Field-Reversed Configuration Power Plant Critical Issues

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J.F. Santarius, G.A. Emmert, H.Y. Khater, E.A. Mogahed, and C.N. Nguyen
Fusion Technology Institute
University of Wisconsin
1500 Engineering Dr.
Madison, WI 53706
(608)263-2252

L.C. Steinhauer
Redmond Plasma Physics Lab
University of Washington
14700 NE 95th St., Suite 100
Redmond, WA 98052
(425)881-9380

G.H. Miley
Fusion Studies Lab
University of Illinois
100 NEL
103 S. Goodwin Ave.
Urbana, IL 61801-2984
(217)333-3772

ABSTRACT

An Engineering Scoping Study of a Field-Reversed Configuration (FRC) burning D-T fuel is being performed by the Universities of Wisconsin, Washington, and Illinois. The effort concentrates on tritium-breeding blanket design, shielding, radiation damage, activation, safety, environment, plasma modeling, current drive, plasma-surface interactions, economics, and systems integration. A systems analysis code will serve as the key tool in defining a reference point for detailed physics and engineering calculations plus parametric variations. Advantages of the cylindrical geometry and high β (plasma pressure/magnetic-field pressure) are evident.

I. INTRODUCTION

Field-reversed configuration (FRC) power plants appear likely to provide an excellent balance between potential reactor attractiveness and technical development risk. In particular, (1) the linear, cylindrical FRC geometry facilitates the design of tritium-breeding blankets, shields, magnets, and input-power systems, and (2) the high FRC β (plasma pressure/magnetic field pressure) increases the plasma power density and allows a compact reactor design. The surface heat flux is moderate, however, because much of the fusion power is carried by the plasma flowing to the end chamber walls. The present research project investigates critical issues for FRC fusion power plants, concentrating mainly on engineering areas. The issues being studied include tritium-breeding blanket design, radiation damage, activation, safety, environment, shield design, economics, plasma particle and power balance, current drive, plasma-surface interactions, and systems integration. Both D-T and D-^{3}He fuels appear likely to perform well in FRC power plants, but the focus of the present study is on D-T fuel.

With regard to fusion development, the FRC provides a good balance between physics uncertainty and engineering attractiveness. The trade-offs among physics, engineering, safety, and environmental considerations have only recently gained prominence—partly due to the difficulties encountered when the fusion community realistically faced engineering issues in designing ITER. While the physics obstacles on the FRC development path should not be underestimated, excellent progress is being made by the small worldwide FRC research community.[1] The key physics issues include operation at large s (average number of radial gyroradii), startup with reasonable power, and sustainment. From an engineering standpoint, an FRC burning D-T fuel appears capable of being built with near-term technology to a large extent. The main exceptions are the materials used for the first wall, blanket, and shield, which will be subject to high neutron fluences with consequent radiation damage and activation. If the more difficult physics requirements of D-^{3}He fuel can be achieved, essentially all necessary FRC technology appears to be in hand, benefits would be gained from direct conversion, and environmental and safety characteristics would be substantially improved.[2]

Excellent recent progress in FRC physics motivates the present work and makes the research especially timely. Highlights include indications that natural minimum-energy FRC states exist, demonstration of stability to global MHD modes, stable operation at moderate s (plasma radius/average gyroradius), startup by merging two spheromaks to form an FRC, and current drive by rotating magnetic fields.[1]

II. OBJECTIVE AND TASKS

The objective of the present scoping study is to investigate the critical engineering issues for D-T FRC electric power plants. The main tasks involved in this research and the institutions with primary responsibility are:

- *University of Wisconsin*
  - ☐ Coordination
  - ☐ Systems analysis and economics
  - ☐ Tritium-breeding blanket design
  - ☐ Radiation shielding and damage
  - ☐ Activation, safety, and environment
- *University of Washington*
  - ☐ Plasma modeling
  - ☐ Current drive
- *University of Illinois*
  - ☐ Plasma-surface interactions
  - ☐ Plasma exhaust handling

If time and resources permit, other areas will be investigated. Those with the potential for having an important positive or negative impact on the design include maintenance, high-T_c superconducting magnets, energy conversion, liquid-metal first walls, and systems integration.

III. STATUS

The status of the study as of June, 1998, which is about nine months into the two-year project, is as follows:

- The University of Wisconsin systems code, which was written initially for tokamaks, has been modified to include FRC physics, engineering, and economics based on University of Washington physics models and University of Wisconsin engineering and economics models.
- The University of Wisconsin is developing an innovative tritium breeding blanket concept and evaluating the possible use of previously published blanket designs.
- The University of Wisconsin is assessing options for radiation shielding and evaluating required shield thicknesses.
- The University of Washington has generated plasma-physics and current-drive models. These will be refined as the study progresses.
- The University of Illinois is investigating plasma-surface interactions and plasma exhaust handling.

A. Plasma and Current Drive Modeling

The initial calculations use relatively simple quasi-equilibrium profiles:

$$\begin{aligned}
U(u) &\equiv \kappa\frac{\tanh^{-1}\sigma u}{\sigma} && (1)\\
u(r,z) &\equiv \frac{r^2}{R^2}-1 && (2)\\
R(z) &\equiv R_0\left(1-z^4/b^4\right)\right)^{\frac{1}{2}} && (3)\\
B(r,z) &= B_e\tanh U && (4)\\
j_\theta(r,z) &= \frac{cB_e}{4\pi}\cdot\frac{2r}{R^2(z)}\frac{dU}{du}\mathrm{sech}^2\,U && (5)\\
p(r,z) &= p_m\mathrm{sech}^2\,U && (6)\\
n(r,z) &= n_m(\mathrm{sech}\,U)^{2/\Gamma} && (7)\\
T(r,z) &= T_m(\mathrm{sech}\,U)^{2(\Gamma-1)/\Gamma} && (8)
\end{aligned}$$

where $\{r,\theta,z\}$ define a cylindrical coordinate system, $a = 2^{\frac{1}{2}}R_0 =$ separatrix radius, $b =$ FRC half length, $R_0 =$ radius of the 0-point at the midplane, $j =$ current density, $p =$ pressure, $n =$ density, $T =$ temperature, $c =$ speed of light, $B_e =$ external magnetic field, and the subscript m signifies the maximum value. κ and σ are small shaping parameters, and $\Gamma \sim 4/3$.

Viable FRC startup and sustainment methods with reasonable input powers are being sought. The requirements for two promising methods, rotating-magnetic-field (RMF) current drive and merging spheromaks, are being determined and modeled.

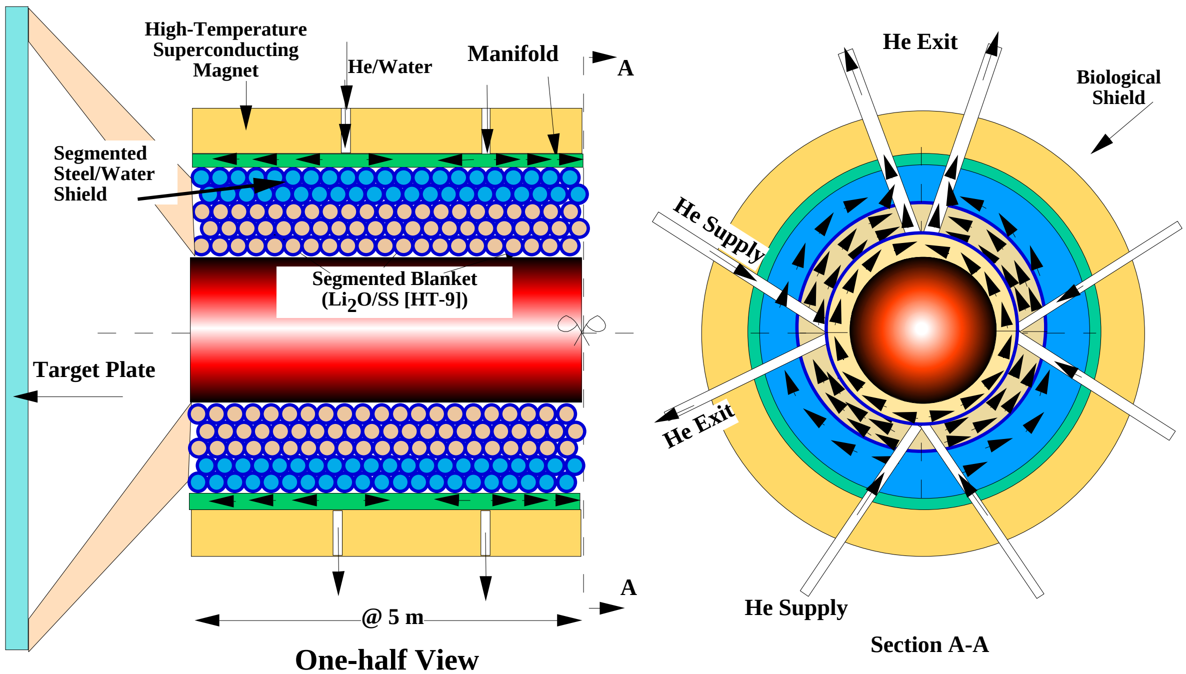

Figure 1: *One-half view and cross section of a preliminary concept for the tritium-breeding blanket.*

B. Tritium-Breeding Blanket Design

A tritium-breeding blanket for an FRC is in many ways simpler than for a tokamak. The extremely high tokamak magnetic fields lead to large toroidal field coils which, along with the toroidal geometry, reduce maintenance access and usually require splitting blanket modules into several submodules and translating them toroidally for removal. In an FRC, the cylindrical geometry and low magnetic field allow removal of *single* modules containing the first wall, blanket, shield, and magnets. If liquid-metal coolants are used, the MHD pressure drop will also be substantially reduced by the low magnetic field and short flow paths. A key question is whether it will be necessary to use materials requiring long-term development, such as SiC or V, or use can be made of nearly off-the-shelf materials, such as low-activation ferritic and austenitic steels.

We are pursuing parallel courses for scoping tritium-breeding blanket designs: (1) evaluate new ideas and (2) assess established concepts. Figure 1 shows the initial blanket concept, which uses Li_2O breeder, austenitic or ferritic steel structure, and helium coolant. Stainless steel structure and water coolant would be used for the shield. Another possibility is use of the Mirror Advanced Reactor Study (MARS)[3] blanket, shown in Fig. 2, with key parameters given in Table 1. This approach would take advantage of the geometrical similarity of FRC and tandem-mirror core regions.

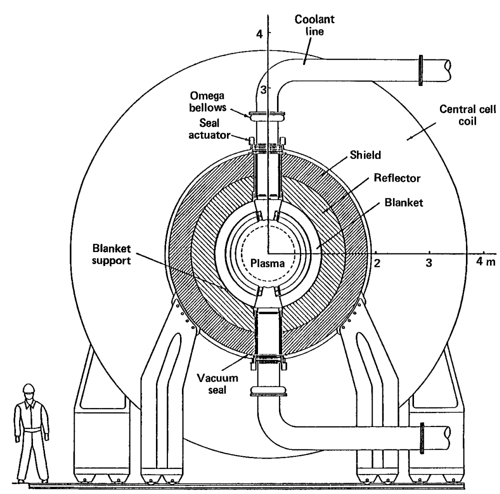

Figure 2: *Cross section of the MARS tritium-breeding blanket.*

Table 1: *Parameters for the MARS Blanket.*

Structure	HT-9 Steel
Coolant	Li17 Pb83
Breeder	Li17 Pb83
r_w	0.6 m
Γ_n	4.3 MW/m^2
M	1.36
TBR	1.15
η_{th}	42%

Table 2: *MINIMARS central cell parameters.*

Number of modules	24
Module length	2.8 m
Module radius	1.75 m
Magnet thickness	0.06 m
Peak B field	3.1 T
Current density	37 MA/m^2
Current	30 kA

C. Plasma-Surface Interactions and Plasma Exhaust Handling

Plasma-surface interactions are being studied in both the fusion core and end tanks. Most transport losses are along the magnetic field lines to an end tank or direct energy converter. There, flux tubes can be expanded to reduce heat fluxes and particle erosion of surfaces. Thus, while plasma exhaust issues are similar to those encountered in a tokamak divertor, in an FRC the ability to employ larger surface areas significantly alleviates design difficulties.

An important advantage of FRC power plants is that they are not subject to the type of disruption experienced by a tokamak, where the energy from the thermal quench gets deposited inside the fusion core chamber on divertor plates or the first wall. Analogous MHD instabilities in FRC's will cause the plasma to flow along the magnetic flux tube and deposit in an end tank, where the flux tube can be expanded to mitigate the effect and space exists for a more robust design. This avoids the tokamak's extremely difficult divertor design problem and also helps keep material ablated by plasma-wall interactions caused by an instability from coating the fusion chamber in unpredictable locations.

The steady-state heat flux on an FRC first wall results mainly from bremsstrahlung radiation, because almost all charged particles will follow magnetic flux tubes to the end tanks. Although the first wall surface heat loads resulting from the radiation will be ~1 MW/m^2, the heat flux will be fairly uniform, and the FRC should not experience the steady-state and much higher peak and average surface heat fluxes of the tokamak divertor.

D. Radiation Shielding, Activation, Safety, and Environment

Several neutronics and safety advantages exist for a D-T FRC. The FRC geometry allows a high coverage fraction for tritium breeding and hence allows efficient breeding using a solid breeder with possibly no need for a beryllium multiplier. Eliminating the beryllium multiplier would allow the use of water inside the vacuum vessel while maintaining good safety by eliminating the risk of severe hydrogen production caused by the beryllium-water interaction. An FRC has no analogue to tokamak disruptions, thus providing a significant safety advantage by lowering the vulnerability to an accidental release of radioactivity.

E. Magnets

For the FRC magnets, the MINIMARS[4] central-cell magnets, which were radially thin coils that covered the tandem mirror central cell nearly uniformly, would be suitable for the present design. Parameters for the MINIMARS central-cell magnets are given in Table 2. Even more leverage would be gained by the use of high-temperature superconductors, which should be more robust against quenching and require less shielding, thereby allowing larger internal heat deposition by radiation.

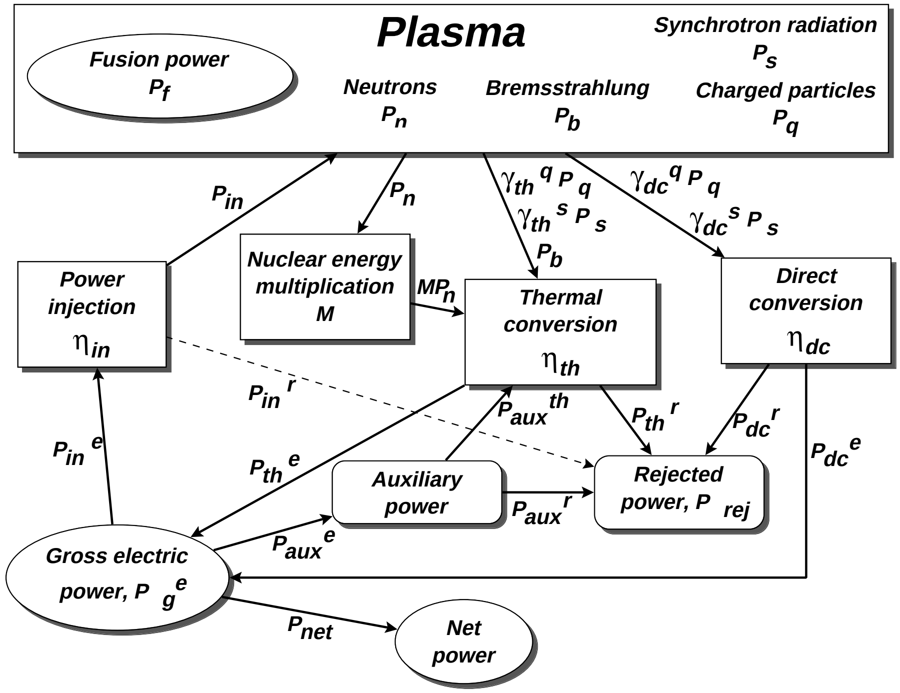

Figure 3: *FRC plasma and plant power flow.*

F. Systems Analysis and Economics

The University of Wisconsin systems code contains plasma physics, engineering, and economics models, and its power flow model is illustrated in Fig. 3. The code is presently being benchmarked, and trade-offs among various reactor design options are being assessed.

IV. SUMMARY

The high power density and cylindrical geometry of D-T FRC's should allow them to overcome the major engineering obstacles facing D-T tokamaks. From a fusion energy development perspective, FRC's occupy the important position of leading on the path of high power density and relatively simple, linear geometry that should speed engineering progress once the physics obstacles are overcome.

ACKNOWLEDGMENT

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