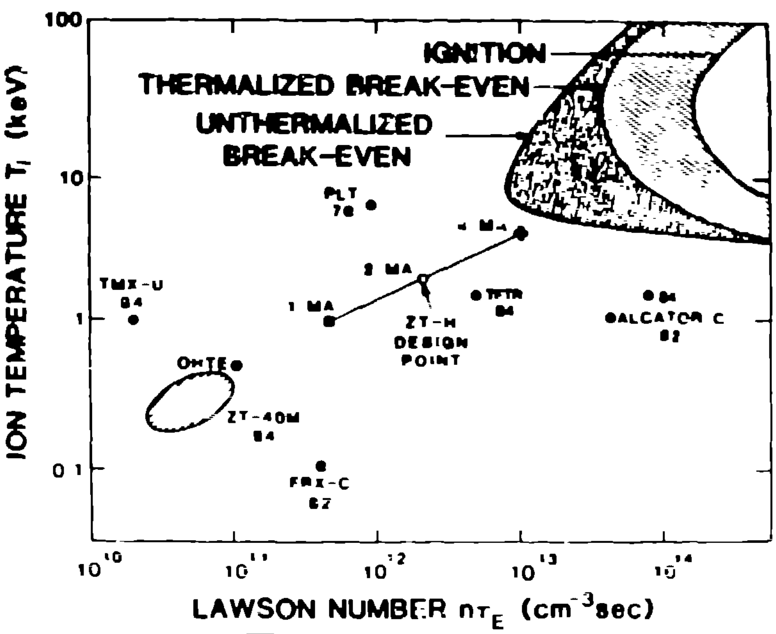

Fig. 3. Comparison of past and projected physics performance of the principle and other confinement concepts given on Fig. 2.

poloidal coils. The resulting high-beta plasma is particularly amenable to confinement by low-field (low-current) copper-alloy coils that can be separated from the plasma by the minimum thickness (0.5-0.7 m) required for a blanket to breed tritium and to recover the fusion energy. the absence of thick shields required of superconductors considerably reduces the mass of both coil and blanket/shield systems, projecting 1000-2000 tonne FPCs rather than 20,000-30,000 tonne units envisaged for superconducting reactors of comparable power output. In addition to operating with plasma current densities that are sufficient for ohmic heating the plasma to ignition, considerably simplifying an otherwise major complexity for fusion, the close coupling of toroidal and poloidal currents (fields) in the near-minimum-energy RFP plasma promises a unique ability to rectify externally applied voltage oscillations and to drive the plasma current with no net change in the poloidal flux linking the torus, the very mechanism that sustains the q << 1 RFP configuration promises a means for low-frequency, low-power current drive.

These ideas form the basis of the compact RFP reactor(9-12) and have received considerable substantiation by modern RFP experiments and theory.(41) This progress is occuring in five RFP areas, plasma setup/formation, plasma sustainment/dynamo, confinement, plasma-wall interaction, and steady-state current drive. The RFP field configuration (i.e., strong on-axis toroidal field of magnitude comparable to the poloidal field and decreasing to small, reversed value outside the plasma) would on the basis of classical resistive theory be expected to decay to a uniform field profile of magnitude equal to that imposed on the external toroidal-field coils. The RFP configuration can be shown,(43) however, to represent a near-minimum-energy state, and plasma relaxation through an internal plasma dynamo action maintains a net poloidal plasma current and associated toroidal field, using the poloidal-field circiut as an energy supply. Hence, once slowly formed, the RFP configuration can be maintained as long as toroidal voltage is applied. Furthermore, sustainment through the dynamo effect can be used to ramp slowly the toroidal current and create internal toroidal flux; the dynamo action to date has more than tripled the toroidal flux during such slow current ramps.(41)

Given the ability to form slowly and then to sustain the RFP configuration, the transport properties in these high-beta, ohmically heated discharges and the scaling to the reactor regime become of paramount interest. Modern RFP experiments show a scaling of plasma pressure with the square of the plasma current, indicating that the plasma beta is constant (i.e., $2nk_BT = \beta_\theta B_\theta^2/2\mu_0 = \beta_\theta I_\phi^2/r_p^2$). This constant-beta scaling has been predicted theoretically(44) and suggests an energy confinement time, $\tau_E \sim f(\beta_\theta) r_p^\lambda I_\phi^\nu$, where $\nu = 1-1.5$. The higher value of ν results from classical scaling of plasma resistivity with temperature ($\eta = Z_{eff}/T^{3/2}$, as is observed experimentally), the assumption of constant β_θ, and the constraint that $I_\phi/\pi r_p^2 n$ (an electron streaming velocity) is constant. Figure 3. shows the extension of this experimentally verified scaling towards the next step in the RFP program, ZT-H,(41) DT ignition, and reactor conditions. Specifically, the observed scaling predicts for the Lawson parameter, $n\tau_E = \beta_\theta I_\phi^{5/2}/Z_{eff}$, and is approximately independent of size. Approaches to ignition and burn emerge that stress plasma current and select plasma size only to meet engineering heat-transfer and plasma-wall-interaction constraints (i.e., $n\tau_E = 1/Z_{eff}$), these approaches generally favor less-expensive, low-technology, smaller, and more flexible systems along the development path to commercial fusion.

Plasma performance and the technology of the plasma-wall interaction soon become linked for the RFP. Control of local field errors in these smaller experiments become crucial to attaining high current and good confinement at high temperature in small systems. The role of gas recycle from the vacuum wall is important. Low-Z limiters are only beginning to be used, and the ability to maintain the discharge density (i.e., control plasma "pump out") by proper pre-discharge wall conditioning is being developed successfully.(41) Progress along the $n\tau$–T trajectory displayed in Fig. 3. and the size and cost of the required devices will be determined in large part by an evolving understanding and control of this plasma-wall interaction, first by the implementation of passive limiters (for short-pulse operation) and then by use of either (poloidal) arrays of pumped limiters(9) or toroidal-field magnetic divertors.(45)

The strong, non-linear coupling of the toroidal and poloidal circuits through the near-minimum-energy, continually relaxing RFP promises a non-invasive means to drive steady-state current by low-frequency (few Hz at reactor conditions) oscillation of the B_ϕ and B_θ circuits.(46) By proper phasing of the toroidal and poloidal voltages applied to the RFP, poloidal and toroidal fluxes can be oscillated with strong linkage (i.e., helicity, $K = \int_V \mathbf{A}\cdot\mathbf{B}dV_p$ where $\mathbf{B} = \nabla\times\mathbf{A}$) into the plasma on one half cycle and removed with less flux linkage (or helicity) on the following half cycle. Given that anomolous flux absorption through the plasma dynamo occurs, a means to inject net magnetic helicity into the plasma as it is resistively consumed provides for a low-technology method to sustain the plasma current indefinitely. Partial tests of this oscillating field or "F-Θ pumping" current drive have been successful,(41) but full tests must await hotter, less-resistive plasma. Current drive by electrostatic (i.e., dc) injection of magnetic helicity, rather than the low-frequency electromagnetic pumping described above, may also be possible by proper arrangement of electrodes in the plasma scrapeoff, this method being suggested for the gun-sustained spheromak.(19,42)

Spheromak Compact Torus (CT) - A CT is an axisymmetric torus that has no magnet coils, conducting walls, or vacuum surfaces linking the torus. With only poloidal field and in an elongated (prolate) form required for stability, the high-beta (0.8-1.0) FRC results. The spheromak is a CT with both B_θ and B_ϕ fields, and, like the RFP, both are comparable in magnitude and generally configured as described by the Taylor near-minimum-energy state.(43) Spheromaks have been generated using magnetized co-axial plasma guns [CTX,(42,47) BETA-II(48), combined fast-pulsed Z- and Θ-pinch techniques (PS-1),(49) and electrodeless flux-core formation techniques (S-1),(50)]. Reactor projections have been made for spheromaks formed by flux-core(51) and magnitized-gun(19) techniques.

In addition to the attributes of strong ohmic heating, high plasma and even higher engineering beta, and the efficient use of resistive (equilibrium) coils to give a high-mass-utilization FPC, the simply connected CT magnetic geometry gives added simplification and reduces even further the impact of the FPC and the overall cost equation for fusion. Formation techniques based on a magnetized co-axial electrode also gives the added promise for an ex-reactor divertor for impurity control as well as the proper arrangement of electrodes to inject magnetic helicity with an externally applied dc voltage, dc current drive through electrodes immersed only in the plasma scrapeoff become possible. Hence, toroidal flux emerging from the magnetized gun electrodes links a small fraction of poloidal flux at the outer flux surfaces and helicity is injected at a rate required to sustain the plasma against resistive decay as well as supplying power losses incurred in the divertor and the edge-plasma regions. Experimental evidence has been reported for such sustainment over times that are ten times the magnet-energy decay time.(42) Generally, present-day electrode-sustained spheromaks have a higher impurity content and poorer confinement, operation in the detached or separated mode giving the better results (~100 eV, $n\tau_E = 4(10)^{15}$ s/m^3) for these relatively small ($R_T + r_y$ = 0.40-0.70 m) plasmas. The development of cleaner and more efficient (with respect to helicity injection) electrode systems represent key areas of research.(47) Generally, the spheromak represents a logical and highly attractive extension from the already promising reactor improvement project for the RFP.

Spherical Torus (ST) Tokamak - The plasma performance for the tokamak as measured by τ_E, β, and current drive, depends strongly on plasma shape [aspect ratio ($1/\epsilon$), elongation (κ), triangularity (d), and indentation (e)] and current profile. For a given value of q > 2-3, critical MHD beta limits increase with ϵ or I_ϕ, τ_E tends also to increase with I_ϕ, but high-frequency current drive expectedly becomes more power intensive at high values of I_ϕ. Coupled with the goal to reduce FPC size, mass, and cost by reducing the plasma major radius, the ST concept has emerged(31) with $1/\epsilon$ = 1.5-2.0, I_ϕ = 15-20 MA, q = 2.5, and β > 0.2. The ST reactor embodiment requires all structure except the toroidal-field-coil return conduction to be removed from the region inboard of the plasma. Conventional tokamak equilibrium causes a natural plasma elongation of κ = 1.5-2.0 for these low-aspect-ratio systems, and, although q = 2.5 on average, the poloidal field can be comparable to the toroidal field at the plasma outboard side, high-beta plasmas with reduced toroidal fields result.

Significant paramagnetism is also predicted for the ST configuration, wherein the on-axis toroidal field can exceed the vacuum field by a factor of 2-3. A curious tokamak configuration results that in shape outwardly resembles that of a spheromak with a hard-core conductor, exhibits a paramagnetism like that more strongly operative in RFPs and spheromaks, but is stabilized according to traditional tokamak lore (q > 2-3). The degree to which the ST, RFP, and spheromak have access to minimum-energy states and the character of the MHD processes that create channels for plasma relaxation are determined by the q(a) and $\mu = \mu_0 \vec{j}\cdot\vec{B}/B^2$ profiles, both of which differ considerably amongst these concepts.

Since room is not available at the inboard region of the ST for poloidal field solenoid, a non-inductive means is needed both to initiate and to drive the large toroidal current. While high-frequency waves may drive current in low-density plasma, the strong paramagnetism make tempting the postulate that oscillating-field or "F-Θ pumping" current drive may be applicable to the ST tokamak as well. For this current-drive mechanism to work, however, two basic premises must be fulfilled: a) $\mu = \mu_0 \vec{j}\cdot\vec{B}/B^2$ is nearly constant (a measure of the minimum-energy state), and b) a relatively quiescent relaxation process is available to provide the plasma a channel(s) for relaxing to the near-minimum-energy state after externally imposed perturbations without disrupting. Although the constancy of μ can be readily tested with an appropriate equilibrium code, the availability of non-disruptive relaxation channels for a q > 2-3 tokamak that in richness approaches the q << 1 RFP and the q < 1 spheromak remain as open questions. A partial indication is given by the spheromak, which is capable of non-disruptive relaxation through a small number of low-n (toroidal) modes. Answers to these questions are actively being developed for the ST as one of a number of significant tokamak improvement (Fig. 2.) that may lead to smaller reactors not unlike those suggested for PFD systems.

CONCLUSIONS

A number of options and opportunities exist for significant improvement in the prospects for commercial fusion power based on the principal tokamak as well as other concepts. The inter-relationship amongst the options are becoming clearer as physics understanding develops. The directions of significant improvement lead to systems that assume more of the task of plasma confinement, heating, and sustainment through self-generated fields rather than by imposing these functions exclusively on complex and costly engineering systems that surround a low-power-density plasma. Central to the needed reduction in FPC size, cost, and complexity is the use of efficient and closely coupled copper coils that ideally provide only an equilibrium function. Plasma systems that are poloidal-field dominated offer unique promise in this regard and may include tokamak variants. Although the tokamak physics data base is better developed than that for PFD systems like the RFP or spheromak, the degree to which these advanced tokamaks must extrapolate from that data base is not unlike that for the other approaches. Advances in these other concepts have been astounding, and the promise is great for development paths that alter considerably the previously assumed trend of ever-escalating device size and cost. A less costly but bolder and more-flexible development path to commercialization is anticipated for both these PFD systems as well as appropriately tailored variants of the principle tokamak.

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