TITLE: CONCEPTUAL DESIGN OF A REVERSED-FIELD PINCH FUSION NEUTRON SOURCE

LA-UR-88-814

TI88 007895

AUTHOR(S) C. G. Bathke, CTR-12
R. A. Krakowski, CTR-12
R. G. Manzanares, CTR-4
R. L. Miller, CTR-12
K. A. Werley, CTR-12

SUBMITTED TO International Symposium on Fusion Nuclear Technology
Tokyo Prince Hotel
Tokyo, Japan

DISCLAIMER

This report was prepared as part of work sponsored by an agency of the United States Government. It is hereby stated that the United States Government and any agency thereof, makes no warranty, expressed or implied, for the accuracy, completeness, or legal liability, or for any consequences arising from the use of the information contained herein. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

By acceptance of this article, the publisher recognizes that the U.S. Government retains a nonexclusive, royalty free license to publish or reproduce the published form of this contribution, or to allow others to do so, for U.S. Government purposes.

The Los Alamos National Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy.

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

CONCEPTUAL DESIGN OF A REVERSED-FIELD PINCH FUSION NEUTRON SOURCE*

C. G. Bathke, R. A. Krakowski, R. G. Manzanares, R. L. Miller, and K. A. Werley,

Los Alamos National Laboratory, Los Alamos, NM 87545 USA

ABSTRACT

The conceptual design of an ohmically-heated, reversed-field pinch (RFP) operating with a 5-MW m^2 steady-state DT fusion neutron wall loading while generating $\sim 100\text{-MW}$ total fusion power is presented. These results are also useful in projecting the development of an economic source of DT neutrons for large-volume ($\sim 10\text{ m}^3$) fusion nuclear testing.

* Work done under auspices of US DOE

1. INTRODUCTION

A strong experimental database is evolving from a number of relatively small reversed-field-pinch (RFP) devices.¹ Consequently, the design and construction of the next-step RFPs are well under way in both the US and the European Economic Community.² A recent study of the commercial prospects of the RFP as a high-power-density, compact fusion reactor³ has been completed, and a strong economic potential is shown should the physics established by existing RFPs extrapolate through the next-step devices to the reactor regime. Preliminary scoping studies of RFPs with characteristics between these next-step devices and the reactor regime recently examined the potential of a RFP ignition/burn device as a steady-state source of DT fusion neutrons⁴.

These results are used to characterize the RFP as a fusion test facility (FTF). This steady-state device would function as a low-to-moderate-Q, driven or marginally ignited FTF with the main goal being the generation of fusion-relevant DT neutron currents ($I_{nn} \sim 1\text{-}10\text{ MW/m}^2$) from plasmas that are sufficiently small to operate with a total fusion power of $\sim 100\text{ MW}$ and without large expenditures in driver power. Central to the viability and/or feasibility of this compact approach is the ability to manage heat and particle fluxes in an RFP that differs little in size from the next-step devices².

The basic approach adopted by this FTF-RFP study first developed a quantitative understanding of the available operating space. Cost estimates were also made in the early stages of these analyses to provide guidance. Upon selecting a design point from the steady-state parametric analysis,¹ a two-dimensional vacuum magnetics computation established the position of the equilibrium-field (EF) and ohmic-heating (OH) coils, subject to the usual constraints imposed by equilibrium and startup (i.e., plasma breakdown, OH-coil stresses and power) requirements. A one-dimensional RFP transport model was used to estimate radial density and temperature profiles, particularly as determined by the impurity-seeded, high-radiation plasma needed to smooth heat fluxes in these compact systems. Both the detailed coil configuration and plasma profiles were then used in the plasma-circuit simulation to determine the ohmically-heated startup transient leading to oscillating-field current drive (OFCD)⁵ and steady state. With the basic parameters for the plasma, magnetics (coil sizes, powers, forces), impurity and fueling control, and current drive established,¹ including all vacuum shielding and cooling requirements, a first estimate of the maintenance and testing requirements and capabilities has been formulated. Given below are the results of each of these subsystem analyses, which combine to give a quantitative mechanical and operational definition of the RFP as a facility for fusion nuclear testing.

2. RESULTS

2.1. Parametric Systems Studies

Parametric results⁴ are expressed as contours in a plasma current-radius (I_ϕ - r_p) phase space, which is particularly useful in that it measures indirectly two major cost items: (a) coils and power supplies (I_ϕ) and (b) the torus (r_p). The parametrics model generates on a plot of I_ϕ versus r_p lines of constant neutron wall loading, I_w (MW/m²), average first-wall heat flux, q_s (MW/m²), total fusion power, P_F (MW), constant electron or ion temperature, $T_{i,e}$ (keV), constant electron density, n_e (m⁻³), constant electron streaming parameter, $\xi = v_D/v_{the}$, constant current density, j_ϕ (MA/m²), constant plasma loop voltage, V_ϕ (V), constant Lawson parameter, $n\tau_E$ (s·m³), and ignition parameter. Selecting the following range for key variables defines the design window used to guide this study: $I_w = 1$ -5 MW/m², $P_F \leq 100$ MW, and $q_s \leq 5$ MW/m². Table I lists the main parameters for an $r_p = 0.3$ -m device operated at $I_w \sim 5$ MW/m²; Reference 4 reports the sensitivity of this design to the main physics parameters listed in Table I.

2.2. Magnetics Configuration

A two-dimensional vacuum magnetics model was used to establish the details of a closely coupled OH- and EF-coil sets subject to the usual equilibrium, stress, and power constraints, with the latter two being estimated from a plasma circuit time-dependent simulation. The poloidal-field (PF) coils were nominally positioned a distance from the plasma, Δb , equal to the plasma radius, $r_p = 0.3$ m. Both OH and toroidal-field (TF) coils are fabricated from aluminum alloy to reduce activation at the expense of ~50% more power consumption. Since the EF-coils represent the main steady-state power drain, water-cooled copper alloy was selected for the EF-coils. Figure 1 gives a torus cross section. Routine (i.e., daily to monthly) maintenance and servicing would be through horizontal motions outboard of the torus, with the torus divided into relatively independent quadrants separated by four toroidal-field divertors (Sec. 2.3). Installation and longer-term maintenance of the OH and EF coils would occur by vertical access. Coil positions are optimized so that magnetic flux from the back-biased OH-coil is excluded from the plasma chamber to a level that meets maximum vertical-field constraints for efficient plasma breakdown. The main PF-coil parameters are listed on Table II. As seen from Fig. 1, these constraints are met by a PF-coil set of sufficient outboard openness to allow horizontal removal of either divertor assemblies or torus quadrants including the TF coils.

The TF coils are aluminum-alloy bands positioned at the minor radius immediately outside the $\Delta b = 0.3$ -m-thick shield/test-cell region. The TF coils generally operate at low fields; during the startup phase, most of the toroidal flux within the plasma is provided from the PF-coil set through the RFP dynamo.¹ The magnitude of the radial magnetic-field ripple relative to the poloidal field, $\Delta B_R/B_\theta \leq 0.003$, is chosen to assume acceptably small magnetic islands relative to the distance between the toroidal-field reversal layer and the separatrix. Applying this constraint leads to the TF-coil design summarized in Table III and illustrated on Figs. 1 and 2. The moderate centering and overturning forces on the TF-coils would be supported by a toroidal strong-back that also serves as the outer surface of each blanket quadrant.

The divertor-field (DF) coils represent the last major component of the FTF RFP magnetics design. Each of four poloidally-symmetric TF divertors (Sec. 2.3) consists of a single TF nulling coil with flanking coils positioned at each side to minimize the perturbation of the reversed toroidal field. Table III also gives the main parameters for the DF-coils.

2.3 IMPURITY CONTROL

Central to the operation of a compact, high-power-density RFP, be it an ignition test device, an FTF, or a commercial reactor, is the control of high heat and particle fluxes. A number of options for controlling the particle and energy exhaust was examined,¹⁶ including an armored first-wall, a pump limiter, and a range of magnetic divertors. Since the poloidal magnetic field is dominant in the plasma edge, poloidally-symmetric limiters and divertors are required to provide magnetic field line connection lengths that are sufficient for radial diffusion of energy and reduced peak heat fluxes. Furthermore, diversion of the minority (toroidal) field requires less power and minimally perturbs the plasma. A closed TF divertor was found to concentrate the heat and particle fluxes to the collector plate. The open divertor configuration shown in Fig. 3 avoids this drawback by moving the collector plate closer to the field null, at this point magnetic flux surfaces are expanded and poloidally symmetric, as contrasted to the poloidally-asymmetric concentration of field lines that presents problems for the closed divertor. The open divertor geometry also allows room for a larger collector plate. The plate heat flux without plasma radiation is within a factor of 3 of the design heat flux of $q_{plv} = 1.1/11 \text{ MW/m}^2$; a reduction in the plate heat flux is easily achievable by increasing radiation losses by impurity injection (e.g., 0.1% Xe).

As a result of the soft beta limit¹ observed in RFPs, plasma parameters and the energy confinement are unaffected by the addition of impurities. This behavior is in marked contrast to

devices not operating at a beta limit in which injecting high-Z impurities increases both the radiation and the total energy loss, and, therefore, degrades energy confinement. The divertor need only remove sufficient impurities so that the impurities in the core plasma can be controlled at the design levels. While an open divertor configuration does not entrain impurities, it must nevertheless physically isolate the hot core plasma from the collector plate to protect both the plasma from neutral atoms and a possibly uncontrolled source of impurities, as well as protecting the plate from erosion. The minimum separation distance needed to isolate the collector plate from the core plasma is about four neutral-atom ionization mean-free-paths. Typical parameters for the plasma in front of the collector plate ($T_e \sim T_i \sim 10$ eV and $n_e \sim 10^{21} \text{ m}^{-3}$) give a mean-free-path of 0.2 mm. The design shown in Fig. 3 locates the plate 15 mm from the core plasma, which is 18 mean-free-paths.

Radial core-plasma, radial edge-plasma, and axial edge-plasma transport calculations for the open divertor give the key parameters for the open-divertor design: peak edge-plasma density and temperatures are $n_e \sim 1.2 \times 10^{20} \text{ m}^{-3}$, $T_e \sim 100$ eV, and $T_i \sim 250$ eV, with core-plasma and total radiation fractions of 0.816 and 0.925, respectively. The density and temperature near a tungsten-coated divertor plate are about 10^{21} m^{-3} and 10 eV, respectively, which results in negligible erosion; the heat flux normal to the plate is 3.5 MW m^{-2} .

Plasma density and temperatures near the water-cooled ferritic steel first wall for the divertor configuration described above are 10^{20} m^{-3} and 1 eV, respectively, with negligible transported heat flux and a radiation heat flux of about 2 MW m^{-2} . Wall erosion by plasma particles is negligible; erosion by charge-exchange neutral atoms, however, is 0.11 mm/yr. Lowering the separatrix temperature (e.g., lower heat flux or higher density), however, has a large effect on the first-wall parameters. A high-Z coating on the first wall would eliminate erosion concerns.

A geometry calculation in conjunction with a heat-flux constraint determines the shape of a divertor plate (Fig. 3), which is located at an θ angle relative to a field line. The thermal-hydraulics design was carried out on the inboard side of the divertor plate, where the heat flux is highest and the space is minimum. The heat flux normal to the divertor plate accounts for transport along field lines, flux-surface expansion, and radiation. The flux-surface expansion includes the inverse radial dependence of the poloidal field. The radiation heat flux is modeled as consisting of two parts: (a) a radial flux consisting of the core-, edge-, and half of the divertor-radiated power, and (b) the other half of the divertor-radiated power being deposited directly on the collector plate area. The simplest divertor-plate coolant arrangement was selected, which has poloidally symmetric coolant headers

located external to the divertor nulling coil (Fig. 3). These headers supply and receive coolant water through copper tubes that form the divertor surface. The total coolant path between headers is less than half a meter, and the copper tubes are coated with tungsten to form the collector plate surface.³ The collector-plate thermal-hydraulic parameters are given in Table IV.

2.4 CURRENT DRIVE

An inductive but oscillatory (i.e., not consumptive of electromagnetic flux) means of steady-state current drive has been proposed for the RFP.⁵ Intrinsic plasma processes related to turbulence and/or resistive instabilities generate voltage and current within the plasma in order to increase or reduce poloidal flux, so that the magnetic flux linkage or helicity is held constant and the plasma resides near a minimum energy state. This nonlinear coupling between plasma and magnetic fields is strong in RFPs and can be used to rectify current oscillations in external coils into a net steady-state current within the plasma. A power balance imposed at the plasma surface, a definition of the plasma internal magnetic energy, and a positive Faraday's Law ($V_\phi = d\psi/dt$) yields an expression for the toroidal voltage around the plasma, V_ϕ , in terms of the poloidal voltage, V_θ , and the plasma geometry (r_p, R_T). Oscillations of $V_{\phi,\theta}$ in proper phase at frequency less than $\sim 2\pi/\tau_R$ can give a net time-averaged current, I_ϕ , with $V_\phi = 0$ (i.e., no net flux change), where τ_R is the instability relaxation time responsible for poloidal-flux generation. Hence, a non-intrusive means to drive current using primarily the main confining coil system to drive low-frequency, low-amplitude plasma oscillations becomes possible.

An assessment of the OFCD engineering efficiency requires the modeling of the circuit elements external to the plasma to account for all power dissipation. Circuit equations are derived^{3,5} for poloidal and toroidal current paths and are labeled (θ, ϕ) according to the current direction. The circuit elements simulated are the plasma, first wall, TF coils, a portion of the OH-coil windings, the blanket, a primary EF-coil set, and a secondary EF-coil set. Calculations with an electrically continuous first wall indicate a need for resistive breaks or gaps. The model for circuit elements with electrical gaps is described in Ref. 3 or 5.

The OFCD results parameterize reactive and dissipative powers as a function of $\delta\phi/\phi_0$. The operating window of $\delta\phi/\phi_0$ is bounded above and below because of a loss of field reversal. The upper bound is the result of oscillations in ϕ being too large in amplitude at a shallow reversal ($I = B_z(r_p)/B_z = 0.1$). The lower bound is the result of oscillations in I_ϕ being too large ($\sim 5\%$) and, hence, the oscillation in the pinch parameter ($\psi = B_z(r_p)/B_z = 1$), being too large.

which results in a loss of reversal because adherence to a near-minimum-energy state (e.g. the $I'-\Theta$ curve) is required. The $\delta\phi/\phi_0$ operating window completely disappears below 25 Hz. A summary of typical OFCD parameters is given in Table V.

2.5 MAINTENANCE AND TESTING GEOMETRY

2.5.1 General Layout

A number of key design choices determine the main features of the FTF/RFP test geometry. As seen from Fig. 4, the torus is dominated by the PF-coil system, with the TF-coil geometry shown being determined by: (a) field-ripple constraint ($\Delta B_R/B_0 = 0.003$), (b) TF-coil power consumption (~ 10 MW), and (c) maximized outboard openness for purposes of testing accessibility. The electrically close-coupled PF-coil geometry shown in Fig. 1 was chosen to minimize the capital cost, EF-coil power requirement, and mechanical forces under both operational and fault-mode conditions. Lastly, the choice of four divertor sections equally spaced around the torus and the relatively low mass of even the dominant PF-coil set (22.6-tonne inner OH coils, 154.3-tonne EF coils plus outer OH coils) combine with the other constraints to suggest: (a) vertical installation and maintenance of PF-coil sets and the total torus, and (b) horizontal maintenance of individual torus quadrants, divertor subassemblies, and torus-quadrant subassemblies.

The horizontal (radial, outboard) maintenance and testing scheme used on the more routine operation requires an outboard access gallery that is sufficiently broad to allow removal and transport of an entire (6.6 tonne) torus quadrant, (first wall, blanket shield, and TF-coils) with divertor assembly (0.8 tonne) and quadrant blanket shield maintenance servicing being the more frequent operation. The need to support both normal (~ 187 MN) and off-normal (~ 56 MN, no plasma current) EF-coil forces combine with the desire for an outboard toroidal service gallery to require vertical tension/compression bars to support the outboard EF-coil forces; these bars would be of nominal 0.26-m^2 cross section and are located at the outboard side of each of $N_{TF} = 28$ TF coils. As shown in Fig. 2, $N_{TF} = 28$ EF-coil bulkheads of 70-mm thickness and 1.5-m radial extent assures EF-coil interspace deflections of less than 1 mm under operating conditions; the vertical tension/compression bars would react the EF-coil forces through each of the corresponding bulkheads. This overall arrangement is shown in Fig. 3.

The choice of both long-term and short-term maintenance schemes and the means to react the EF-coil forces both influence strongly the placement of the vacuum boundary. Initial estimates

focused on a vacuum chamber into which the entire torus (including all coils) would be placed and in which all short-term (routine) maintenance/servicing operations would occur. The need to react these operational loads and to make and break coolant lines under vacuum was judged to render less probable the perceived advantages of a large vacuum chamber: eliminating or reducing frequent exposure of the torus to atmospheric pressure and eliminating the complexities related to making vacuum connections directly on the torus. Overly large vacuum chambers were also projected if the major operating forces had to be reacted under vacuum. Consequently, a close-fitting vacuum geometry was selected in which all vacuum connections are made primarily at the toroidal shell forming the outer boundary (i.e., TF-coil strongback) of each test quadrant and within the TF-coil set. As illustrated in Fig. 2, and in more detail in Fig. 3, breakable/reweldable seams between divertor sections and torus quadrants would allow removal/replacement of each without disturbing the other. Each replaceable divertor assembly would be comprised of divertor plates and associated coolant header/manifold, the captive nulling coil, and associated vacuum pump shielding. All TF coils, nulling coils, test-cell quadrants, and first-wall tube banks and associated coolant headering and manifolding together form each of four toroidal quadrants. The cryogenic vacuum pumps, attached to each of four divertor sections would be removed for replacement of either divertor or test quadrants. Similarly, diagnostics and fuel-pellet injectors would have to be disconnected and removed.

2.5.2 Test Arrangement. A given toroidal quadrant test space could be comprised of a) two "clam-shell" hemi-quadrants, b) single testing spaces fabricated between and under each of the $N_{TF} = 28$ TF coils, c) or an arrangement of insertion test rods or "drawers" tailored to support irradiation of multiple specimens or small single-effects tests, some of which could be instrumented. While the latter approach least interferes with the machine operation, only the outboard portions of the test regions surrounding the plasma would be utilized.

Although not available in great detail or for specific tests arrangements at this point in the conceptual FTF RFP design, the preliminary test configuration described above suggests a number of broad testing categories. These categories are summarized below

Plasma-Materials-Interaction (PMI) Tests: These tests would probably be limited to the few divertor sectors, each giving approximately $A_{PMI} = 1 - 1.5 \text{ m}^2$ of area where direct plasma-material interactions could be observed. The overall design suggested above allows for relatively frequent, non interfering horizontal maintenance service of each divertor section. Since the divertor, as well as the first-wall functions directly

impact overall device operation (e.g., availability), it is likely that tests involving these components will be selected to minimize the impact on the overall machine availability and, therefore, on the bulk radiation tests.

Bulk Radiation Tests: The use of the fusion neutrons can be broadly classified into integrated blanket/shield tests performed within a given test quadrant *versus* smaller, less-integrated tests, a number of which could be conducted in a given quadrant. Each of the four quadrants could in principle be used independently to test fully integrated blanket/shield concepts. The available volume between the first wall and TF-coil array amounts to 8.0 m^3 or 0.3 m^3 per TFC, with the between under-coil testing volume split being 1:0.5. Hence, it may be possible to dedicate two-three quadrants to fully integrated blanket shield tests ($I_{av} = 4.3\text{ MW/m}^2$, $B_{\theta} = 7\text{ T}$, $B_{\phi} = 0.5\text{ T}$, $q_s = 1.7\text{ MW/m}^2$), and one-two quadrants to finer scale tests ranging from breeder-coolant tests of a few per quadrant to single specimen materials test of one per TF-coil sector (0.05 m^3 , highly instrumented (interactive) to many per TF coil (high-fluence, multiple specimen, passive). The flexibility in individual TF-coil design (i.e., vertical outboard return leg, breakable conductors, etc.) to accommodate a range of active, subsystem tests should be noted.

3. SUMMARY AND CONCLUSIONS

The physics required to realize the high-fluence (3.4 MW/yr. m^2 per year at 80% availability), high-volume (10 m^3), low-power (124 MW fusion power, 152 MW input power) neutron source will not be available until after the RFPs presently being constructed have operated. The database for this fusion nuclear-testing facility, therefore, would not be available until after the year 1995. Given that high-current RFPs in relatively compact geometry can be demonstrated within this time frame, however, an efficient and cost effective neutron source that is capable of providing a majority of small-sample and integrated test needs for fusion could be available by the year ~ 2000.

REFERENCES

1. H. A. Bodin, R. A. Krakowski, and S. Ortolani, "The Reversed-Field Pinch: from Experiment to Reactor," *Fus. Technology* **10**, 307 (1986).
2. D.B. Thomson (ed.), "Proceedings of the International Workshop on Engineering Design of Next Step Reversed Field Pinch Devices", Los Alamos National Laboratory Report LA-11139-C (July 13-17, 1987).
3. The TITAN Research Group, "The TITAN Reversed-Field Pinch Reactor Study--The Final Report," University of California-Los Angeles, GA Technologies, Inc., Los Alamos National Laboratory, and Rensselaer Polytechnic Institute UCLA-PPG-1200 (to be published 1988).
4. C. G. Bathke, R. A. Krakowski, R. L. Miller, and K. A. Werley, "A DT Fusion Neutron Source Based on the Reversed-Field Pinch," Proceedings of 12th IEEE Symposium on Fusion Engineering, Monterey, CA (October 12-16, 1987).
5. C. G. Bathke, "The Oscillating-Field Current Drive System," Proceedings of 12th IEEE Symposium on Fusion Engineering, Monterey, CA (October 12-16, 1987).
6. K. A. Werley, C. G. Bathke, and R. A. Krakowski, "Edge Plasmas and Plasma Wall Interactions in an Ignition-Class Reversed-Field Pinch," Proceedings of 12th IEEE Symposium on Fusion Engineering, Monterey, CA (October 12-16, 1987).

Figure 1. Elevation view of OH-, EF-, and TF-coil sets for FTF RFP showing the support structure and horizontal access to blanket test regions.

Figure 2. Plan view of FTF RFP torus showing (a) TF coils that meet the $\Delta B_r / B_0 = 0.003$ ripple constraint and, (b) the poloidally symmetric toroidal field divertor, and (c) vacuum boundary and pumping station.

Figure 3. Detailed plan view of divertor and vacuum connections

Figure 4. isometric view of FTF RFP torus and coil geometry

TABLE I. DEVICE PARAMETERS FOR FTF-RFP

Parameter	Value
Plasma major radius, $R_T(m)$	1.8
Plasma minor radius, $r_p(m)$	0.3
Plasma aspect ratio, $A = R_T/r_p$	6
Plasma shape	circular
Plasma volume, $V_P(m^3)$	3.20
First-wall area, $A_{FW}(m^2)$	23.45 ^(a)
Blanket/shield thickness, $\Delta h(m)$ ^(b)	0.30
Blanket/shield volume, $V_{BLK}(m^3)$	10.23 ^(c)
Pinch parameter, $\Theta = B_\theta(r_p)/(B_\phi)$	1.52
Reversal parameter, $F = B_\phi(r_p)/(B_\phi)$	-0.12
Poloidal field at plasma edge, $B_\theta(T)$	6.8
Toroidal field at plasma edge, $B_\phi(T)$	-0.52
Safety factor, $q(r_p) = F \Theta^{-1}$	0.013
Poloidal beta, β_θ	0.10
Average electron temperature, $T_e(kV)$	9.00
Average ion temperature, $T_i(kV)$	8.53
Average electron density, $n_e(10^{20} m^{-3})$	6.87
Effective plasma atomic number, Z_{eff}	1.69
Toroidal plasma current, $I_\phi(MA)$	10.20
Plasma current density, $j_\phi(MA m^{-2})$	36.08
Lawson parameter, $n\tau_E(10^{20} s m^{-3})$	0.78
Anomalous ohmic heating of ions, $f_{OHM}^{(d)}$	0
Ohmic power in plasma, $P_{Oh}(MW)$	24.7
Fraction of alpha-particle energy to plasma, f_α	1.0
Fusion power, $P_f(MW)$	124.2
Plasma Q-value, $Q_p = P_f/P_{Oh}$	5.03
First-wall heat flux, $q_s(MW m^{-2})$	1.72
Diverter peak heat flux, $q_{div}(MW m^{-2})$	3.5
Fusion neutron first-wall loading, $L_n(MW m^{-2})$	4.3
Plasma loop voltage, $V_\phi(V)$	2.42
Streaming parameter, $\xi = v_{pe}/v_{thi}$	0.0058
Poloidal flux, $I_p I_\phi(W b)$	64.14
Transport scaling parameter, $\nu(r_p) \propto I_\phi^2 r_p^2$	1.25

(a) Theoretical value. Divertors reduce first-wall coverage to 0.87.

(b) Assumed for purposes of subsequent magnetics calculation

(c) Theoretical value. Divertors and first-wall coolant headers reduce blanket coverage to 0.78

(d) Actually observed in RFP experiments

TABLE II. PF-COIL PARAMETERS FOR THE FTF, RFP

Parameter	Value	
	OH Coils	EF Coils
Current (MA)	22.57 ^(a) (-34.03) ^(b)	11.46 ^(c)
Volume (m ³)	14.67	18.28 ^(d)
Mass (tonne)	43.42	133.47 ^(d)
Joule losses (MW)	84.81 ^(a) / 193.20 ^(b)	54.26 ^(c)
Peak field (T)	10.85 ^(b)	4.12 ^(c)
Current density (MA/m ²)	8.5-24.1 ^(b)	9.0-11.8 ^(c)
EF-Coil vertical field index	--	0.62
OH-Coil stray vertical field (mT) ^(b)	1.20(1.33)	--
PF-Coil vertical transparency (%)		72.1

(a) Forward-bias values.

(b) Back-bias values.

(c) Mean steady-state values.

(d) Includes EF trim coil.

TABLE III. TF- AND DF-COIL DESIGN PARAMETERS FOR THE FTF REF

Parameter	Value		
	TF Coils	DF Coils	
		Nulling	Flanking
Number of TF coils	28	4	8
Major radius (m)	1.80	1.86	1.85
Minor radius (m)	0.70	0.54	0.53
Radial thickness (mm)	138.1	43.8	62.0
Toroidal thickness (mm)	132.0	43.8	62.0
Current per coil (kA)	167.8	384.0	192.0
Current density (MA/m ²)	9.2	200.	50
Total ohmic power (MW)	8.8 ^(a)	29.9 ^(b)	5.9 ^(a)

(a) Aluminum coils are used to reduce neutron activation

(b) Copper coils are used to minimize ohmic dissipation.

TABLE IV. COLLECTOR PLATE THERMAL-HYDRAULICS PARAMETERS FOR THE FTF/RFP

Coolant pipe inner diameter, $d_{ID}(mm)$	10.0	
Tungsten coating thickness, $\Delta r_w(mm)$	2.0	
Water coolant velocity, $v(m/s)$	2.5	
Water inlet temperature, $T_{H_2O}^m(C)$	100.0	
Degrees below saturation temperature, $\Delta T_{SAT}(C)$	20.0	
Volumetric heating in metal and water, $Q(MW/m^3)$	67.0	
Pipe material	Cu	SS
Pipe thickness, $\Delta r(mm)$	1.0	0.5
Water outlet temperature, $T_{H_2O}^{out}(C)$	110.	109.
Water inlet pressure, $P_{H_2O}^m(MPa)$	0.25	0.24
Water pressure drop, Δp	0.06	0.06
Critical heat flux, $CHF(MW \cdot m^2)$	4.0	4.0
Peak pipe temperature, $T(C)$	280	355
Normalized total stress, σ/σ_y	0.05	0.34

TABLE V. OFCD PARAMETERS FOR THE FTF RFP

Parameter	Value
Average Plasma Current, I_0 (MA)	10.20
Drive Frequency, f (Hz)	60
Toroidal-flux swing, $\delta\phi/\phi_0$	0.035
Θ Variation	1.466 - 1.581
F Variation	-0.043 - -0.196
Toroidal/Poloidal Circuit (MW):	
Plasma Poynting power, P_P^*	2.826 76 114.88
Plasma dissipation, P_Ω	23.28 0.0
First-wall dissipation, P_{FW}	0.00 0.00
Blanket dissipation, P_B	1.56 17.68
Terminal Reactive Power, P_r^* (MW)	
OH coils	3.26
TF coils	944.28
EF coils	67.19
Trim coils	817.60
Coil Dissipation, P_c (MW)	
OH coils	0.00
TF coils	10.64
EF coils	54.63
Trim coils	26.28
Real (lost) Terminal Power, P_r^L (MW)	
OH coils	0.06
TF coils	44.15
EF coils	52.61
Trim coils	37.27
TF-coil dc power, P_{TF}^{SS} (MW)	8.82
EF-coil dc power, P_{EF}^{SS} (MW)	52.61
Power-supply dissipation, P_{PS} (MW) ^(a)	18.32
Total dissipation, P_D (MW)	152.41
Current-drive power, P_{CD} (MW)	90.98
Current-drive efficiency, I_0/P_{CD} (A/W) ^(b)	0.11

(a) Assumes the power supplies are 99% efficient.

(b) This efficiency is based on total power consumed in the system. An equivalent estimate for r/f current drive in tokamaks is ~ 0.06 A/W assuming a conversion efficiency of 0.3.