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ZTI: PRELIMINARY CHARACTERIZATION OF AN IGNITION CLASS REVERSED-FIELD PINCH†

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ABSTRACT

A preliminary cost-optimized conceptual design of an intermediate-step, ignition-class RFP device (ZTI) for the study of alpha-particle physics in a DT plasma is reported. The ZTI design reflects potentially significant cost savings relative to similar ignition-class tokamaks for device parameters that reside on the path to a viable commercial RFP reactor. Reductions in both device costs and number of steps to commercialization portend a significantly reduced development cost for fusion. The methodology and results of coupling realistic physics, engineering, and cost models through a multi-dimensional optimizer are reported for ZTI, which is a device that would follow the 2-4 MA ZTH on a $\sim$ 1996-98 timescale.

1. INTRODUCTION

Two major ($I_p = 1\text{-}2\text{ MA}$) reversed-field-pinch (RFP) experiments (ZTH¹ and RFX²) will become operational in the 1992-3 time frame. A recently completed RFP reactor study, TITAN³, suggests an operationally, economically, and environmentally attractive commercial power plant. Since the cost of developing fusion is related strongly to the number of major device steps needed to reach the commercial end-product, and since this cost is becoming a concern, a cost-optimized conceptual design of an intermediate-step, DT-ignition-class RFP device (ZTI) has been initiated.^{4,5}

The ZTI device would study alpha-particle physics in a reactor-relevant configuration. The design procedure begins with the selection of ignition-class plasma parameters, taken here as $Q_p \equiv P_F/P_H \geq 5$, where P_F is the fusion power and P_H is the externally induced plasma heating power. This heating power for RFPs is provided entirely by ohmic dissipation in the plasma (i.e., $P_H = P_\Omega$), whereas tokamaks require auxiliary heating (i.e., $P_H = P_\Omega + P_{AUX} \sim P_{AUX}$). The ZTI study connects with CPRF/ZTH¹ by using the same plasma size (i.e., $R_T = 2.4\text{ m}$ and $r_p = 0.4\text{ m}$), although plasma size is one of a number of key parameters subject to variation. Conservative extrapolations of experimental energy-confinement times, parabolic ion-density

profiles, and high impurity levels ($Z_{eff} \approx 2.6$) are used to provide an (ohmic) ignition margin. With guidance from a steady-state, point-plasma model⁶, a zero-dimensional, time-dependent plasma/circuit/first-wall simulator⁷ is used to model the entire plasma transient. Typical plasma parameters during a 5-s period of constant fusion-power are toroidal plasma current $I_p \approx 9\text{ MA}$; peak magnetic fields of $B_{\phi c} = 1\text{ T}$ and $B_{\phi c} = 10.9\text{ T}$ at the TF and PF coils, respectively; plasma temperature $T \approx 11\text{ keV}$; plasma density $n_i \approx 3 \times 10^{20}\text{ m}^{-3}$; $P_F \approx 100\text{ MW}$; and $Q_p \approx 5$. This plasma design point was adopted as a "strawman" for use in examining the requirements of ohmic heating to DT ignition and to perform a cost-optimized magnetics design. The required magnetic-field coils and support structures are then characterized by an optimizing algorithm that examines a range of non-interfering PF-coil locations constrained by flux conservation, the circular plasma surface, shell and TF-coil radial-build requirements, stray-vertical-field limits, and diagnostic-access requirements.

The costing basis that is used as an object function for the coil-configuration optimizer is derived from and calibrated against devices for which detailed engineering designs and associated "bottoms-up" cost estimates exist (e.g., ZTH¹, CIT⁸). The PF-coil currents and voltages are obtained from plasma-equilibrium and circuit analyses, respectively. The resulting (peak) reactive power (340 MVA) and stored energy (2.2 GJ) are used to cost the power supplies, which is the major cost item. For a (copper) coil current density of 25 MA/m^2 , the total coil mass is 90 tonne and requires 66 tonne of structure to support a peak force of 106 MN (acting on the top inner-most OH coil, Fig. 1). The ZTI design has a fusion power density of 1 MW/m^3 within the coil envelope (14 MW/m^3 in the plasma), fusion-power-to-mass ratio of 588 kW/tonne , and a maximum average first-wall neutron loading of 2 MW/m^2 . These coil-mass and reactive-power figures-of-merit reflect potentially significant cost savings relative to ignition-class tokamaks for a device that resides on the path to a viable commercial reactor.³ The short-pulsed (9.7 s) ZTI design also includes a toroidal-field divertor, but not current drive. The methodology of coupling realistic physics, engineering, and cost models through a multi-dimensional optimizer is described in Sec. II, the results

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Fig. 1. The structural layout of the cost-optimized ZTI "strawman" design point.

are presented in Sec. III, and the conclusions are given in Sec. IV.

II. MODELS

The models used in the ZTI design study consist of a steady-state, point-plasma model⁶; a zero-dimensional, time-dependent, coupled plasma and circuit model⁷; and a cost-based optimization model of the coils and support structures⁴. The steady-state plasma model estimates the plasma parameters (e.g., plasma current, I_p) at ignition and sensitivities of the ignition parameters to physics assumptions and extrapolations (e.g., energy confinement time, impurity levels, profiles, etc.). The time-dependent model subsequently estimates the plasma parameters of a high Q_p , slightly subignited plasma for the study of alpha-particle physics with minimal fusion power while minimizing burn-control problems normally associated with DT ignition. The optimization model determines the minimum-cost coil set and associated support structures that are required to provide the plasma transient suggested by the time-dependent simulations.

II.A STEADY-STATE MODEL

The steady-state model⁶ specifies independent of size: separate ion and electron energy balances; fixed particle fractions for $H, D, T, {}^3He, {}^4He, C$, and Fe ; and charge neutrality. An assumed magnetic-field profile yields the current-density profile. Profile effects are included through radial averages over assumed density, temperature, and current-density profiles. This model is used parametrically to determine a range of possible operating points in a density-temperature phase space, for fixed parameters like Q_p , beta, plasma current, etc. Selecting a plasma size then gives

the starting point for the time-dependent and cost-optimizing models that are used to generate all plasma, engineering, and cost parameters.

II.B TIME-DEPENDENT MODEL

The time-dependent model couples the external electrical circuit to a profile-averaged, zero-dimensional plasma model. The PF circuit consists of the equilibrium-field (EF) coils, the ohmic-heating (OH) coils, and the plasma. The plasma model includes separate ion and electron energy balances; particle continuity for $H, D, T, {}^3He$, and 4He ; charge neutrality; specified impurity fractions that adjust to input from a first-wall evaporation/sputtering model; and a soft-beta limit that spoils confinement if the plasma pressure exceeds experimentally observed limits. Experimental plasma profiles and transport properties are used to generate self-consistent plasma-equilibrium configurations that in turn are adjusted as the DT-fueled burn proceeds. The concentrations and energies of the fusion products (i.e., $H, {}^3He, {}^4He, T$) are followed in time with a Fokker-Planck slowing-down formulation to account for time lags in fusion-product heating and for fast-ion pressure. Power generated in the plasma impinges on a two-region approximation to the first wall, one-dimensional temperature and stress distributions within the first wall are calculated as a function of time, and an evaporation/sputtering model describes (carbon) impurity influx to the plasma using a heat-flux concentration factor.

II.C COST-OPTIMIZATION MODEL

The cost-optimization model determines the minimum-cost PF-coil set that will inductively initiate, ignite, and sustain a plasma with predetermined (goal) properties (e.g., plasma resistance, R_p ; I_p ; R_T ; and r_p). This model uses a nested double optimization: (a) two inner, serial optimizations that satisfy the physics constraints; and (b) an outer optimization that minimizes device cost subject to a range of engineering constraints. Specifically, the inner optimizations first locate the EF coils by minimizing deviations from a circular plasma equilibrium and subsequently locate the OH coils by minimizing the stray vertical field. Each coil set is constrained to a separate toroidal surface. The EF-coil currents and the magnetic flux provided to the plasma by the EF coils are determined through a parameter f_ψ that is varied by the outer optimization to minimize device cost. The OH-coil currents are derived from a simplified circuit analysis of the EF coils, the OH coils, and the plasma that also yields the coil power and energy requirements for costing purposes. This circuit model uses analytic time trajectories of the plasma current and resistance that are derived from the time-dependent model to determine the coil currents and voltages as well as the transfer resistance required during an initial OH-coil discharge phase.

The outer optimization varies the shape and position of the toroidal surfaces to which the EF and OH coils are separately constrained. The OH coils are constrained to a surface of rectangular cross section to simplify the

support structures, as is shown in Fig. 1, and the EF coils are constrained to a surface of elliptical cross section to ensure equilibrium solutions. The engineering constraints imposed during the outer optimization eliminate overlapping structures. These structures consist of the PF-coils, the support structures, the TF-coils, and diagnostic systems. All coils are sized using a nominal (peak) coil current density $j_c = 25 \text{ MA/m}^2$. The OH-coil forces are reacted by a simple combination of a toroidal cylinder inboard of the torus, horizontal girders located above and below the torus, and vertical posts placed outboard of the torus. The EF-coil forces are reacted by toroidal cylinders positioned inboard and/or outboard of the torus and by the previously mentioned horizontal girders placed above and below the torus. The thicknesses of these support structures are established by hoop-stress, buckling, compression, or tension constraints⁹, using a nominal maximum design stress $\sigma_{MAX} = 200 \text{ MPa}$. A maximum coil deflection⁹ of 1 mm determines the number of girders. The TF-coil positions are scaled from ZTH,¹ with an additional $\Delta r = 0.22 \text{ m}$ being added to the radial build to accommodate a toroidal-field divertor that occupies approximately a (minor) radial space equal to one plasma minor radius.^{3,10} Space is reserved for diagnostics in an outboard access way (Fig. 1) that is of a nominal height $h = r_p$ and centered about the equatorial plane.

The genesis of the costing model⁴ used in the optimization is the database and procedure developed as part of the 1985 TFCX design effort¹¹. Adjustments have been made to align these costing accounts and algorithms with the more recent CIT⁸ and ZTH¹² designs. The costing procedures and bases generated as part of recent designs of near-term tokamaks, as well as farther-term commercial fusion reactors and fission reactors, have also been monitored, but the results from near-term fusion devices have received primary emphasis in developing the costing model used herein. All costs are reported in 1988 dollars (~10% increase takes these costs to 1990) and include hardware, engineering, installation, contingency, and project (management) costs. The costing model has 21 major accounts and tabulates a total ("green field") cost that includes project management costs following accepted practices.^{8,12} The cost model reproduces the ZTH¹² and an early (1986) CIT⁸ cost projections; comparisons are made with this earlier CIT design.

The optimization algorithms used¹³ are based on variants of the method of steepest descent. Side-stepping techniques have been incorporated to circumvent the consequences of the barrier constraints imposed to eliminate structural overlap.

III. RESULTS

The scaling of the energy confinement time is critical to the design of an ignition device. The following scaling suggested by Taylor and Conner¹⁴ and based on resistive

interchange modes is used here:

$$n_e \tau_E = C_{ir} I_\phi^3 N_e^{-1/2}, \quad (1)$$

where n_e (10^{20} m^{-3}) is the electron density, N_e (10^{20} m^{-1}) is the electron line density, τ_E (s) is the energy-confinement time, and I_ϕ (MA) is the plasma current. Fitting Eq. (1) to the RFP database¹⁵ yields a scaling coefficient $C_{ir} = 0.0033$ as shown in Fig. 2. Although ZTH requires a large extrapolation in energy-confinement time from the present database, ZTI is a small extrapolation from ZTH along the path to the reactor.

Fig. 2. Plot of Taylor-Connor¹⁴ scaling expressed in the form of $n_e \tau_E$. Also shown are a least-squares fit to the present experimental database;¹⁵ the 95% confidence levels for the least-squares (two-parameter) fit; and the projected ZTH,¹ ZTI, and TITAN reactor³ operating points.

III.A Ignition Current

Using Eq. (1), the steady-state model⁵ predicts DT ignition at a minimum current of $I_\phi^* = 8.2 \text{ MA}$ for parabolic density and temperature profiles and impurity fractions of 0.02 carbon and 0.002 iron ($Z_{eff} = 2.6$); these impurity concentrations characterize operation with limiters rather than divertor operation. The sensitivity of the ignition current to changes in C_{ir} is shown in Fig. 3 for the above-cited profile and impurity assumptions. To provide additional safety margin, the design plasma current is conservatively taken as the maximum for which sufficient space exists for the PF coils and support structure for ZTH-like dimensions. Consequently, a design current of 9 MA was adopted. Operation with a cleaner plasma (0.01 carbon, 0.0005 iron, $Z_{eff} = 1.6$) requires only $I_\phi^* = 6.9 \text{ MA}$ to ignite, so the 9-MA design point would provide a 30% safety margin in current (a 100% safety margin in $n_e \tau_E$) for divertor operation.

Fig. 3. Sensitivity of the energy-confinement-time scaling coefficient, C_{τ} , to the minimum ignition current, I_{ϕ}^* .

III.B PLASMA PARAMETERS

An ignited DT plasma poses two potential problems for the experimentalist: (a) a relatively high fusion power complicates machine design (increased activation, higher heat fluxes, additional heat-rejection equipment, etc.); and (b) thermal instability at ignition complicates machine control. These problems can be reduced by designing a slightly subignited machine that retains reactor-relevant alpha-particle physics (i.e., $Q_p \geq 5$). The time-dependent model, therefore, was then used to determine the plasma density corresponding to $Q_p \geq 5$ in a 9-MA plasma. The density was controlled by a refueling algorithm that held I_{ϕ}/N_e constant (one of a number of options). The results of the time-dependent simulation are shown in Fig. 4, and the associated plasma parameters are given in Table I. Variations in the current risetime have no impact on the steady-state plasma properties. The time between attaining the flat-top current and achieving a steady-state plasma (~ 2.1 s) is determined by the alpha-particle thermalization time (~ 0.36 s) rather than energy-confinement time ($\tau_E = 0.27$ s). The subignited plasma described in Table I can be driven to ignition (i.e., $Q_p \rightarrow \infty$ and thermal runaway without a beta limit) by either raising the current to $I_{\phi} = 9.5$ MA, lowering the impurity fractions to $f_C = 0.016$ and $f_{Fe} = 0.0016$ ($Z_{eff} = 2.4$), or raising the density-control parameter to $I_{\phi}/N_e = 6.0 \times 10^{-14}$ Am at constant I_{ϕ} , [i.e., n_e decreases with $n\tau_E$ increasing according to Eq. (1)]. Operation with plasma currents above the $I_{\phi} = 9.0$ -MA strawman design would require modest (4-5%) increases in major radius for the same stress and coil current density.

III.C ENGINEERING PARAMETERS

The optimization model requires a discharge length to

Fig. 4. The time trajectories for key plasma parameters.

TABLE I. ZTI and CIT⁸ Physics Parameters

Parameter	ZTI	CIT
Plasma major radius, $R_T(m)$	2.40	1.22
Plasma minor radius, $r_p(m)$	0.38	0.45
Elongation, κ	1.0	1.8
Toroidal plasma current, $I_{\phi}(MA)$	9.0	9-10
Refueling parameter, $I_{\phi}/N_e (10^{-14} Am)$	5.6	1.6
Pinch parameter, $\Theta = B_{\theta}(r_p)/(B_{\phi})$	1.54	0.26
Reversal parameter, $F = B_{\phi}(r_p)/(B_{\theta})$	-0.2	~ 1.0
Poloidal magnetic field, $B_{\theta}(r_p)(T)$	4.74	2.75
Toroidal magnetic field, $B_{\phi}(r_p)(T)$	-0.61	10.4
Poloidal beta, β_{θ}	0.14	~ 0.7
Toroidal beta, β_{ϕ}	0.06	0.05
Electron temperature, $T_e(keV)$	10.8	10-12
Electron density, $n_e(10^{20} m^{-3})$	3.5	~ 5
Ion density, $n_i(10^{20} m^{-3})$	3.0	~ 5
Effective plasma atomic number, Z_{eff}	2.59	≤ 1.5
Fusion power, $P_F(MW)$	98.4	300
Ohmic dissipation in plasma, $P_{\Omega}(MW)$	20.4	~ 0
Auxiliary heating, $P_{AUX}(MW)$	0	20
Average first-wall heat flux, $q_s(MW/m^2)$	1.03	2.27
First-wall neutron loading, $I_w(MW/m^2)$	2.02	6.8
Plasma loop voltage, $V_{\phi}(V)$	2.26	0.03
Streaming parameter, $\xi = v_{De}/v_{th}(10^{-3})$	5.63	1.66

determine the flux swing provided by the PF coils. The current rise time is 2.6 s, assuming a maximum plasma current ramp rate of $\dot{I}_{\phi} = 20$ MA/s during an OH-coil discharge phase and 2 MA/s when the OH-coil current is driven negative. After attaining the steady-state current flat-top, a 2.1-s period is required to achieve a thermal steady state. Specifying a 5-s

period for studying alpha-particle physics, a total discharge length of 9.66 s results. The cost-optimized coil configuration for this basecase is shown in Fig. 1, and the associated engineering parameters are given in Table II. The PF-coil configuration suggested in Fig. 1 fits closely to the TF coils in order to maximize the inductive coupling to the plasma. A cost breakdown for the 268 M\$(1988) ZTI costs by major accounts is shown in Fig. 5. The power supplies provide 2.2 GJ of stored energy to the (~ 300 K) copper coils with a peak dissipated power of 282 MW representing the dominant cost. A cost breakdown for an early (1986) CIT design⁸ using the same costing model is also shown in Fig. 5. Although that CIT design is comparable in plasma temperature and current, the costs are higher by a factor of 1.9 primarily because of the high toroidal field and associated power supplies required to drive the TF coils; physics and engineering parameters for CIT (1986)⁸ are also shown in Tables I and II.

Fig. 5. A cost breakdown of the ZTI costs in 1988 dollars. Also shown is the cost breakdown of an early (1986) CIT design.⁸

An important issue for the $\tau_D \approx 9.7$ s ZTI is impurity control: whether the use of an armour-plated first wall is adequate or if toroidal-field divertors are needed. The time-dependent model was used to examine the sensitivity of a graphite-armoured ZTI to peaking of the first-wall heat flux. The peaking model assumes that the spatially averaged heat flux, q_s , is increased by a factor f_{pk} over a fraction f_{pk}^{-1} of the first-wall area. The effect of varying f_{pk} on the first-wall temperature, T_{FW} , and the effective plasma atomic number, Z_{eff} , is shown in Fig. 6. The carbon cladding becomes sufficiently hot for $f_{pk} = 30$ to overwhelm the plasma with evaporated carbon and to force early plasma termination at 9.5 s in the discharge; higher values of f_{pk} lead

to earlier discharge termination. The first-wall temperature rises because the model assumes constant I_ϕ and I_ϕ/N_e that increases the ohmic heating and the electron temperature and decreases the DT density and temperature with increasing Z_{eff} . If a more experimentally relevant constraint of constant toroidal loop voltage were enforced, then the plasma current and wall temperature would collapse with the large influx of carbon. Under either constraint, the conclusion remains the same; the plasma will prematurely terminate for $f_{pk} \geq 30$ because of carbon evaporation. The extent to which vapor shielding of the localized hot spot or local redeposition ameliorates this problem remains to be determined. In any case, flux peaking factors in excess of ~ 30 necessitate the use of magnetic-divertor impurity control.

Fig. 6. The effect of heat-flux peaking on time trajectories of T_{FW} and Z_{eff} .

The cost sensitivity of the ZTI design to various design parameters is shown in Fig. 7. The device cost is most sensitive to decreases in the equilibrium-flux parameter, f_ϕ , which is one of seven parameters varied by the outer optimization to minimize device cost. The second most sensitive parameter is the plasma current, I_ϕ , which is projected to cost ~ 10 M\$/MA; in the RFP, ignition safety margin is purchased with additional current. A more relevant parameter with which to study ignition margins is C_{ir} of Eq. (1). In studying the economic impact of decreasing C_{ir} , the reference design size was increased to $R_T = 3$ m and $r_p = 0.5$ m in order to ensure space for the OH coils at $j_c = 25$ MA/m² for the higher- I_ϕ devices; Z_{eff} was also decreased from 2.6 to 1.6 to take credit for the divertor for which space (and cost) has been included in all cases reported. For this adjusted basecase, decreasing C_{ir} by factors of two and four leads to current increases

of 32% and 74%, respectively, in the new basecase ignition current (I_ϕ decreased from 8.2 MA to 6.9 MA when Z_{eff} was decreased); the respective cost increases are 7.4% and 21%. The cost variations caused by changes in C_{tr} do not simply reflect the costs of current increases as reported on Fig. 7. Decreasing C_{tr} also increases the plasma density because the optimal I_ϕ/N_e associated with the minimum ignition current I_ϕ^* is approximately constant over a wide range of C_{tr} values; the resulting decrease in the equilibration time and discharge length makes the increase in power-supply cost less than predicted by a simple current increase.

Fig. 7. Sensitivity of ZTI cost to various parameters.

The importance of operating at the optimal value of I_ϕ/N_e associated with I_ϕ^* is suggested in Fig. 7. The cost of providing the additional $\Delta r = 0.22$ m of radial build required for a toroidal-field divertor is 23.4 M\$. One approach to dealing with a tight configuration is to increase the coil current density, j_c , and/or the maximum design stress of the support structure, σ_{max} . The device cost is relatively insensitive to σ_{max} because the optimal PF-coil configuration for satisfying the physics constraints is primarily determined by j_c . Cost sensitivity to j_c is large, however, because over half of the energy provided by the power supplies is dissipated in the PF coils. Providing additional diagnostic access, h , is not a major cost factor, at least for access on the outboard side.

The position of ZTI along the development path to a commercial fusion reactor is illustrated by Table III and Fig. 2. The ZTI device is an intermediate step between the ZTH¹ device and the TITAN³ reactor, as measured in terms of I_ϕ , β , $n_e \tau_E$, and n_e . The ZTI plasma is reactor grade by virtue of the plasma temperature. The ZTI device addresses the reactor issue of compactness (economics, operability, etc.) through the parameters j_ϕ and p_F (Table III) and

TABLE II. ZTI and CIT⁸ Engineering Parameters

Parameter	ZTI	CIT
Peak OH-coil current, I_O (MA)	20.2	18.5
Peak EF-coil current, I_E (MA)	6.1	21.1
Peak coil magnetic field, B_c (T)	10.9 ^(a)	21.7 ^(b)
Peak coil force, F_c (MN)	106 ^(a)	588 ^(b)
Total coil mass, M_c (tonne)	90	350
Peak coil Joule losses, P_c (MW)	282	>123
Total stored energy, W_{PS} (GJ)	2.2	3.9
Poloidal magnetic energy, W_{B_θ} (MJ)	636	801
Peak react. power, P_c^* (MVA)	340	>860
Structure mass, M_s (tonne)	66	^(a)
Fusion-power-core volume, V_{FPC} (m ³)	110	98
Power density, (MW/m ³):		
• Fusion-power-core, p_{FPC}	1	3
• Plasma, p_F	14	61.5
Fusion-power-core mass, M_{FPC} (tonne)	167	380
Fusion-power-to-mass ratio, P_F/M_{FPC} (kW/tonne)	588	1,070
Discharge length, τ_D (s)	9.7	6.4

(a) PF-coil value.

(b) TF-coil value.

(c) Assumed part of TF-coil mass.

addresses the associated engineering issues related to q_r and I_w . While the technologies required of ZTI are generally applicable to the reactor, requirements related to fully steady-state operation (e.g., heat removal, current drive, etc.) would not be addressed. The progression of parameters suggested on Table III, however, heuristically indicates the possibility of a reduction in the number (and size) of major device steps towards commercialization, along with an attendant reduction in the overall cost and time for fusion development.

4. SUMMARY AND CONCLUSIONS

Conceptual design studies of an RFP ignition-class ($Q_p = 5$) device, ZTI, indicate the possibility of obtaining reactor-relevant burning plasmas in a ZTH-sized device ($R_T = 2.4$ m and $r_p = 0.38$ m) with low fusion power ($P_F \approx 100$ MW) while relying only on ohmic plasma heating in a $I_\phi = 7$ -9 MA plasma. The major cost driver is plasma current (10 M\$/MA), which over a wide range of physics safety margin amounts only to a few tens of millions of dollars (i.e., ~ 10 M\$ per unit uncertainty factor in $n\tau_E$). Studying ignition may be relatively inexpensive (268 M\$) compared to a tokamak because of the inherent characteristics of the poloidal-field-dominated RFP, wherein a much larger portion of the confining magnetic field is produced by the plasma rather than by expensive external coils. Furthermore, ZTI would provide burn information for a plasma in a configuration and using a technology that lie directly on a trajectory to an economic and operationally attractive commercial reactor.³

TABLE III. Comparison of Steps to RFP Commercial Reactor

Parameter	ZTH ¹	ZTI	TITAN ³
Plasma major radius, $R_T(m)$	2.4	2.4	3.9
Plasma minor radius, $r_p(m)$	0.4	0.4	0.6
Toroidal plasma current, $I_p(MA)$	2-4	9	18
Plasma current density, $j_p(MA/m^2)$	4.4-8.8	19.7	15.9
Plasma beta, β	0.05	0.06	0.11
Plasma temperature, $T(keV)$	2-4	11	10
Plasma density, $n_e(10^{20}/m^{-3})$	1.25-2.5	3.5	9.3
Plasma Q-value, Q_p	—	4.8	80.7
Lawson parameter, $n_e\tau_E(10^{20}/m^{-3}s)$	0.025-0.125	0.95	1.92
First-wall heat flux, $q_s(MW/m^2)$	0.7	1	4.6
First-wall neutron loading, $I_w(MW/m^2)$	—	2	18
Plasma power density, $p_F(MW/m^3)$	—	14	83
Blanket	no	no	yes
Divertor	no	yes	yes
Current Drive	no	no	yes
Coil Technology	normal	normal	normal/superconducting

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