A major purpose of the Technical Information Center is to provide the broadest dissemination possible of information contained in DOE's Research and Development Reports to business, industry, the academic community, and federal, state and local governments.

Although a small portion of this report is not reproducible, it is being made available to expedite the availability of information on the research discussed herein.

1

LA-UR- 85-3984

RECEIVED BY OSTI DEC 06 1985

CONF-851102--45

LA-UR--85-3984

DE86 003677

TITLE: PROSPECTS FOR FUSION APPLICATIONS OF REVERSED-FIELD PINCHES

AUTHOR(S): C. G. Bathke
R. A. Krakowski
K. L. Hagenson, Phillips Petroleum Company

SUBMITTED TO: 11th Symposium on Fusion Engineering
Austin, Texas
November 18-22, 1985

Los Alamos Los Alamos National Laboratory
Los Alamos, New Mexico 87545

FORM NO 836 R4
ST NO 2629 5/81

MASTER

PROSPECTS FOR FUSION APPLICATIONS OF REVERSED-FIELD PINCHES*

C. G. Bathke, R. A. Krakowski, and R. L. Hagenson**
Los Alamos National Laboratory, Los Alamos, NM 87545

Abstract: The applicability of the Reversed-Field Pinch (RFP) as a source of fusion neutrons for use in developing key fusion nuclear technologies is examined. This Fusion Test Facility (FTF) would emphasize high neutron wall loading, small plasma volume, low fusion and driver powers, and steady-state operation. Both parametric tradeoffs based on present-day physics understanding and a conceptual design based on an $\sim 1\text{-MW/m}^2$ (neutron) driven operation are reported.

Introduction

The toroidal, axisymmetric Reversed-Field Pinch (RFP) confines high-beta plasma in a configuration with strong ohmic heating and low-field coils. On the basis of good experimental results[1] and promising reactor projections,[2,3] a multi-mega-ampere device, ZT-H, has been proposed.[4] Intermediate between the ZT-H (r_p = 0.4-m plasma radius, I_ϕ = 4-MA toroidal plasma current) and the compact reactor embodiments (r_p = 0.71 m, I_ϕ = 18.4 MA) are a number of RFP devices that can serve technology [Fusion Technology Facility (FTF)[5], I_ϕ = 7-8 MA], DT-ignition (I_ϕ = 8-10 MA), and reactor-demonstration (I_ϕ > 15 MA) functions. In order to define better key steps in the RFP development path and in support of a broader assessment of fusion technology,[5] conceptual design studies of an RFP device with FTF-like qualities[5] are being conducted. The RFP/FTF would be a high-current extension of ZT-H,[4] utilizing current drive and active impurity/ash control. Guided by systems studies, a conceptual design of the RFP/FTF is performed using coupled models for a) ohmic-heating and equilibrium-field coils; b) time-dependent plasma/circuit simulations using experimental scalings; c) oscillating-field (F-Θ pumping) current-drive simulations; d) edge-plasma simulation and first-wall thermal-mechanical/thermal-chemical analyses; and e) magnetic-divertor impurity control. An RFP/FTF design and required physics/technology database resulting from this study are described.

Reversed-Field Pinch Concept

The primary confining field, B_θ, in an RFP is poloidal and is generated by a toroidal plasma current, I_ϕ. The RFP plasma supports a toroidal bias field, B_ϕ, to stabilize sausage (m = 0) and elliptical (m = 2) distortions. Grossly unstable MHD modes with wavelengths longer than the minor radius of an electrically conducting shell are stabilized by the shell on a short time scale and by feedback coils for longer times. If the toroidal bias field is slightly reversed near the plasma edge, the resulting magnetic shear in the plasma-edge region is sufficient to stabilize local pressure-driven and current-driven instabilities. This stabilization occurs at relatively high values of the normalized plasma pressure, $\beta = 2nk_BT/(B^2/2\mu_0)$.

The key descriptive parameters in the minimum-energy RFP theory[6] are the pinch parameter, Θ, and the reversal parameter, F, which are defined as $\Theta = B_\theta(r_p)/\langle B_\phi\rangle$ and $F = B_\phi(r_p)/\langle B_\phi\rangle$, where $\langle B_\phi\rangle$ is the average toroidal field within the zero-temperature plasma radius, r_p, which is also taken here as the conducting shell. The locus of minimum-energy states, as described in an F-Θ phase space confirmed by experimental F-Θ traces, shows the plasma residing within a region of F-Θ space where F < 0 and 1.2 < Θ < 1.6. For the purposes of ignition/burn, RFP/FTF,[5] and reactor[2,4] studies, the F-Θ constraint is enforced both during startup and burn.

Evidence for nearly classical resistivity in RFP plasmas exists,[7] giving a strong indication of an efficient plasma dynamo to maintain the RFP field configuration. Unlike the tokamak, a close electrical coupling exists between the poloidal and toroidal circuits through the RFP plasma. This coupling also provides in principle a means to drive toroidal current noninductively[2,8] at low frequency (50 Hz for the reactor[2]). Preliminary experimental evidence in support of these ideas recently was reported.[1] This oscillating-field (F-Θ pumping) current drive serves as the basis for a steady-state RFP/FTF design reported herein.

A potential problem of enhanced plasma transport caused by the RFP dynamo remains. Generally, the field-line breaking and reconnecting that may be at the base of the RFP dynamo[9] is expected to reduce energy confinement within internal regions of the plasma. An empirical expression for the scaling of global confinement time from small, ohmically heated experiments is used. Specifically, the combination of pressure balance $[T = \beta_\theta I_\phi(I_\phi/N)]$, plasma-energy balance $(nT/\tau_E = \eta j_\phi^2)$, and classical resistivity $(\eta = 1/T^{3/2})$ predicts that $\tau_E/r_p{}^2 = \beta_\theta^{5/2}(I_\phi/N)^{3/2}I_\phi{}^{3/2}$ or $n\tau_E = I_\phi^{5/2}$. The RFP plasma burn simulations utilize an empirical scaling of the form, $\tau_E/r_p^2 = C_\nu I_\phi^\nu f(\beta_\theta)$. The parameters C_ν and ν have been calibrated with existing experimental results,[1] although direct experimental evidence for the r_p^2 and β_θ scaling remains to be generated.

In summary, a strong experimental database is evolving from a number of small RFP devices. This database has provided the foundation for the next major, mega-ampere RFPs presently under consideration by the US[3] and EEC.[10] This database is summarized below.

- robust dynamo initiation and sustainment
- slow current ramp after low-energy RFP formation
- constant-beta scaling ($nk_BT = I_\phi^2$)
- temperature increases with current
- current density sufficient for strong ohmic heating
- confinement time increases with current ($\tau_E = I_\phi^\nu$, ν = 1.0-1.5)
- dynamo coupling of poloidal and toroidal circuits to suggest low-technology current drive

*Work performed under the auspices of US Department of Energy

**Phillips Petroleum Co., Bartlesville, OK

Parametric Design

Design Models

Plasma Model. An optimum RFP/FTF design generally establishes a ceiling on total capital (core size, support power) and operating (support power, fuel requirements) costs for a system that maximizes neutron first-wall loading, device availability, and experimental volume (and first-wall area) and minimizes plasma volume and total fusion power. Since the means and constraints by which to optimize the RFP/FTF are not well established, the reactor equations described in Appendix A of Ref 2 were first solved parametrically in steady state simply to establish the main physics parameters for small RFPs. A simplified model of the coils was used to obtain an initial estimate of core mass, power consumption, and possible startup senarios; detailed circuit and magnet analyses were then performed on the basis of design points suggested by these steady-state analyses.

A driven, small RFP operating with both high particle density and current density was judged as most appropriate for the FTF application. A DT-ignited RFP generally would generate fusion powers above the P_F = 100-MW upper limit for an FTF[5], although the exact limit depends on the plasma beta and transport scaling assumed. The average first-wall heat flux, I_{Qw}, the ohmic power delivered to the plasma, $P_{\Omega p}$, and the ohmic power consumed by the coil set, $P_{\Omega c}$, were monitored in steady state along with the neutron first-wall loading, I_w, and the total fusion power, P_F, for a given plasma beta. Although Z_{eff}, plasma aspect ratio, $A = R_T/r_p$, the transport scaling parameter, ν, the pinch parameter, Θ, and the anomalous ion heating were varied, the basecase selected Z_{eff} = 1, A = 6, ν = 1.0-1.25, F = -0.1 (corresponding Θ determined from plasma equilibrium and a modified Taylor theory for a given beta), and no anomalous ion heating. Both F-Θ pumping current drive and active impurity control (either poloidal pumped limiters[2] or toroidal-field magnetic divertors) were investigated.

Results from the steady-state plasma simulations are displayed on plots of plasma current versus plasma minor radius where either I_{Qw}, I_w, P_F, or $P_{\Omega p}$ were held fixed. Figure 1 illustrates a design plot for basecase parameters with ν = 1.25 and β_θ = 0.06 or 0.10. Given the constraints of $I_w > 1$ MW/m^2, $P_F < 100$ MW, and $I_{Qw} < 5$ MW/m^2, a design "window" is defined in Fig. 1. On the basis of present experiments[1] ($I_\phi < 0.5$ MA, r_p = 0.15-0.2 m) and projected near-term experiments[3] (I_ϕ = 2-4 MA, r_p = 0.3-0.4 m), it was judged that I_ϕ = 7-8 MA and r_p = 0.3-0.4 m represents a region of reasonable extrapolation from the next generation RFPs. A representative design point is also indicated on Fig. 1 for more detailed exploration of the β_θ = 0.1 case. This I_w = 1-MW/m^2 design is not ignited although increasing the current from the I_ϕ = 7.6 MA value to > 9 MA would give I_w = 4-5 MW/m^2 and DT ignition. Before preliminary engineering parameters for the RFP/FTF design point can be tabulated, however, an estimate of the steady-state power consumption in and size of the confining poloidal-field coils (PFCs) and toroidal-field coils (TFCs) is needed.

Magnet Model. Reduced to the simplest terms, the RFP converts large currents in external poloidal-field coils, $I_{\phi c}$, to nearly equally large currents, I_ϕ, in a toroidal plasma. This plasma current both confines and heats the high-β DT plasma. Rather than minimizing the cost of energy, as is done for a power reactor design,[2,4] the RFP/FTF would maximize I_w while minimizing the power delivered to the coils and plasma as well as total plasma size, fusion power, and tritium requirement. On the basis of the designs suggested on Fig. 1, the RFP/FTF design task then becomes one of current and power management in a ZT-40M/ZT-H class of devices.

A simplified model is used to estimate the mass of and power dissipated in the coils. A more detailed circuit and plasma equilibrium analyses is then performed to give better estimates of startup scenarios, coil stresses, and volt-second requirements. Past analyses using this simplified approach, which

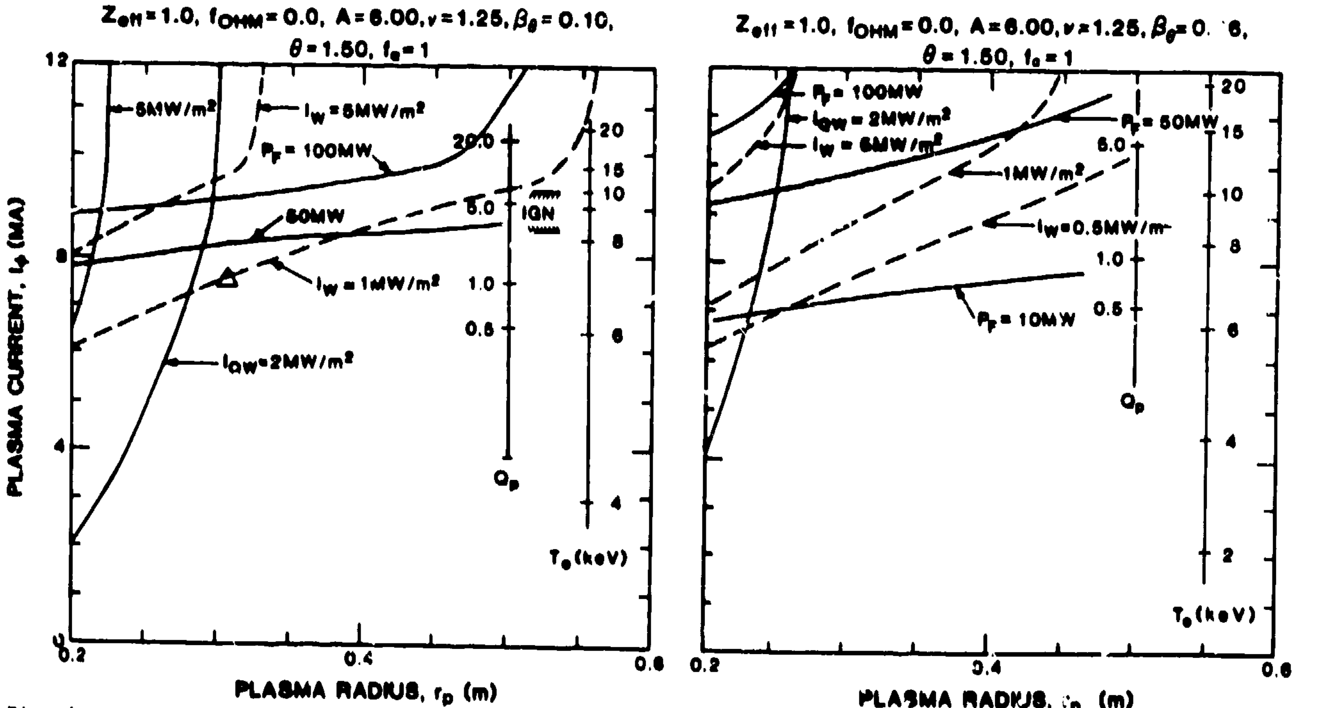

Fig. 1. Interdependence of plasma current, I_ϕ, on plasma radius, r_p, for a range of average first-wall heat fluxes, I_{Qw}, neutron first-wall loadings, I_w and total fusion power, P_F. For all cases, the plasma beta, is taken as 0.06 and 0.10, respectively, the transport parameter, ν = 1.25, and no anomolous ion heating assumed. A design window is defined by $I_w > 1$ MW/m^2, $P_F < 100$ MW, and $I_{Qw} < 5$ MW/m^2.

treats both TFCs and PFCs as homogeneous shells, have proven to be adequate and verifiable.[2,4] The ratio of ohmic power dissipated in the idealized PFC/TFC set to that delivered to the plasma is given in Fig. 2., where $x = r_p/r_w$ is a minor radial filling fraction for the plasma, the plasma and coil resistivities are η_p and η_c, respectively, g_{OHM} is a plasma profile factor for ohmic heating[3], λ is the coil conductor filling fraction ($\lambda = 0.7$), and L is a geometric factor. The coil-to-plasma current ratio for an ideal closely-coupled plasma/coil system initiated with a bipolar current swing is used. The tradeoff between normalized PFC thickness, $\delta_{c\theta}/r_w$, and plasma aspect ratio is given on Fig. 2 for the case where the TFC standoff distance from the plasma, Δb, equals the first-wall radius, and $x = r_p/r_w = 1$; the condition $\Delta b = r_p$ gives an experimental volume equal to three times the plasma volume. Selecting a value of A = 5 based on near-optimal coupling of coil currents with plasma current allows the dependence of M_c/r_p^3 and $j_{c\theta}/j_\theta$ on $\delta_{c\theta}/r_w$ also to be displayed, where M_c is the coil mass and j_θ, $c\theta$, are the plasma and coil current density, respectively. The coil standoff distance from the first-wall, $\Delta b = r_w$, provides space both for tests and shielding to be located between the first wall and the TFC set. The impact of an ideally coupled PFC for A = 5 is also shown by the A = 5(Δb = 0) curve. The RFP/FTF is not designed to breed tritium.

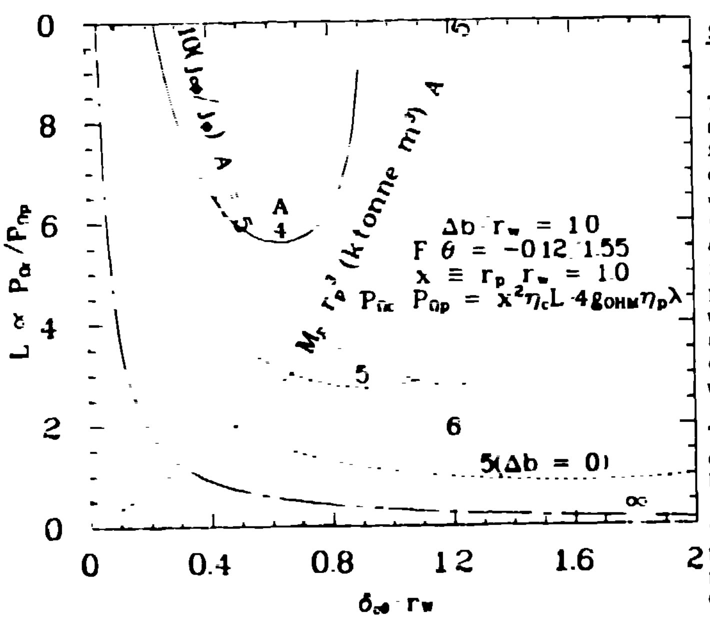

Fig. 2. Dependence of ohmic power losses in coils, $P_{\Omega c}$, relative to that in the plasma, $P_{\Omega p}$, for a homoginized coil model, showing the dependence on plasma aspect ratio, $A = R_T/r_p$, poloidal coil thickness, $\delta_{c\theta}$, for a TFC standoff $\Delta b = r_w$ and a minor radial plasma filling fraction x = 1. Shown also is the mass of the coil set, M_c, and the ratio of PFC to plasma current density, $j_{c\theta}/j_\theta$.

The coil power requirements given on Fig. 2 are based on a bipolar inductive swing and peak current conditions. An inductive pulse for this small system at most would last for only ~ 10-20 s. Application of F-Θ pumping[2,4,8] to drive the plasma current, either before or after peak current, would allow the ohmic-heating-coil (OHC) current to be driven to zero. Generally, the OHC power requirements should be adequate to supply current-drive losses if transferred to the F-Θ pumping current-drive system.

Interim Design Point

The combination of plasma and coil results given, respectively, on Figs. 1 and 2 are used to develop a first estimate of the RFP/FTF design point(s) for subsequent, more-detailed plasma simulation and engineering analysis. Table I summarizes geometric, plasma, and magnet characteristics for a single device that would generate a neutron current over the range I_w = 1-5 MW/m² while assuring that the fusion power is held below ~ 100 MW. These designs are also identified on the β_θ = 0.1 plot in Fig. 1. The ohmic power delivered to the plasma is held below 30 MW, and the average physical heat flux on in-vacuum components is below $I_{\Omega w}$ = 2 MW/m². A summary of the computational basis of this example is given in the list of footnotes accompanying Table I. Although this design is subignited (Q_p = 1.0), an increase of ~ 25% in the 7.6-MA design current (I_w = 1 MW/m²) would result in ignition and an increase in I_w to ~ 5 MW/m² (Fig. 1). Generally, the I_w = 1-MW/m² driven design serves here as the basecase. This basecase design is for a transport parameter ν = 1.25 and a poloidal beta of β_θ = 0.1. Higher or lower transport and/or beta would shift the design window; Fig. 3 gives this sensitivity of this physics design to variations in ν and β_θ.

Conceptual Engineering Design

Startup

The small size of the RFP/FTF designs listed on Table I gives L/R times for both the plasma and coil sets that are sufficiently short to make desirable some form of current sustainment. A bipolar startup is envisaged, with the PFCs serving as an energy store used to initiate a low-current, low-energy (~ 0.5-1.0 keV) RFP; a purely inductive startup through a resistive transfer is sufficiently stressing and inefficient to preclude its use for attaining the final plasma conditions listed in Table I. Instead, the PFCs would be charged in a reversed-bias condition to a state not unlike that of the final, full-plasma-current condition. A resistive transfer in time t_R = 1-2 s would form an RFP that is subsequently ramped in a longer time to achieve the final steady-state plasma. This slow current ramp would initially be driven directly from the power grid, with F-Θ pumping possibly being applied prior to current flat-top if the plasma resistance becomes sufficiently low. The plasma would then be taken to the final conditions, and the F-Θ pumping current drive would thereafter sustain the plasma. The OHC current at this point decays to zero. Optimization of this startup and sustainment scenario to minimize power, magnetic flux consumption, and technology requirements is required. The crucial tradeoff between coil cost and technology (i.e., voltage, power, and volt-second requirements) and the overall approach to the F-Θ pumping drive coils is examined with a time-dependent plasma/circuit simulation of the design suggested in Table I.

Magnetics

__Equilibrium__. The PFC design follows the procedure outlined in Ref. 2. The PFCs are subdivided into two functional sets: a) equilibrium-field coils (EFCs) to provide a vertical magnetic field of the appropriate magnitude, B_v, and index, n, to ensure horizontal and vertical equilibrium, respectively, and b) OHCs to provide the bulk of the inductive flux swing, L_pI_ϕ, without introducing magnetic field into the plasma region. For the 1-MW/m² neutron-wall-loading case given on Table I, B_v = 1.25 T, 0 < n < 0.65, and L_pI_ϕ = 44.0 Wb are required. The coil design is further constrained to maintain the peak current density below 10 MA/m², to minimize the total o[illegible] losses, and to avoid coil overlap. A coil design which fits snuggly about the TFCs and is representative of the shell model used in the parametrics model (Figs. 1 and 2) is shown in the lower half of Fig. 4. This "snug" coil design consumes 14.3 MW in the EFCs ($6.3\ MA/m^2$ average current density) and 20.4 MW in the OHCs ($6.8\ MA/m^2$ average current density). The "snug" design gives the minimum ohmic loss (34.7 MW) and represents a 58% increase over that predicted by the parametrics model. Maintenance of an experimental access to the region inside the PFCs would require the removal of a portion of the 207-tonne PFC set. A more maintainable and accessible design is shown in the upper half of Fig. 4. This design provides an opening on the outboard side through which quadrants of the torus, inclusive of the TFCs or divertors, could be moved. This "open" design consumes 14% more ohmic power (38.0 MW) and is 9.5% more massive (236.0 tonnes) compared to the "snug" design.

The TFCs shown in Fig. 4 have a thicker radial build (0.08 m) than predicted by the parametrics model in order to accommodate discreet coils with a uniform current density of $6.7\ MA/m^2$ and uniform cross-sectional area. The resulting TFCs occupy less volume and, hence, consume less ohmic power (1.7 MW). The minimum number of TFCs is estimated to be ~ 24 in order to maintain acceptably small ripple ($\Delta B_R/B_\theta < 0.001$, where ΔB_R is the amplitude of the radial helical magnetic-field perturbation) and sufficiently small magnetic islands at the plasma edge [$\Delta r/(r_p-r_v) < 1$, where r_v is the reversal surface radius and Δr is the island width].

Circuit/Burn Simulation. The coil inductances and resistances associated with the "open" design were used in a time-dependent plasma/circuit simulation[2] of the startup of the $1\text{-}MW/m^2$ neutron-wall-loading design. This simulation indicates an initial OHC back bias of -25 MAmp-turns is required to provide the necessary flux swing and the associated resistive losses in the plasma. This back-biased condition creates an initial (peak) Von Mises stress of 143 MPa, which is within the design constraint (200 MPa). The peak inductive power during ramp-up of current is estimated to be 270 MW draw from the grid and the consumption of ~ 45 Wb; the ZT-40M experiment[1] requires ~ 20 MW and ~ 1 Wb, and the ZT-H experiment[3] is estimated to require 100 MW and 20-30 Wb. All coils together require 270 MW (peak) from the power grid just before current flattop (at 3.3 s), at which point the power required drops to the previously quoted 38 MW.

Current Drive

The steady-state conditions suggested by the parametrics code and plasma/circuit simulations are assumed to be maintained by F-Θ pumping current drive.[2] The current-drive analysis is performed with a time-dependent simulation[2] of the plasma response to sinusoidal fluctuations in the poloidal-field and toroidal-field circuits. A 90 degree phase difference between the two circuits is imposed to maximize current-drive efficiency. For the design values of F and Θ and other plasma parameters, a frequency > 200 Hz is needed in order to maintain toroidal-field reversal during the current-drive phase. The drive frequency can be lowered by operating at slightly higher values of Θ and correspondingly deeper reversal with only modest (~ 1-2 MW) increases in TFC ohmic losses. Restricting the toroidal flux swing to be $\delta\Phi/\Phi_o = 0.03$ at 200 Hz would result in a current modulation, $\delta I_\phi/\langle I_\phi\rangle = 0.017$. Although the poloidal-field and toroidal-field circuits provide comparable resistive power to the plasma (11 MW and 18 MW, respectively) the high-Q poloidal-field circuit requires a peak and RMS reactive power of 3.6 GW and 2.3 GW, respectively. Such high power levels can be handled inexpensively [~ \$10/kVAR(reactive)]) and losses below ~ 1%.

Core Integration

Figure 4 schematically illustrates the essential elements of the RFP/FTF. Combined with the engineering parameters listed in Table I, selection of an active

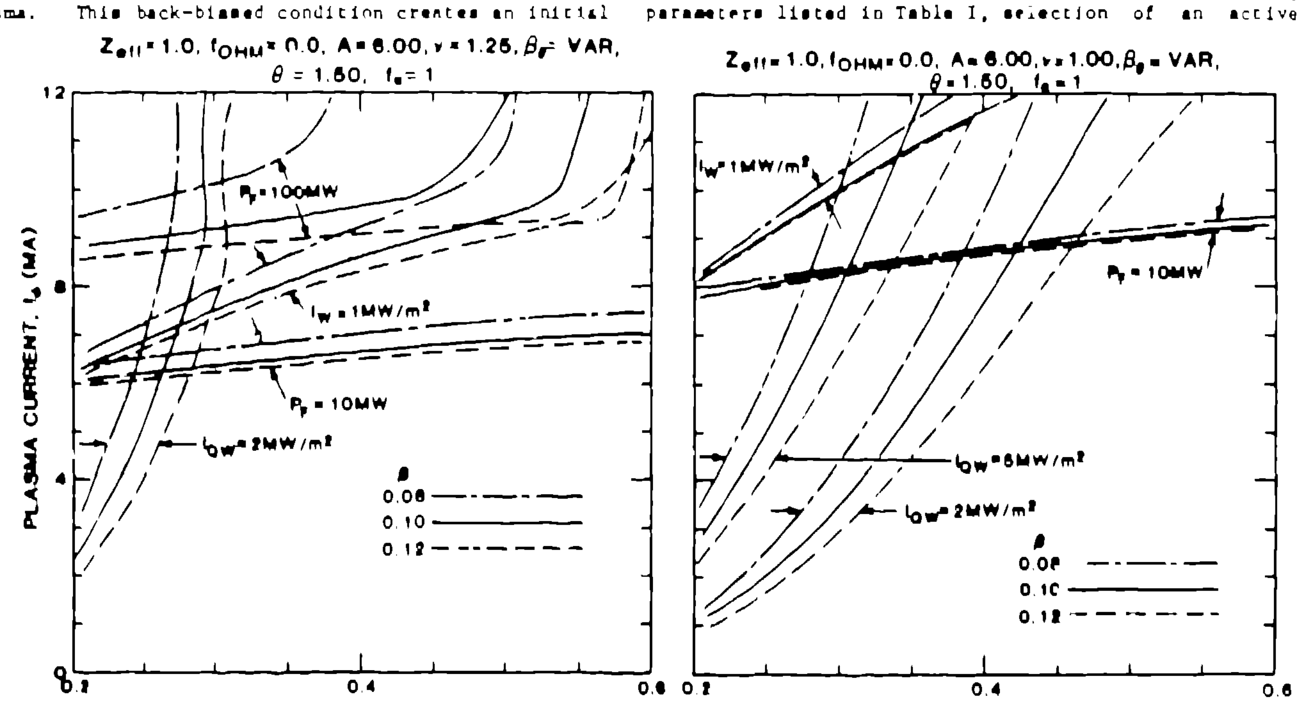

Fig. 3. Sensitivity of basecase design window and design point to variations in plasma beta, β_θ, and transport parameter, ν.

impurity control scheme, detailed neutronics, vacuum, thermal-hydraulic, mechanical, power-handling, and operational computations will lead to an engineering integration of sufficient detail to allow accurate economic and technological assessments to be made. This core integration activity will pursue the "open" configuration depicted in Fig. 4, which can accommodate both horizontal access to test specimens without PFC demounting, as well as toroidal-field divertor chambers.

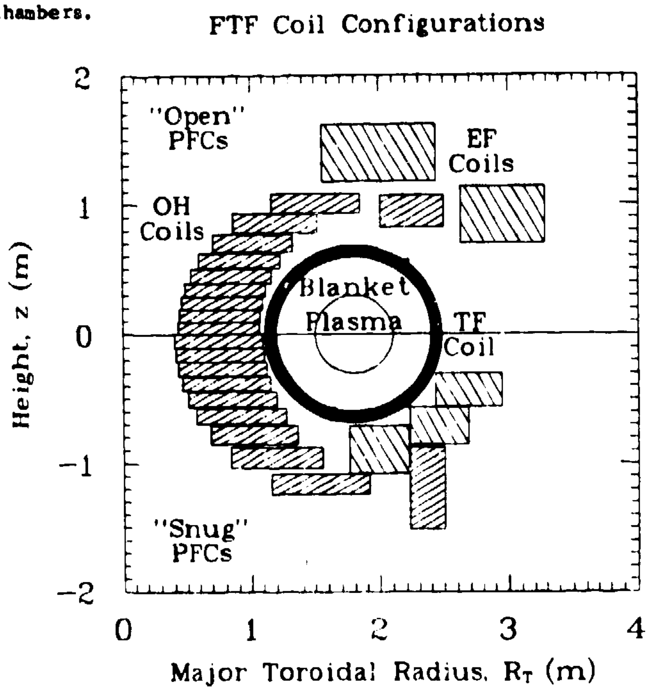

Fig. 4. Comparison of a more maintainable/accessible "open" PFC configuration (upper half plane) with a more-efficient "snug" PFC configuration (lower half plane).

Conclusions

Scoping studies of a Fusion Test Facility (FTF) based on a long-pulsed or steady-state Reversed-Field Pinch (RFP) have been performed. The FTF constraints suggested in Ref. 5 have led to the following general features of an ohmically heated neutron source:

- Small physical size, total fusion power, and driver power (ZT-40M/ZT-H size)[1,3] while maintaining high neutron first-wall loading to give small capital and operating costs.
- Confinement scaling based on present RFP experimental results.[1]
- Sub-ignition operation with the possibility of anomalous ion heating[1] (not assumed here) to minimize further the power input and plasma size while maximizing neutron first-wall loading.
- State-of-the-art resistive TFC and PFC to minimize ohmic power requirements in a compact (200 tonne) device.
- Steady-state oscillating-field current drive (F-Θ pumping) consistent with the experimentally observed[1] RFP plasma dynamo, but remaining to be demonstrated.
- Impurity control options provided by poloidal pumped limiters or toroidal-field divertors.
- Moderate-to-high beta (0.05–0.15) operation, also consistent with experiment.[1]

The RFP program is not yet ready to embark on a device of the class suggested in Table I. The next-step RFP devices presently being designed and proposed for construction,[3] however, represent the necessary and significant intermediate step to the RFP/FTF (and ultimately for the RFP reactor), as these systems strive for plasma currents in the range I_ϕ = 2-4 MA. Key issues in both physics and technology can be identified with the above list, many of which will be resolved by the next RFP devices being planned.[3,10]

References

1. R. S. Massey, R. G. Watt, P. G. Weber, G. A. Wurden, D. A. Baker, C. J. Buchenauer, et al., "Status of the ZT-40M RFP Experimental Program," Fus. Technol. 8 (1), 1571 (1985).
2. R. L. Hagenson, R. A. Krakowski, C. G. Bathke, R. L. Miller, M. J. Embrechts, N. M. Schnurr, et al., "Compact Reversed-Field Pinch Reactors (CRFPR): Preliminary Engineering Considerations," Los Alamos National Laboratory report LA-10200-MS (August 1984).
3. P. Thullen and K. F. Schoenberg (eds.), "ZT-H Reversed-Field Los Alamos National Laboratory document LA-UR-84-2501 (1985).
4. C. Copenhaver, R. A. Krakowski, N. M. Schnurr, R. L. Miller, C. G. Bathke, R. L. Hagenson, et al., "Compact Reversed-Field Pinch Reactors (CRFPR): Fusion-Power-Core Integration Study," Los Alamos National Laboratory report LA-10500-MS (August 1985).
5. M. Abdou (ed), "FINESSE: A Study of the Issues, Experiments, and Facilities for Fusion Nuclear Technology Research and Development (Final Report)," University of California at Los Angeles report (to be published, 1985).
6. J. B. Taylor, "Relaxation of Toroidal Plasma and Generation of Reversed Magnetic Field, Phys. Lett. 33, 1139 (1974).
7. K. F. Schoenberg, R. W. Moses, Jr., and R. L. Hagenson, "Plasma Resistivity in the Progress of a Reversed-Field Dynamo," Phys. Fluids 27(7), 1671 (1984).
8. K. F. Schoenberg, R. F. Gribble, and D. A. Baker, "Oscillating Field Current Drive for Reversed Field Pinch Discharges," J. Appl. Phys. 56 (9), 2519 (1984).
9. K. A. Werley, R. A. Nebel, and G. A. Wurden, "Transport Description of the Rise Time of Sawtooth Oscillations in Reversed-Field Pinches," Phys. Fluids 28(5), 1450-1453 (May 1985).
10. H. A. B. Bodin and G. Rogstani, "The RFX Experiment Technical Proposal," Culham Laboratory report RFX-R1 (April 15, 1981).

Table I. DEVICE PARAMETERS FOR FUSION TEST FACILITY (FTF) BASED ON REVERSED-FIELD PINCH (RFP)

<table>
<tr><th rowspan="2">PARAMETER</th><th colspan="2">FUSION NEUTRON FIRST-WALL LOADING</th></tr>
<tr><th>1 MW/m^2</th><th>5 MW/m^2</th></tr>
<tr><td><u>GEOMETRY</u></td><td></td><td></td></tr>
<tr><td>Plasma major/minor radius, R_T/r_p(m)</td><td colspan="2">1.80/0.30[a]</td></tr>
<tr><td>Plasma/blanket volume, $V_p/V_{BLK}(m^3)$</td><td colspan="2">3.2/9.6</td></tr>
<tr><td>First wall area, $A_{FW}(m^2)$</td><td colspan="2">21.3</td></tr>
<tr><td>Blanket/shield thickness, Δb(m)</td><td colspan="2">0.30[b]</td></tr>
<tr><td><u>PLASMA</u></td><td></td><td></td></tr>
<tr><td>Pinch/reversal parameter, Θ/F</td><td colspan="2">1.50/-0.074</td></tr>
<tr><td>Poloidal field at plasma edge, B_θ(T)</td><td>5.1</td><td>6.4</td></tr>
<tr><td>Poloidal/total beta, β_θ/β</td><td colspan="2">0.10/~ 0.05</td></tr>
<tr><td>Average electron/ion temperature,[c] T_e/T_i(keV)</td><td>4.7/4.5</td><td>11.2/10.8</td></tr>
<tr><td>Average electron density, $n_e(10^{20}/m^3)$</td><td>6.9</td><td>4.6</td></tr>
<tr><td>Plasma current/current density, $I_\phi(MA)/j_\phi(MA/m^2)$</td><td>7.6/26.8</td><td>9.6/33.8</td></tr>
<tr><td>Lawson parameter,[d] $n\tau_E(10^{20}\ s/m^3)$</td><td>0.97</td><td>0.89</td></tr>
<tr><td>Ohmic/fusion power in plasma, $P_{\Omega p}/P_F$(MW)</td><td>29.0/29.2</td><td>13.4/133.0</td></tr>
<tr><td>Plasma Q-value, $Q_p = P_F/P_{\Omega p}$</td><td>1.0</td><td>9.9</td></tr>
<tr><td>First-wall average heat flux, $I_{Qw}(MW/m^2)$</td><td>1.6</td><td>1.9</td></tr>
<tr><td>Poloidal flux, L_pI_ϕ(Wb)</td><td>40.7</td><td>51.3</td></tr>
<tr><td>Kinetic/magnetic energy stored in plasma, W_p/W_B(MJ)</td><td>1.5/154.1</td><td>2.4/245.6</td></tr>
<tr><td>Plasma resistive decay time, L_p/R_p(s)</td><td>10.6</td><td>36.7</td></tr>
<tr><td><u>MAGNETS</u></td><td></td><td></td></tr>
<tr><td>Total ohmic power to coils, $P_{\Omega c}$(MW)</td><td>25.0</td><td>39.8</td></tr>
<tr><td>Toroidal-Field Coils (TFC)</td><td></td><td></td></tr>
<tr><td>• current-center radius/thickness, $r_{c\phi}/\delta_{c\phi}$(m)</td><td colspan="2">0.63/0.05</td></tr>
<tr><td>• current density,[e] $j_{c\phi}(MA/m^2)$</td><td>6.7</td><td>8.5</td></tr>
<tr><td>• power consumption, P_{TFC}(MW)</td><td>3.0</td><td>4.8</td></tr>
<tr><td>• mass, M_{TFC} (tonne)</td><td colspan="2">17.4</td></tr>
<tr><td>Poloidal-Field Coils (PFC)</td><td></td><td></td></tr>
<tr><td>• current center radius/thickness, $r_{c\theta}/\delta_{c\theta}$(m)</td><td colspan="2">0.80/0.30</td></tr>
<tr><td>• curent density,[e] $j_{c\theta}(MA/m^2)$</td><td>6.7</td><td>8.5</td></tr>
<tr><td>• power consumption, P_{PFC}(MW)</td><td>21.9</td><td>35.0</td></tr>
<tr><td>• mass, M_{PFC}(tonne)</td><td colspan="2">126.0</td></tr>
<tr><td>• PFC inductance,[f] $L_c(10^{-6}\ H)$</td><td colspan="2">2.01</td></tr>
<tr><td>• coil current, $I_{c\phi}$(MA)</td><td>10.14</td><td>12.80</td></tr>
<tr><td>• solenoid flux, $L_cI_{c\phi}$(Wb)</td><td>20.3</td><td>25.7</td></tr>
<tr><td>• coil L/R time (s)</td><td colspan="2">9.4</td></tr>
</table>

(a) Plasma radius taken at T = 0 surface and is greater than radius of reversal layer, first-wall and plasma radius taken as equal.

(b) Taken to be equal to first-wall radius and represents an upper bound on experimental/test volume.

(c) Bessel-function model pressure profiles assumed, $P(r) = J_o^2(\mu r)$, with n(r) and $T(r) = J_o(\mu r)$. A resistance form factor of $g_{OHM} = 4.7$ was computed to be consistent with these profiles.

(d) The experimentally calibrated scaling,[1] $\tau_{ce} = 0.085\ I_\phi^{1.25}(MA)r_p^2$, was used with $\tau_{pi} = 4\tau_{ce}$ and τ_E computed as the global energy confinement time. No anomalous ion heating was assumed.

(e) Current density in TFCs and PFCs equal, magnitude set by PFCs size and power consumption.

(f) Taken as $L_c = \mu_o R_T[\ln(8R_T/r_{c\theta}) - 2.0]$.