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Advanced Spheromak Fusion Reactor

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ADVANCED SPHEROMAK FUSION REACTOR

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ABSTRACT

The spheromak has no toroidal magnetic field coils or other structure along its geometric axis, and is thus potentially more attractive than the leading magnetic fusion reactor concept, the tokamak. As a consequence of this and other attributes, the spheromak reactor may be compact, and produce a power density sufficiently high to warrant consideration of a liquid "blanket" that breeds tritium, converts neutron kinetic energy to heat, and protects the reactor vessel from severe neutron damage. However, the physics is more complex, so that considerable research is required to learn how to achieve the reactor potential. Critical physics problems and possible ways of solving them are described. The opportunities and issues associated with a possible liquid wall are considered to direct future research.

I. INTRODUCTION

The relative simplicity and potential low cost of electricity make the spheromak an attractive magnetic fusion energy concept. (1) The magnetic fields are generated primarily by plasma currents except for a vertical field, required to support the hoop stress, which is generated by a set of purely solenoidal coils. Current is driven by an external coaxial gun which generates linked toroidal and poloidal magnetic fluxes ("helicity") which drive current in the core of the spheromak through a magnetic dynamo. It is anticipated that the ohmic heating from this current will be sufficiently strong that no auxiliary heating source will be required to reach reactor conditions. (2) The result is a compact plasma, roughly spherical in shape, with a minimum number of auxiliary systems. We will discuss various features of a fusion reactor based on this configuration.

II. REACTOR CONFIGURATION

A conceptual spheromak reactor is shown in Fig. 1. The fusion plasma is confined within a magnetic separatrix with toroidal, magnetic flux surfaces; the geometry shown has two X-points. Helicity injection is from a coaxial, electrostatic gun located above the upper magnetic X-point in the example shown; power flowing from the confinement region is diverted into the upper and lower coaxial regions which act as divertors. It may be possible to unbalance the magnetic configuration so that most of the energy and particles flow to the lower region, thus separating the divertor from the coaxial gun and potentially improving the power and particle handling characteristics of the reactor.

Energy confinement in the plasma will determine the optimum relationship among density, temperature, magnetic field, and plasma dimensions. However, deductions from low temperature experiments extrapolate to an extremely optimistic value at reactor conditions, and it is highly likely that different physics will determine the actual confinement. Accordingly, we use a simple estimate for thermal power production to provide approximate dimensions, with the neutron power flux at the wall determining the approximate dimensions and parameters.

Assuming electron and ion temperatures $T_e = T_i = 20$ keV and equal deuterium and tritium densities, $n/2$, expressed in terms of β (the ratio of plasma and magnetic pressures), the deuterium-tritium power density, P_{DT}, at a magnetic field, B, is

$$p_{DT} = 1.14\beta^2 B^4 \quad \mathrm{MW/m^3}\,. \tag{1}$$

Taking a simple volume estimate $V = \pi\kappa R^3$ with wall (flux conserver) radius, R, and elongation $\kappa = 1.5$, then gives for the total fusion power production

$$P_{DT} = 5.4\, \beta^2 B^4 R^3 \quad \mathrm{MW}\,. \tag{2}$$

Estimating $\beta = 0.1$ (consistent with Sec. V) and 40% conversion efficiency to electrical power, we require $B^4R^3 = 4.6\times10^4$ $\mathrm{T^4m^3}$ for a 1 GWe reactor.

The average neutron wall loading follows as

$$p_w = 0.065\, P_{DT}/R^2 \tag{3}$$

Thus, for 1 GWe and $p_w = 8$ MW/m^2 we estimate $R = 4.5$ m. and $B = 4.5$ T, so $n = 1.4\times10^{20}$ m^{-3}. The actual operating point will undoubtedly differ from this when the energy confinement and wall loading limits are better understood. Hagenson and Krakowski assumed a wall loading of more than twice our value and consequently found a more compact reactor.

The blanket, located in the annular region surrounding the plasma, is envisioned to consist of liquid lithium or lithium salts, although solid or other configurations would also be possible. We will later discuss the possibility of having no wall between a flowing liquid blanket and the plasma, thereby greatly reducing the effects of neutron damage and possibly increasing the power density at the surface. Note that if ohmic ignition is possible, the spheromak will allow a blanket without radial port penetrations. Alternatively, Perkins (3) has suggested that the simplicity of the spheromak configuration could make a liquid/NaK potboiler reactor possible.

III. MAGNETIC COILS AND PLASMA CONFIGURATION

The hoop stress of the plasma current is supported by a solenoidal magnetic field which is curved to provide the detailed shape and elongation of the magnetic flux surfaces. Additional solenoidal coils are required in the coaxial gun to provide the poloidal flux, as discussed in Section VI. An example of a magnetic flux configuration is shown in Fig. 2. The zero flux surface, corresponding to the flux on the geometric axis, is indicated by the heavy line. A conducting wall lies on this surface to slow the response of global perturbations to the magnetic geometry.

IV. STABILITY CONTROL; FEEDBACK SYSTEMS

As noted in the previous section, the conducting wall will slow the response of the plasma to global perturbations. Of particular concern is the tilt/shift mode, is stabilized on the fast time-scale by this wall. On the resistive time scale of the wall, typically several 10s of milliseconds, wall currents will decay and the mode is expected to become unstable. There may be plasma effects which will stabilize the mode; for example in the tokamak plasma rotation is known to be strongly stabilizing for similar modes, although the detailed physics may differ for the spheromak. Thus, the conservative assumption given present knowledge is that a feedback system will be required. Thus eight feedback coils, four above and four below the midplane, are shown schematically in Fig. 1.

Although a detailed analysis of feedback for a spheromak has not been made, results from other devices provide confidence that stabilization can be provided. The tilt/shift mode is the plasma equivalent of the instability of a current-carrying ring supported in a magnetic field. Feedback systems were used in levitrons to control the ring position to fractions of a millimeter; (4-6) MHD modes have been stabilized in a tokamak (7) and in a reversed-field pinch in the presence of a resistive wall, (8) and the vertical (m=0) mode is routinely stabilized in shaped tokamaks. Demonstrating feedback control, if needed, will be an important part of an advanced spheromak experiment.

V. ENERGY CONFINEMENT

The magnetic properties of spheromaks are confirmed experimentally, but confinement time as reported in the literature has not appeared impressive, even taking into account the small size of experiments. Yet, by 1990 the CTX spheromak at the Los Alamos National Laboratory had produced T_e up to 400 eV with purely ohmic heating. (9) In Ref. 2, it is argued that these experimental results imply good confinement in the hot core of the plasma, comparable to early tokamaks and consistent with extrapolating to reactors such as the Hagenson and Krakowski design. (1) Ohmic heating alone may suffice to reach ignition in a spheromak, a great simplification in reactor design if true. Scenarios have also been developed (10) for simultaneously building up the field by helicity injection and the temperature by ohmic heating to achieve ignition, with energy transport rates calibrated to CTX.

Briefly, the argument in Ref. 2 notes that understanding energy confinement requires focusing primarily on the confinement of the energy of the plasma (the usual meaning of τ_E) and the energy stored in the magnetic field. These can be closely related, as discussed below. But because self-organization of the spheromak field continually regenerates field at the walls, any mechanism for destroying magnetic field near the wall is likely to dominate the magnetic field confinement time. This can mask good thermal energy confinement in the core if one examines only a global measure of confinement time determined from the magnetic decay time. Even in the optimum case, the low temperature of the plasma near the wall leads to much greater ohmic heating than in the plasma core. Nevertheless, all ohmic heating becomes irrelevant compared to alpha heating in the approach to ignition, so that the reactor power gain, Q, (fusion power divided by the power required to sustain the magnetic field) can be very large, about 100. (1)

VI. BETA LIMITS

Experiments have found that the plasma pressure can reach $\beta > 0.2$ before a major instability occurs. For a reactor model, we assume that β is limited by the Mercier criterion; the experiment reached a higher value than was estimated for this criterion, which is thus conservative. We have therefore extended previous calculations of the beta limits in spheromaks (11-15) to a geometry with a separatrix which couples to a plasma gun or other external helicity injector.

We examined the effect of the current profile, **j**, (and thus shear) by varying the current profile at fixed safety limit on the magnetic axis, q_0. We characterize the maximum beta by β_p defined as the volume-averaged pressure divided by the surface-averaged, squared poloidal magnetic field:

$$\beta_p \equiv 2\mu_0 \langle p \rangle_{vol} / \langle B_p^2 \rangle_{edge} \tag{4}$$

For the case in which there is a flux hole along the geometric axis without current through it, so that the edge safety factor is 0, we will find that the maximum Mercier-stable β_p is proportional to q_0. The cases with current through the hole will be seen independent of q_0.

We require $\lambda(\psi) \equiv \mu_0 \mathbf{j} \bullet \mathbf{B}/B^2$, with ψ the poloidal flux, to vary monotonically from the magnetic axis to the edge; calculations show that the maximum stable β_p occurs when $d\lambda/d\psi = 0$ on the axis and edge. Figure 3 shows the results of varying q_0 without current in the flux hole along the geometric axis, thus with no external magnetic field, and with current and thus external toroidal field. Clearly, the achievable beta is dependent on the current profile.

If there is current along the geometric axis, and thus a toroidal magnetic field on the separatrix, the safety factor will diverge there. The effect of this divergence is to generate a large shear on the plasma edge while reducing the shear in the volume. For the particular geometry of Fig. 2, with $\lambda =$ const. outside the separatrix, the result is to reduce the maximum beta and make it independent of q_0. The quantitative result is now dependent on the details of the flux-hole size as well as where external currents flow; clearly, further study of the beta-limit for this case is needed. Alternatively, if helicity injection can be achieved with no currents along the geometric axis, the beta can be maximized at a level consistent with the requirements of constructing a reactor of minimum size.

VII. CURRENT DRIVE AND THE COAXIAL PLASMA GUN

The spheromak configuration is close to a minimum energy state subject to the constraint of the conservation of helicity (16), K, a measure of linked poloidal and toroidal magnetic fluxes.

On the resistive time scale, helicity decays and must be replenished to sustain the configuration. The plasma gun is a means of injecting the helicity into the flux-conserving wall by means of a voltage, V_g, across a poloidal flux entering the volume from the gun, ψ_g, yielding a balance (17)

$$\frac{dK}{dt} = 2\left(V_g \psi_g - \int \eta \mathbf{j} \bullet \mathbf{B} dV\right) \tag{5}$$

where the integral is over the volume. In the approximation that λ is constant on the open flux, the injected current is $I_{inj} = \lambda \psi_g / \mu_0$, yielding a gun impedance of $\int \eta B^2 dV \big/ \psi_g^2$.

We estimate the gun parameters from the power injected, $V_g I_g$, which balances the net power losses, P_{DT}/Q. Consequently, for $Q = 100$ the injected power is 1% of the fusion power, about 250 MW in the present case. The magnetic field of 5 T with $R \approx 3$ m implies a plasma current of about I_p=60 MA; an amplification factor (I_p/I_g) of 60 will require a gun current of 1 MA at a voltage of 250 V. Estimating $\lambda = 5/R$, the gun flux is ~ 0.6 Weber, requiring a area of 0.1-0.2 m^2 and, at a circumference of $2\pi R$, a flux bundle width of a few cm. The open flux is thus small compared with the closed flux (~30 Weber), as required for the edge ohmic power losses to be small.

VIII. DIVERTOR

The magnetic geometry shown in Figs. 1 and 2 has a natural divertor. Although the poloidal configuration is similar to that of a tokamak, the distance along a field line to the divertor region is shorter because of the lack of external toroidal field. On the other hand, the system is effectively open as there is no interference from toroidal field coils, so that the plasma losses can be spread over as large an area as needed to handle power losses.

The fusion alpha particle power in a 1 MWe plant is about 500 MW; consequently, the power carried in the scrape-off layer is three times that injected by the gun. The edge plasma will thus be relatively, reducing the ohmic losses from the gun current. An increase in electron temperature from 50 eV to 200 eV decreases the ohmic edge losses by a factor of 8, so the additional heating is beneficial to the efficiency of helicity injection. Experiments will be required on injecting helicity in the presence of the plasma losses, but the effects of the fusion power should be positive for reactor operation primary system.

IX. BLANKET

The blanket in a spheromak reactor could be liquid lithium or liquid salts, that serve all the functions of a blanket in other fusion concepts (heat transfer, breeding) and shield the reactor vessel from severe neutron damage. Heat transfer by a flowing liquid wall that absorbs most of the neutron energy may also make possible much higher power density on the wall than would a conventional metal or ceramic wall dependent on coolants flowing through structures in and behind the wall. This duel possibility of neutron protection and high power density motivated the application of a liquid blanket for fusion reactors, first for the Astron magnetic concept, (18) and more recently for the HYLIFE inertial fusion design. (19) Protection from neutrons extends the life of reactor structures and reduces the fluence they receive, in some design studies to the point that components last for the life of the plant and allow "shallow burial" upon decommissioning according to present U.S. regulations. (20) The longer lifetime of components, together with higher power density, could significantly reduce the cost of electricity from fusion power plants. (21)

Several technical problems must be addressed to utilize liquid blankets in a spheromak. (22) (1) The flowing liquid must act like a conducting wall for the tilt/shift instability. There is even a possibility that the mode will be stabilized if there is shear in the velocity of the flowing lithium, as then there is no stationary wall frame for the mode to lock onto. (23) (2) Evaporated liquid must be efficiently ionized and transported away (along field lines) to prevent penetration into the burning plasma, and the ionized vapor must be collected and efficiently recycled without affecting the helicity drive. (3) Access must be available for entry and exit of the liquid and flow paths established that leave ample space for the plasma inside the protective liquid mantle. (4) The liquid medium must be capable of efficient heat transfer and adequate breeding of tritium. (5) The liquid blanket must not contain materials that exacerbate environmental and safety concerns. A liquid wall composed of non-flammable Flibe molten salt appears suitable for this. (22)

The choice between Flibe for a liquid blanket in spheromak reactors versus lithium, as originally proposed (for Astron) (18) may not be straight-forward. On the one hand, the potential severity of a fire with a pure lithium blanket makes non-inflammable Flibe very attractive, but there are also potential liabilities. Whereas neutron bombardment of lithium produces only tritium and harmless helium, the fluorine component of Flibe becomes highly activated, posing a safety problem in the event of an accident. (It is not a waste problem because of the short half-life.) Even a few percent release of fluorine is unacceptable. However, a strong argument is made in Ref. 20 that the afterheat in the Flibe itself is insufficient to cause the temperature rise required to release even this small quantity of fluorine, and bounding calculations show that contact of Flibe with the afterheat of structural elements cannot do so, either. (20) Perhaps the main factor in deciding between lithium and Flibe will be the added complexity of mechanical means of guiding the non-conducting Flibe through the machine (for example, spinning containers (22)) versus the fire hazard of lithium. Whereas controlling the flow of Flibe would require mechanical pumps and, perhaps, the centrifugal force of spinning containers, (22) it is at least conceivable that liquid lithium, which is electrically conducting, will be self-guided by the magnetic field, pushed along by electromagnetic pumps with no moving parts. Such a configuration might give rise quite naturally by creating the spheromak plasma inside a pre-existing liquid lithium blanket flowing along the periphery of the chamber, guided by a weak, externally produced "vertical" field already present. Then, during startup, as helicity injection pumps up the magnetic energy of the spheromak (and vertical field is programmed up accordingly), the spheromak and liquid layer interface would deform together, resembling the plasma-flux conserver interface in present-day experiments, each conforming to the evolving pattern of magnetic field lines. Especially if the liquid blanket allows a somewhat higher power density than we assumed in Section II, one can produce design cartoons packaging the entire spheromak-blanket system inside a vessel comparable in shape and dimensions to the corresponding reactor chambers in LWRTs or gas-cooled fission reactors.

X. ACKNOWLEDGMENTS

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REFERENCES

1. R. L. Hagenson and R. A. Krakowski, "The Spheromak as a Compact Fusion Reactor," Fusion Techn. **8**, 1606 (1985).
2. T. K. Fowler, J. S. Hardwick, and T. R. Jarboe, "On the Possibility of Ohmic Ignition in a Spheromak," Comments Plasma Phys. Controlled Fusion **16**, 91 (1994).
3. L. J. Perkins, private communication (1995).
4. O. A. Anderson, et al., ""Plasma Production and Heating in the Superconducting Levitron," *Plasma Physics and Controlled Nuclear Fusion Research 1971*, (IAEA, Vienna, 1971), Vol. I., p. 103.
5 R. Freeman, et al., "Confinement of Plasmas in the Spherator," *op cit.*, Vol. I., p. 27.
6. M. W. Alcock, et al., "Plasmas Produced by Neutral Injection in the Culham Levitron," *Plasma Physics and Controlled Nuclear Fusion Research 1974*, (IAEA, Vienna, 1974), Vol. II., p. 77.
7. A. W. Morris, et al., "Feedback Stabilization of Disruption Precursors in a Tokamak," Phys. Rev. Letters **64**, 1254 (1990).
8. B. Alper, "A Review of Results from the HBTX Reversed Field Pinch, Phys. Fluids B **2**, 1338 (1990).
9. T.R. Jarboe, F.J. Wysocki, J.C. Fernández, I. Henins, and G. J. Marklin, "Progress with Energy Confinement Time in the CTX Spheromak," Phys. Fluids B **2**, 1342 (1990).
10. J. S. Hardwick, *Ohmic Ignition in a Spheromak*, Ph.D. Dissertation, Dept. of Nucl. Eng., Univ. of Cal., Berkeley, 1995.
11. M. Okabayashi and A. M. M. Todd, "A Numerical Study of MHD Equilibrium and Stability of the Tokamak," Nucl. Fusion **20**, 571 (1980).
12. P. Gaiter, R. Gruber, and F. Troyon, "Numerical Study of the Ideal-MHD Stability Limits in Oblate Spheromaks," Nucl. Fusion **21**, 1399 (1981).
13. S. C. Jardin, "Ideal Magnetohydrodynamic Stability of the Spheromak Configuration," Nucl. Fusion **22**, 629 (1982).

14. R. M. Mayo and G. J. Marklin, "Numerical calculation of Mercier Beta Limits in Spheromaks," Phys. Fluids **31**, 1812 (1988).

15. G. Marklin, "MHD Stable High Beta Spheromak Equilibrium," Proc. US-Japan Workshop Field-Reversed Config. with Steady-State High-Temp. Fusion Plasmas and. 11th US-Japan Workshop. Comp. Toroids, Nov. 1989, LANL report LA-11808-C, p 181, May 1990.

16. J. B. Taylor, "Relaxation and Magnetic Reconnection in plasmas," Rev. Mod. Phys. 58, 741 (1986); "Relaxation of Toroidal Plasma and Generation of Reverse Magnetic Fields," Phys. Rev. Letters 33, 1139 (1974).

17. J. B. Taylor and M. F. Turner, "Plasma Current Drive by Helicity Injection in Relaxed States," Nucl. Fusion **29**, 219 (1989).

18. N. C. Christofilos, "Design for a High Power Density Astron Reactor," LLNL Report UCRL-72957 (1971); J. Fusion Energy **8**, 97 (1989).

19. R. W. Moir, R. L. Bieri, X. M. Chen, et al., "HYLIFE-II: A Molten-Salt Inertial Fusion Energy Power Plant Design – Final Report," Fusion Techn. **25**, 5 (1994).

20. J. S. Latkowski, *Inertial Fusion: A Clearer View of the Environmental and Safety Perspectives*, Ph.D. Dissertation, Dept. of Nucl. Engineering, Univ. of Cal., Berkeley, 1996.

21. R. W. Moir, "The Logic Behind Thick, Liquid-Walled Fusion Concepts," Nuclear Eng. Design **29**, 34 (1995).

22. R. W. Moir, "Liquid First Walls for Magnetic Fusion Energy," LLNL Report UCRL-ID-123902, March 28, 1996.

23. J. H. Hammer and L. D. Pearlstein, "Control of Tilt/Shift Instabilities in the Spheromak," presented at the Workshop on Establishing the Physics Basis Needed to Access the Potential of Compact Reactors, Oak Ridge, July 19-22, 1994 (unpublished). See M. Peng, et al., Fusion Techn. **29**, 210 (1996) for a workshop summary.

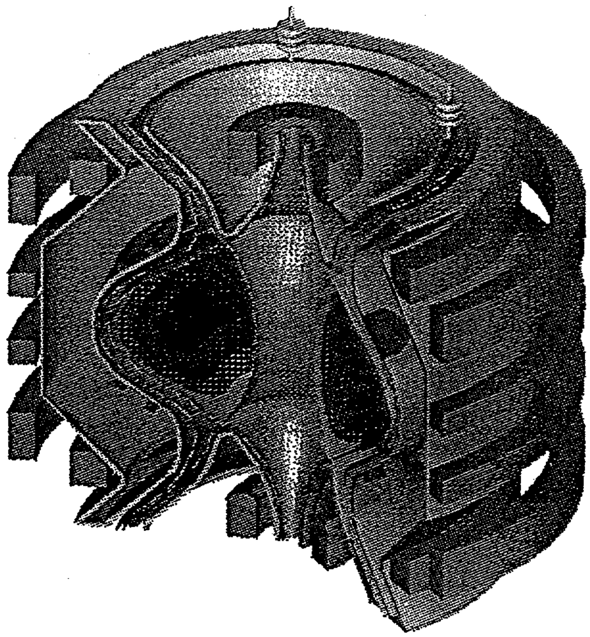

Fig. 1 Spheromak reactor

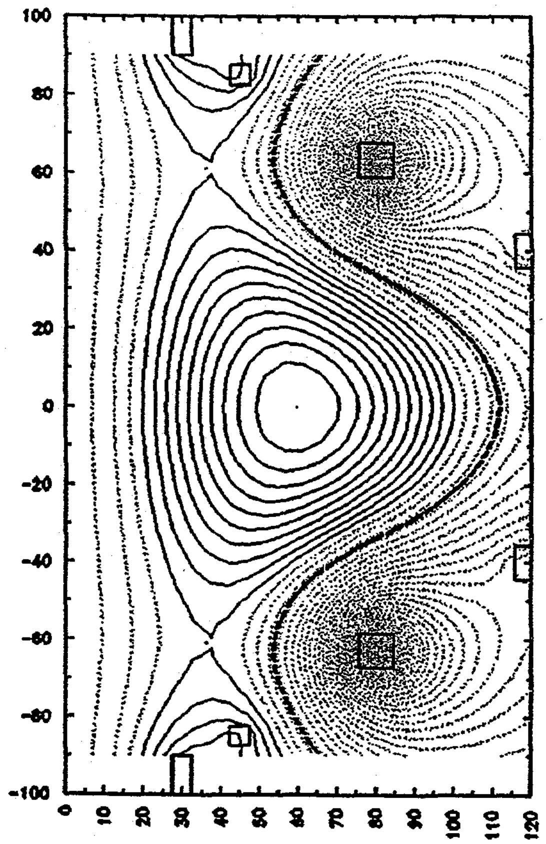

Fig. 2 Magnetic flux surfaces for the reactor design of Fig. 1.

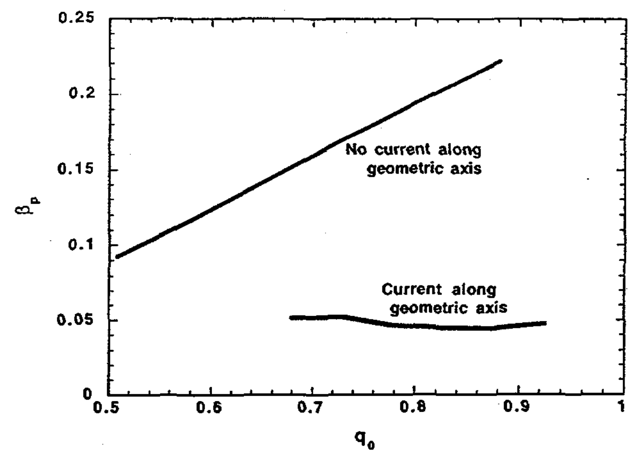

Fig. 3 Maximum Mercier-stable β_p as a function of q_0 with and without current along the geometric axis.