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ENGINEERING DESIGN OF A COMPACT RFP REACTOR (CRFPR)*

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ABSTRACT

The results of a previously-reported comprehensive parametric systems analysis of the Reversed-Field Pinch (RFP) with minimum-cost systems that would operate with resistive water-cooled copper coils and higher first-wall neutron current ($15\text{--}20\text{ MW/m}^2$). These minimum-cost, "compact" RFP reactors (CRFPRs) have system power densities that can be comparable with fission power plants and, therefore, are 10-30 times smaller than most superconducting approaches. Reported herein are initial results of a conceptual engineering design of key fusion-power-core (FPC) subsystems.

I. INTRODUCTION

A comprehensive economic systems model of the reversed-field pinch (RFP) reactor was used to analyze the complete spectrum of possible reactor designs.¹ Reactor systems with fusion-power-core (FPC, i.e., first wall, blanket, shield, and magnetic coil) masses more than an order of magnitude less than superconducting systems projected considerable economic savings.^{1,2} Superconducting fusion systems typically have FPC mass utilizations of 3-10 tonne/MWt, which is to be compared with the total mass of a fission plant (excluding concrete but including rebar) of ~ 10 tonne/MWt. Compact reversed-field pinch reactors (CRFPRs) of minimum cost¹ typically would operate with FPC mass utilizations of 0.4-0.5 tonne/MWt for a DT-fueled machine, resulting in capital cost savings by factors of 1.5-2.0 compared to the low-power-density superconducting systems.^{1,2} Projected total direct costs (TDC) become relatively insensitive to large changes in FPC unit costs and uncertainties in physics and technology because of the considerable reduction in FPC cost compared to the Balance of Plant (BOP); instead of the Reactor Plant Equipment (RPE) being 50-70% of the TDC, as projected for most

of the other approaches, the CRFPR projects $\text{RPE/TDC} \leq 0.3$.

Other CRFPR characteristics are expected to influence the system design positively. Low FPC mass allows offsite production and complete pre-installation testing of the operational unit, potentially leading to shorter construction times and more rapid "block" maintenance. In fact, the entire first wall and blanket structure (280 tonne), possibly along with the complete toroidal-field-coil set (140 tonne), would be removed as a single 420-tonne unit. The unique aspects of RFP confinement, which are discussed below, allow the use of resistive magnet coils while maintaining an efficient reactor; massive superconducting coils and shields are eliminated, resulting in a considerable reduction in FPC mass utilization and increase in system power density (i.e., total thermal power divided by the FPC volume) to values similar to a fission power reactor ($10\text{--}15\text{ MWt/m}^3$). In addition, the plasma current density is sufficient to heat the plasma to ignition solely by Ohmic dissipation. These features minimize the long-lead development of systems such as large superconducting magnets, auxiliary plasma heaters, and remote maintenance schemes for massive structures.

The advantages anticipated for the CRFPR are not achieved without penalty and/or uncertainty. The first-wall neutron current is increased by factors of 3-4 to $15\text{--}20\text{ MW/m}^2$, with average surface heat fluxes of $4\text{--}5\text{ MWt/m}^2$. Plasma limiters, refueling, and the need for magnetic divertors remain to be fully quantified. High-power-density ($\geq 200\text{ MWt/m}^3$ in the coolant) breeding blankets are required, but a considerable reduction in the total mass of breeding material results. Although the first-wall neutron loading is 3-4 times that postulated for most low-power-density systems, the total first-wall/blanket (FW/B) mass that must be discarded or recycled because of radiation damage remains the same for most fusion systems ($\sim 200\text{--}400$ tonne/yr), with only a fraction of the low-power-density system being removed, while the entire FW/B of the

*Work performed under the auspices of the USDOE

compact system is removed during a similar maintenance interval. Radiation effects to room-temperature copper coils and (inorganic) electrical insulators that are protected by a thin (~ 0.6 -m-thick) heat-recovering/tritium-breeding blanket must also be further quantified. Startup, burn sustenance, assembly/maintenance procedures, and general integration of the FPC into an efficient thermal-conversion/BOP system represent other areas presently receiving attention for the compact option. The ongoing work³ reported herein focuses on these key engineering subsystems in preparation for a complete reactor systems design. This effort is summarized in Sec. II. Summary conclusions are given in Sec. III.

II. CRFPR REFERENCE DESIGN POINT

A. Theoretical and Experimental Basis

The RFP is an axisymmetric toroidal device⁴ in which the primary confinement field, B_θ , is poloidal and is generated by a toroidal current, I_ϕ , flowing in the plasma, as is illustrated in Fig. 1. The RFP plasma supports a toroidal bias field, B_ϕ , to stabilize sausage

FIG. 1. Sample magnetic-field minor radial profiles for the RFP and tokamak with a reactor cross section showing the plasma, toroidal-field coil (TFC) and poloidal-field coil (PFC).

($m = 0$) and elliptical ($m = 2$) distortions. Grossly unstable MHD modes with wavelengths longer than the minor radius, r_w , of an electrically conducting shell are stabilized by the shell for short times and by feedback coils for longer times. If the toroidal bias field is slightly reversed near the plasma edge, $r_p \leq r_w$, the resulting high magnetic shear in the plasma edge region is sufficient to stabilize local pressure and current-driven modes as specified by the Mercier criterion.⁴ This stabilization is expected to occur at relatively high values of $B_\theta \sim 0.3$ - 0.5 , low values of toroidal fields at coils, and for plasma currents that are not limited by the tokamak $q \equiv (B_\phi/B_\theta)(r_p/R_T) > 2$ -3 constraint. A sample stable RFP field profile is compared in Fig. 1 to that for a tokamak. A theory of minimum-energy plasma states, described by Taylor⁴, specifies the desired RFP state having $F < 0$ and $1.2 < \theta < 1.6$, where $F \equiv B_\theta(r_w)/\langle B_\theta \rangle$, $\theta \equiv B_\theta(r_w)/\langle B_\theta \rangle$, and the average toroidal field is $\langle B_\theta \rangle \equiv (2/r_w^2) \int_0^{r_w} B_\theta r dr$. Experiments⁵ have verified the strong tendency of the plasma to remain near $F \approx - (0.1-0.2)$ and $\theta \approx 1.5$ with $B_\theta = 0.2$; these conditions are used in the reactor calculations. Experimental plasma discharges have also exhibited classical plasma resistivity which admits the attractive option of Ohmic heating of a D-T plasma to ignition. The experimentally derived energy confinement time for an Ohmically-heated plasma, $\tau_E(\text{OHMIC})$, is also consistent with the energy loss rate required for the plasma to dispose of the classical Ohmic heating input. The energy confinement time required for an economically optimized reactor $\tau_E(\text{OPT})$, is shown¹ to be 20-50 times less than $\tau_E(\text{OHMIC})$ at reactor temperatures (15-20 keV).

Finally, a remarkable property of the RFP plasma is the ability to transform toroidal plasma current, I_ϕ , into poloidal current and internal toroidal flux, B_ϕ ($I_\phi = I_0 + B_\phi$). This "dynamo" action dissipates little energy beyond that required by classical Ohmic losses. The RFP configuration can then be maintained indefinitely through this dynamo action, given sufficient transformer flux to drive I_ϕ . Furthermore, a unique steady-state current drive has been theoretically identified;⁶ this "F-0 pumping" uses the dynamo action and operates by electrically cycling the TFC and PFC coils in phase at low frequency (kHz).

B. Design-Point Determination

The unified parametric systems model described in Ref. 1 surveyed potential reactor design points using a methodology that predicts those systems with the lowest cost-of-electricity (COE). A 1000-MWe(net) base case was selected and subjected to a parametric

analysis that varied plasma transport, beta, blanket thickness, plasma density and temperature profiles, normal versus superconducting coils, D-T versus catalyzed-DD fuel, net-electric output, and first-wall/blanket lifetime.^{1,3} Using the base case, the transport scaling, τ_E , was varied parametrically to give a design point that minimizes the COE. The resulting value for confinement time, $\tau_E(\text{OPT})$, equals $\sim 1/8$ of the prediction of empirical (Alcator) tokamak transport for the design reported in Table I.

The effect of varying the poloidal beta, β_θ (which in RFPs nearly equals the total beta) is shown in Fig. 2 for the fully-optimized DT-fueled design. The high-resiliency of the design to changes in beta is a direct consequence of the reduced FPC size and the reduced role the FPC plays in determining the TDC. The dependence of COE on assumed first-wall/blanket lifetime for the DT/CRFPR design given in Table I is depicted in Fig. 2. The design point uses a wall lifetime of 15 MW yr/m² ($L_w = 19.5$ MW/m²) and can operate with a factor of ~ 2 reduction in first wall/blanket lifetime (and L_w) without seriously increasing the COE. The degraded $L_w = 8\text{--}10$ MW/m² system, however, has a FPC that is increased in mass by 1,240 to 2,200 tonnes, which still represents a considerable reduction compared to the superconducting approaches.

TABLE I
PARAMETER SUMMARY FOR FULLY-OPTIMIZED
1000-MWe DT-FUELED CRFPR DESIGN

PARAMETER	VALUE (a)
First-wall radius, r_w (m)	0.75
Major toroidal radius, R_T (m)	3.80
Minor system radius, r_s (m)	4.0
Blanket thickness, Δb (m)	0.60
First-wall/blanket mass (tonne)	277
Fusion-power-core mass, M (tonne)	1,240
Mass utilization, M/P_{TH} (tonne/MWt)	0.36
Toroidal plasma current, I_p (MA)	18.4
Poloidal field at plasma, B_θ (T)	5.2
Plasma temperature, T (keV)	20.0(10.0)
Plasma density, n ($10^{20}/\text{m}^3$)	3.4(6.7)
Energy confinement time, τ_E (s)	0.23
Alcator coef. τ_E/nr_p^2 (10^{-21} s m)	1.37(0.76)
Poloidal beta (Maxwellian species), β_θ	0.2
Total thermal power, P_{TH} (MWt)	3,400
FPC power density, P_{TH}/V_c (MW/m ³)	12.0
Recirculating power fraction, c (b)	0.17
Neutron first-wall loading, I_w (MW/m ²)	19.6
Unit total cost, UTC (\$/kWe)	1,510
Cost of electricity, COE (mills/kWh)	42.5

(a) $T(r) = \text{CONST } (T(r) = J_p(ar))$.

(b) Includes 0.07 for auxiliary needs and ~ 0.1 to supply Ohmic losses to coils and plasma.

FIG. 2. Dependence of the Table-I CRFPR design point on the assumed poloidal beta, β_θ and first-wall lifetime.

Economically optimal system power densities typically occur when the FW/B must be replaced annually; low values of system power density or first-wall neutron loading increase system mass while excessive first-wall loadings require frequent changeout, reducing plant availability.

C. Engineering Subsystems

A scaled layout of the 1000-MWe(net) CRFPR design summarized in Table I is given in Fig. 3. A summary discussion of the major FPC subsystems is presented in this subsection, with a comprehensive listing of key engineering parameters being given in Table II.

The plasma is contained within a separate 20-mm-thick first wall shell composed of a water-cooled copper alloy (M2C/AMAX). The heat-transfer surface consists of 774 coolant tubes (24-mm-diameter, 3-mm-thick) encircling the minor circumference of the plasma. These tubes must withstand an average surface heat flux of 5 MW/m², are designed to operate at heat fluxes as high as 9 MW/m², and are capable of withstanding 10^6 thermal cycles during the structure lifetime. Subcooled water at 8.3 MPa

FIG. 3. Scaled layout of the 1000-MWe(net) DT/CFETR design point reported in Tables I and II.

(1200 psia), flowing at 10 m/s enters the first-wall shell at 423 K and exits at 503 K, resulting in a maximum local structural temperature of 590 K.

Both pumped-limiter and magnetic-divertor particle control schemes are being considered, with the major design effort to date being devoted to the former.³ For plasma-edge conditions that allow 90% of the Ohmic and alpha-particle power generated within the plasma to be radiated uniformly to the first-wall/limiter surfaces, 24 poloidal pumped limiters covering 30% of the first-wall area would allow 21% of the plasma particle loss to flow into the limiter slot while maintaining the energy density onto the limiter leading edge below 6 MW/m². The plasma-edge models have not yet been developed sufficiently to permit accurate estimate of sputter erosion, although this rather than heat-transfer per se is viewed as the premier first-wall/limiter problem. Immediately behind this first-wall/pumped-limiter section would be a "second wall", defining with the first-wall a vacuum lumen; this plenum connects to a vacuum enclosure by a radial inner-blanket duct positioned in the equatorial plane on the outboard side of the torus. Eliminating this second wall in favor of radial ducts which cut through the blanket in the minor circumference behind each limiter is also being explored.

A 0.5-m-thick blanket (with 60% enriched ⁶Li) uses flowing Pb₈₃Li₁₇ liquid-metal for both tritium breeding and coolant because of the high blanket power densities (< 200 MWt/m³). This blanket yields a tritium breeding ratio of 1.11 and multiplies the incoming 14.1-MeV-neutron energy by a factor of 1.3. In order to remove heat from the "second wall", that also forms the boundary of the blanket, water cooling may also be required, although PbLi cooling of this critical area may also be possible at somewhat increased MHD pumping losses. Outside the PbLi blanket is located a 0.1-m-thick "third-wall" shield consisting of alternating layers of B₄C/W. This hot shield increases the thermal capture efficiency of this thin blanket from 0.910 to 0.985 of the total thermal power production and would also be cooled by the PbLi. Flowing the PbLi coolant around the minor torus circumference follows the strong poloidal field and opposes the much weaker reversed toroidal field. Accounting for MHD ducting losses incurred by flow outward in the major radial direction, the pumping work can be made less than a few percent of the net electric output without improving the conductivity ratios by incorporating insulating coatings into the blanket structure.

Surrounding the hot-blanket structure are 24, 0.15-m-thick TFCs composed of a water-cooled room-temperature copper alloy. These coils provide a reversed-field of only 0.68 T (~ 1 T at the conductor) and require 17.9 MWe to sustain the Ohmic heating. The TFCs must also produce the initial toroidal bias field, B₀, required before plasma initiation. This field is also taken as ~ 0.7 T, which requires the RFP configuration be established at ~ 20% of the ultimate plasma current of 18.4 MA. The remainder of the toroidal flux within the plasma would be provided by the plasma dynamo effect using the PFC circuit as I_φ is increased, minimum-energy F and θ values are maintained, and ignition occurs. The TFC set stores only 30 MJ using this startup scenario.

The toroidal plasma current, I_φ, is induced by the PFC system. The PFC system is divided into the Ohmic Heating (OH) system, which provides the flux change required to drive the plasma current, and an Equilibrium Field (EF) system, which provides a nearly uniform vertical field required to maintain toroidal and up/down plasma equilibrium. Equal-current-carrying conductor locations are shown for both the OH and EF coils in Fig. 3. PFC conductor sizes are approximate, subject to the ongoing electromechanical design effort. The operational sequence for the PFC (OHC + EFC) circuit is illustrated in Fig. 4, which

TABLE II KEY CFFR ENGINEERING PARAMETERS

FIRST WALL		POLOIDAL-FIELD COILS	
Composition: 60 v/o AMAX Cu(NiZr, 0.06% Ni, 0.15% Zr, 0.4% Cr by weight)		Composition: 70 v/o Cu, 20 v/o PCASS, 10 v/o E ₂ O	
40 v/o H ₂ O (Coolant)		OHC/EFC total mass (tonne)	
Overall thickness (mm)		OHC/EFC total current during burn (MA)	
Total first-wall mass (tonne)		OHC/EFC current density (MA/m ²)	
Number of coolant tubes facing plasma		OHC/EFC dissipated power during burn (MW)	
Tube length/diameter/wall thickness (mm)		PFC maximum stored energy (GJ)	
Subcooled water pressure (MPa/psia)		Maximum dpa/yr	
Coolant velocity (m/s)		Peak resistivity increase (Z/yr)	
Inlet/outlet/maximum wall temperature (K)		POLOIDAL-FIELD (STARTUP) CIRCUIT PARAMETERS	
Average/maximum design surface heat flux (MW/m ²)		Electrical Grid Requirement	
Average power density (MW/m ³)		Maximum Power, P _{GRID} (MW)	
Maximum dpa/yr		Voltage, V _{GRID} (kV)	
Average transmutation rates Ni/Zn (Z/yr)		EFC Characteristics	
Electrical resistivity increase (Z/yr)		Turns, N _{FP}	
Blanket/THIRD WALL		Maximum lead current, I _{FP} (kA)	
Composition:		Maximum voltage, V _{MAX} (kV)	
Blanket:		OHC Characteristics	
Third wall:		Turns, N _{OP}	
(Alternating Layers) 8C v/o W, 10 v/o PCASS, 10 v/o PbLi		Maximum lead current, I _{OP} (kA)	
Blanket thickness (m)		Maximum voltage, V _{MAX} (kV)	
Third wall thickness (m)		Circuit Timing	
Total blanket mass, without coolant (tonne)		Charge time, τ _c (s)	
Neutron multiplication, M _n		Initial plasma rise, τ _R (s)	
Tritium breeding ratio, BR		Final plasma rise, τ _L (s)	
Thermal energy recovery efficiency, ε _g		FORCED-LIMITER PARAMETERS	
PbLi flow rate (kg/s)		Configuration	
Inlet/outlet/maximum wall temperature (K)		Number of limiters, N _L	
Pump work (MJ/Z)		Edge-plasma recycle, R _e	
Peak power density (in LiPb coolant, MW/m ³)		Fraction plasma energy loss radiated, f _{RAD}	
Average power density (MW/m ³)		Design heat flux onto limiter, q ₀ (MW/m ²)	
TOROIDAL-FIELD COILS		Limiter coverage fraction, f _L /(f _L + f _W)	
Composition: 70 v/o Cu, 20 v/o PCASS, 10 v/o E ₂ O		Toroidal extent of limiter, L _L (m)	
Number, N		Fraction particles under limiter, f _p	
Coil radius/width/length (m)		Fraction energy under limiter, f _E	
Mass per coil (tonne)			
Average toroidal field (on-axis/at conductor, T)			
Total stored energy (MJ)			
Total current per coil (MA)			
Toroidal current density (MA/m ²)			
Ohmic dissipation during burn (MW)			
Maximum dpa/yr			
Peak resistivity increase (Z/yr)			

depicts the response of the plasma current during startup. The OHC would be initially charged from the electrical power grid during the period, τ_c , to a current level $I_{OH} = 29.0$ MA; a maximum electrical grid power, P_{GRID} , is required. The charged OHC would then be connected across the EFC after the initial toroidal bias field, $B_{\theta 0}$, is created. At this point the plasma is simultaneously initiated, trapping the $B_{\theta 0}$ field, while a transfer resistor is connected across the OHC/EFC sets. During this rapid current-rise phase, τ_R , the electrical power grid is decoupled from the PFC system. The exponential, resistive decay continues until $V_{OH} = V_{GRID}$, when $I_{\theta} = 11.0$ MA; the electrical grid is again applied to the system, which slowly drives the system to full current over the period τ_L . Because the system is strongly inductive, the plasma current varies linearly from 11.0 to 18.4 MA as P_{GRID} linearly increases. Minimizing P_{GRID} requires $I_{OH} = I_{OH} + I_{FP}$, which defines $I_{OH} = 29.0$ MA and $I_{OH} = 17.9$ MA for $I_{FP} = 11.1$ MA. The recirculating power required to drive the OHC/EFC sets during the burn is 107 MW as

reported in Table II along with example circuit characteristics. Accounting for both Ohmic losses and nuclear heating, a total of 175 MWt must be removed as low-grade heat from the TFCs and PFCs.

The entire first wall, blanket, and TFC set (FW/B/TFC) would be fabricated as a single unit having a total mass of ~ 400 tonnes; nearly 50% of this mass is associated with the tungsten in the third-wall shield and may be replaced by a less massive (and less efficient) material. The entire EFC (FW/B/TFC/PFC) would be located in a vacuum enclosure. Maintenance operation requires removal of the vacuum cover, lifting the top halves of the OHC (200 tonne) and EFC (200 tonne) sets, thereby exposing the 400 tonne FW/B/TFC torus. For the design first-wall loading (~ 10 MW/m²) and assumed first-wall life (~ 15 MW/yr/m²), this torus would be replaced annually, although the TFCs most likely would be recycled. The PFC set is designed to operate for an extended period of time (> 10 yrs). Even if the TFC were not recycled, at 50 \$/kg for fabricated copper

FIG. 4. Operations sequence for the poloidal field coil circuit showing circuit connections and currents versus time for the plasma, OH and EF coil systems.

coils, this would add only ~ 1.0 mill/kWeh to the COE, although the long-term radwaste problem remains to be fully assessed [(e.g., ^{63}Ni (93 yr) and ^{60}Co (5.6 yr)].

III. SUMMARY AND CONCLUSIONS

Although considerably more CRFPR design work and innovation remains, generally the positive prognoses suggested by the parametric systems optimization of Ref. 1 remains largely substantiated by the conceptual engineering designs of key FPC subsystems made to date. Specifically, the advantages of RFP confinement (i.e., Ohmic heating to ignition, moderate-to-high betas, physics-decoupled aspect ratio, low fields at and currents in coils, etc.) have been translated into a conceptual fusion power system that in terms of FPC power density, mass utilization, and fraction of TDC approaches values deemed necessary for any economic nuclear steam supply system. These favorable projections occur not without major uncertainties in both physics (e.g., particle/energy transport, maintenance of acceptable betas at higher temperature and currents, maintenance of the dynamo effect in reactor-grade plasma, use of the dynamo effect in conjunction with the F- θ diagram to give non-inductive current

drive) and engineering (high-heat-flux first walls, high-power-density liquid-metal blanket, first-wall erosion control, acceptable radiation lifetime of FPC materials). None of these problems and uncertainties have been identified as insoluble; the continuance of progress on the experimental front should resolve favorably most of these physics issues; solutions to or resolution of key engineering issues should occur through modest extensions of existing technologies; and a research and D&T program directed toward the attainment of competitive fusion power can be envisaged that is little affected by the FPC cost per se. Lastly, and most importantly, the viability of the CRFPR design does not depend crucially on the attainment of a specific ensemble of parameters within a very narrow range of parameter space: the CRFPR design has shown an amazing resiliency in recovering very competitive costs for a wide range of assumed physics and engineering parameters.

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