SCOPING STUDIES FOR SMALL STEADY-STATE TOKAMAKS FOR DIVERTOR TESTING*

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Abstract

A prime uncertainty in next-generation devices is the divertor performance. For the International Thermonuclear Experimental Reactor (ITER), the divertor limit often plays a more critical role in the operational scenario definition than do beta limit and energy confinement constraints. Hence, it is desirable to test the divertors in an environment as close as possible to that expected in next-step burning plasma experiments. Initial global scoping studies are done for small, steady-state, copper coil, beam-driven tokamaks that are dedicated to divertor testing. The usual ITER global physics models (beta limit, energy confinement, and analytic divertor heat load calculation) are incorporated, and for performance criteria we require that the divertor heat load and plasma collisionality in the edge region be similar to those expected in ITER. The smallest, lowest-cost devices satisfying these constraints tend to have major radius below 1 m, plasma current of 0.5 to 1 MA, and low aspect ratio and costs of a few tens of millions of dollars. Injection powers of about 4 to 5 MW are needed to sustain the plasma current, maintain plasma power balance, and provide the required divertor heat load.

I. Objective

Divertor operation is expected to be a primary obstacle in the operation of a near-ignited tokamak.¹ To further compound the problem, at this time predictions for the divertor conditions of these devices are very uncertain. A test bed for divertor operation, under conditions as close as possible to those expected in high-Q tokamak operation, would be highly desirable. To this end, we search for small, low-cost tokamaks, that would be dedicated solely to the purpose of divertor testing. The only mission requirements for this device are to provide divertor conditions as close as possible to those expected in ITER. To find the "bare-bones" minimum cost, we take credit for existing site facilities located at Oak Ridge National Laboratory (these are mostly power supplies and injection power equipment), although similar results could result for other sites. We formulate the problem in the following section and identify minimum cost solutions in Section III. Minimum cost devices in the aspect ratio range of 1.5 to 3 have major radii ranging from roughly 0.5 to 1.0 m and costs in the tens of millions of dollars.

II. Method

The present analysis uses a simple systems code. In particular, the cost algorithms and magnet models are kept simple for expediency, but they should be adequate for this initial scoping task. The constrained optimization procedure incorporated in other systems codes (TETRA²) are used here. The analysis is done with a version of the STAR³ code. Generally, the global ITER modeling rules⁴ are used, with the exception of plasma current, poloidal fields, and plasma volume calculations, which are modified to apply at low aspect ratio ($A < 2$).

II.1 Assumptions

Many of the configurational/operational scenario assumptions are described in Table 1. These are generally fixed input values. An important assumption used here is the absence of an inductive solenoid. Since the device is to be steady-state, noninductive current drive means will be needed.

II.2 Constraints

We imposed a set of constraints that are described in Table 2. The primary "performance" constraints are that the divertor heat load must be between 5 and 10 MW/m² and that the scope-off plasma be collisional. The plasma power balance, beta limit, and density limit use the scalings imposed for ITER studies. The magnets are constrained by current density limits (see also Tables 3 and 4).

II.3 Free parameters

Table 3 lists the variables (and their bounds) which were allowed to vary to satisfy the constraints. These are R , n_e , T_e , the confinement H-factor, q , B_t , P_{heat} . The aspect ratio, while not listed in Table 3, is varied parametrically.

II.4 Solution Method

In the solution process, we optimize for minimum cost. The costing method is outlined in Table 4 using the unit costs based on the ATF experience.⁵ Note that all costs here are in 1987 dollars (it is estimated that the cost escalation factor at ORNL to 1991 dollars is about 1.75). Also note that credits are taken in the injection power cost category. The solution method is to find the minimum cost device that uses the assumptions listed in Table 1, satisfies the constraints in Table 2, and allows the quantities listed in Table 3 to vary. More description of the modeling used here is given in the following section.

II.5 Description of Models

Divertor Model. The analytic Harrison/Kukushkin model developed in the ITER studies⁴ is used here. A key assumption used in this application is $n_{\text{separatrix}}/n_e = 0.333$, which is similar to that used in the ITER studies. Higher density ratios would alleviate the divertor collisionality and heat load limits. We scale the divertor connection lengths with the edge q and R , appropriately fitted to match the ITER results. Also, the poloidal distance from the x-point to the strike point is taken to be the minimum (0.5 m , $2 \times r_{\text{plasma}}$) as compared to 1.5 m for ITER. We choose a relatively long distance (compared to the plasma size) here to allow ample room in the divertor chamber to accommodate divertor variations and diagnostics. The heat load is limited to a range [$5 \leq H_{\text{div}}(\text{MW/m}^2) \leq 10$] which is in the neighborhood of that expected in ITER before applying safety margins. We choose a range rather than a single value for H_{div} so that this constraint is not too restrictive on the device parameters. It is expected that this device will have some control on the heat load by varying the flux expansion near the strike point [by use of poloidal field (PF) coils close to the divertor].

Physics. The physics modeling generally follows that outlined in the ITER physics document.⁴ The notable exceptions are in the plasma current scaling with field and plasma size/shape—for which we use the methods developed by M. Peng,⁶ which is valid at low and high aspect ratios. These methods from Ref. 6 are also used to find the poloidal field and plasma volume. An important modeling difference is for radiation—we assume 30% of the plasma heating power is radiated, for lack of a better low-temperature line radiation/neutral charge-exchange power loss model. We assume an H plasma here, which affects some beam properties. Also, we have imposed a 60% upper bound to the bootstrap fraction, as many of the cases arrived at have very high poloidal betas, for which the ITER scalings predict bootstrap fractions $> 60\%$.

Current Drive. We note that no inductive capability constraint is used here. There is no central solenoid included in the radial build, and all the current (except for the bootstrap fraction) is assumed to be driven noninductively. Plasma startup is also assumed to be noninductive. In this sense, the cases presented with higher aspect ratio would be less affected if some inductive startup (or hybrid inductive/noninductive long pulse) requirement were imposed. In this case, a thin-wall vacuum vessel or a vacuum vessel with insulated breaks should be used. We assume neutral beam current drive and use the ITER model for current drive efficiency, not accounting for the possible acceleration of beam ions by the ICRF waves. The required beam energy for penetration is also calculated, assuming two e-fold decays to the plasma center.

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Poloidal Field Coils. The PF coils are assumed to be internal. At this point, no inductive current drive requirements are used, so no OH central solenoid is employed. The current density is assumed to be 10^7 A/m², and the currents in the coils are taken from fits of the data in Ref. 6:

$$I_{\text{shaping coil}} = 0.3 \times I_{\text{plasma}} \times A^{1.6}$$

$$I_{\text{vertical field coil}} = 0.4 \times I_{\text{plasma}}$$

(These fits are only good for $A < 4$). The data in Ref. 6 are for currents that produce a single equilibrium. Since flexibility of plasma shape, strike point flexibility, etc., is likely to be needed in this experiment, these scalings may underestimate the PF coil needs. On the other hand, since the PF coils are internal here (as opposed to external in Ref. 6), this should alleviate the PF coil requirements.

III. Results

We examined divertor testing tokamaks ranging from conventional aspect ratio ($A = 3$) down to Spherical Tokamak type aspect ratios (A near 1.5). Figure 1 shows minimum-cost device parameters over this range. The results can be summarized as follows: (a) there is a continuous drop in device cost as the aspect ratio goes from 3 down to 1.6; and (b) the cost increases relatively steeply as A is increased from 1.6 to 2.0. The drop in cost as A goes from 3 to 2 is due to the decrease in device size possible with lower aspect ratios (in this range the minimum minor radius of 30 cm is met). Below $A = 2$, the plasma current begins to increase (because of the confinement and divertor collisionality constraints), which tends to reduce cost benefits of going to lower aspect ratios. Finally, below $A = 1.6$ there is not enough room for the inner TF coil leg, and further decreases in A cause a rapid increase in device size and cost.

Table 5 lists device parameters for $A = 1.6$, $A = 2.0$, and $A = 3.0$ cases. The low- A cases are confinement limited ($H = 2$) but have some beta margin, whereas the higher-aspect-ratio cases are beta limited but have some confinement margin assuming ITER power law scaling. The H factors required for plasma power balance for these cases show a wide spread, depending on the scaling used (we used the ITER Power scaling for these studies). Note that for ITER Offset Linear scaling, all examples in Table 5 would satisfy plasma power balance with "L-mode" confinement (i.e., $H \leq 1$). If lower densities are used for any of these cases, more confinement margin is obtained, at the sacrifice of the scrape-off layer SOL plasma collisionality. Also, these cases have margin in the edge q , implying that higher plasma current could be possible if (a) the current drive efficiency were improved; (b) additional current drive power were available; (c) some inductive current drive "boost" were employed; and/or (d) lower SOL connection lengths were permitted. Higher current could provide additional confinement or beta margin, but lower edge q would reduce the SOL plasma collisionality. Also, the beam energy needed to penetrate to the plasma center for optimal current drive is < 50 keV for the lower- A cases, which allows utilization of existing beam equipment for current drive (1.5-MW credit taken here).

Because of the lower q -values, the higher- A cases are more strongly affected by the divertor collisionality constraint than the low- A cases. Relaxing this constraint would allow lower toroidal field levels, and reduce the cost. Also, for the high- A cases the injection power is governed by the need to maintain a sufficient divertor heat load, rather than for plasma power balance. At low A , the situation is reversed; the injection power requirements are dictated by the need to keep the H factor ≤ 2 , and the resulting heat load is above the minimum 5-MW/m² level.

The total coil power requirements (for all cases in Table 5) range from ~20 to 50 MW as A increases from below 2 to 3. These power levels should be possible with upgrades to existing power supplies. The lower-aspect-ratio example has lower cost, primarily due to the lower magnet and injection power costs. Both of these lower costs are primarily due to the smaller device size and R . Note that we use a multiturn inner TF coil leg—use of a single-turn "centerpost" concept in other applications may permit optimum points at lower A .

IV. Summary

Small divertor testing tokamaks to be used as a vehicle for providing a test bed for ITER-like divertor conditions have been investigated at a preliminary systems code level. In particular, we look at steady-state (no OH central solenoid) copper coil cases, which result in small ($0.5 < R(m) < 1$), strongly heated tokamaks that appear to have costs in the tens of millions of dollars (after site credits are taken). This appears to be an

attractive route to producing high divertor heat loads in a tokamak configuration—and with a facility dedicated to divertor testing.

References

- [1] D. Baldwin et al., "U.S. National Review of the ITER Conceptual Design Activity," Institute for Fusion Studies, University of Texas, March 1991.
- [2] TETRA systems code; see R. L. Reid et al., *ETR/ITER Systems Code*, ORNL/FEDC-87/7, 1988.
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- [4] N. Uckan and the ITER Physics Group, "ITER Physics Guidelines," ITER-TN-PH-0-5, Nov. 1990.
- [5] B. Nelson, private communication, 1991.
- [6] Y.-K. M. Peng and D. J. Strickler, "Features of Spherical Torus Plasmas," *Nucl. Fusion* 26, 769 (1986).

Fig. 1. Aspect ratio of a small steady-state divertor-technology tokamak vs (a) relative minimum cost, (b) major radius, (c) total TF coil resistive power and that from the inner leg, and (d) total injection power (NBI + ICRF) and that portion used only for heating (ICRF).

Table 1. Configurational assumptions used in the device selection

	<u>Comment</u>
Internal PF coils	No OH solenoid used
Gap (first wall / inner TFC leg) = 1 cm	For coolant passages
First wall thickness - midplane = 6 mm	ATF value
Scrape-off thickness - midplane = 3 cm	Same fraction of r_{plasma} as ITER
Gap (First wall/outer TFC leg) ≥ 25 cm	Allow room for vertical field coils
$\kappa_x = 2.05 (1 + 0.44 \epsilon^2)$	Fits ITER, higher elongation at lower A
12 coils	Provides port size needed for beams
All copper coils	See a below
OFHC copper for outer TF legs and PFC	Use low J and low unit costs ^b
Glidcop for inner TF legs if necessary	May have higher unit costs ^b

^aThe need for rapid changeout of divertor test modules/components necessitates the need for demountable TF coils, which rules out the use of superconductors.

^bSee Tables 3 and 4.

Table 2. Description of the global constraints used in this analysis

<u>Constraint</u>	<u>Comment</u>
<i>Overall consistency constraints</i>	
$I_p = I_p(A, R, q_{95}, \kappa, \delta)$	Relates plasma current to field and geometry
$B_o R = B_{\text{max}} R_{\text{tf}}$	Relates the field at the coil to field on axis
Σ inboard radial thicknesses = R	Inboard radial buildup = major radius
<i>Plasma constraints</i>	
Plasma power balance ^a	See below
$\beta \leq \text{limit}$	Troyon scaling with $g = 3 \text{ m-T/MA}$
$n_e \leq \text{disruption density limit}$	Using the Borass (ITER) model with a coefficient corresponding to H-mode confinement
edge $q \geq 3 (1 + 2.6 \epsilon^{2.8})$	Limit is ≈ 3 for $A > 2\text{m}$ higher at lower A
$5 < H_{\text{div}} (\text{MW/m}^2) < 10$	Divertor heat load ^c
$\lambda_{\text{ic}} < \text{divertor connection length}^b$	Divertor plasma is collisional
<i>Configuration constraints</i>	
Plasma minor radius $\geq 30 \text{ cm}^c$	See below
Coil flattop powers $< 60 \text{ MW}^d$	Rough limit for available power supplies
Port size (0.3 m) for tangential beam	Positions outer legs
injection outboard field ripple $\leq 1.0\%$	

^aThe terms included are injection power, radiation (set to be 30% of injection power), and transport power. We let the confinement H-factor vary (subject to $H \leq 2$) for the ITER power-law and offset-linear scaling expressions.

^bSee Section II.5.

^cThis constraint plays an important role in the device size/cost. It is important to ensure that the plasma transport is not dominated by atomic effects and to provide an adequate target for neutral beam injection.

^dThe line power available at ORNL is over 100 MVA, which is greater than would be needed here. The 60-MW limit is indicative of the power supplies available (from ATF magnets) that may be modified for steady-state use here.

^eSee the divertor model description in the Appendix and Section II.

Table 3. Quantities allowed to vary in the device selection process

<u>Quantity</u>	<u>Limit</u>	<u>Comments</u>
<i>Physics</i>		
n_e	$\geq 3 \times 10^{19} \text{ m}^{-3}$ (never met)	Prevent neutral dominated plasma
T_e	$\geq 1 \text{ keV}$	Ensure edge temperature $> 100 \text{ eV}$
H factor (confinement)	≤ 2	Minimum of ITER power and offset-linear scalings
R	-----	Usually determined by r_{plasma} constraint
edge q	≥ 3 (never met)	There is an additional q constraint
B_t	-----	Vacuum toroidal field on axis
P_{heat}	-----	Injection power, no contribution to current drive (see costing)
<i>Engineering</i>		
TFC inner radius	$> 5 \text{ cm}$	Allow adequate room for insulated turns
TFC thickness	-----	Bounded by limits in J_{TF}
J_{TF} , inner leg	$< 70 \text{ MA/m}^2$	Unit cost scales with J
J_{TF} , outer leg	$1 < J (\text{MA/m}^2) < 10$	Determined in cost minimization ^a
Gap (outer TFC-first wall)	$> 0.25 \text{ m}$	Set by ripple limit or port size constraint

^aThe upper bound of 10 MA/m^2 is used to be consistent with the lower unit cost of the outer TF coil legs.

Table 4. Cost method used in the device selection process. Costs are in 1987 \$. It is estimated that today's costs (at ORNL) include an escalation factor of about 1.75

<u>Category</u>	<u>Comment</u>
<u>TF Coils</u>	
Inner coils = 250 \$/kg when $J \leq 30 \text{ MA/m}^2$ 250 \$/kg $\times [J(\text{MA/m}^2) + 10]/40$, when $J > 30 \text{ MA/m}^2$	ATF HF coils = 250 \$/kg
Outer = 150 \$/kg	
<u>PF coils</u>	
150 \$/kg	ATF PF coils = 150 \$/kg
<u>Vacuum vessel</u>	
$10^5 \text{ $/m}^2$	May be too high, from ITER scalings
<u>Support structure^a</u>	
190 \$/kg	ATF = 190 \$/kg
<u>Injection power</u>	
Current drive = $3 \text{ $/W}_{\text{plasma}}$	First 1.5 MW is free (credit for existing equipment)
Heating only power = $3 \text{ $/W}_{\text{plasma}}$	First 2 MW is free (credit for existing equipment)
<u>Cooling system</u>	
$1.4 \times 10^4 \text{ $/MW}$	From ATF. This may increase if the cooling tower capability is exceeded
<u>Divertor costs^b</u>	
300 \$/kg	Not included in ATF, but important here
<u>Fixed costs^c</u>	
2.2 M\$	From ATF

^aSupport mass is taken to be $0.34 \times (\text{TFC} + \text{PFC mass})$, which is also taken from ATF.

^bDivertor costed as two (top/bottom) 25-cm-thick disks extending from the outer plasma edge to the inner edge.

^cAccounts for assembly equipment, design integration, site preparation, and electrical and instrumentation systems.

Table 5. Parameters for minimum cost A = 1.60, A=2.0 and A=3.0 cases.

	A=1.60	A=2.0	A=3.0
<u>Physics parameters</u>			
Major radius (m)	0.48	0.60	0.90
Minor radius (m)	0.30*	0.30*	0.30*
Elongation (x-point)	2.40	2.26	2.14
Plasma current (MA)	0.62	0.55	0.56
Edge q	18.0	12.0	6.5
Field on axis (at coil) (T)	1.4 (5.1)	1.7 (4.1)	2.2 (3.5)
$\langle T_e \rangle$ (keV)	1.0*	1.1	1.0*
n_e (10^{20} m^{-3})	0.69	0.70	0.81
$\beta / \beta_{\text{limit}}$ (%)	4.0 / 4.4	3.2/3.2*	2.6/2.6*
g-Troyon (T-m/MA)	2.8	3.0*	3.0*
H factor, ITER Power / Offset - Linear	2.0*/0.90	2.0*/0.95	1.4/0.75
<u>Divertor parameters</u>			
Heat load (MW/m^2)	8.2	5.0*	5.0*
λ_{ie} / connection length	1.0*	1.0*	1.0*
<u>Injection power</u>			
Current drive power (MW)	3.1	3.3	7.2
Heating power (a)	1.2	2.0	2.0
Bootstrap fraction	0.60*	0.60*	0.55
Beam energy (keV)	40	50	90
Drive efficiency (A/W-m^2)	0.026	0.028	0.025

*At a limit.

^aThis power only heats the plasma and does not contribute toward current drive (i.e., ICRH power). Note that 2 MW of this power is assumed to be available "free."

Table 5 (continued). Device parameters for the A=1.6, A=2.0 and A=3.0 cases

	A=1.60	A=2.0	A=3.0
<u>Inboard TF coil legs</u>			
Current density (MA/m^2)	70*	43	42
Mass (1000 kg)	0.9	2.1	3.8
Resistive power (MW)	23	20	35
<u>outboard TF legs</u>			
Current density (MA/m^2)	10*	10*	10*
Mass (1000 kg)	12	18	35
Resistive power (MW)	4.3	6.4	12.2
<u>Cost breakdown (1987 M\$)</u>			
Inner TF coil legs	0.4	0.7	1.2
Outer TF coil legs	1.8	2.8	5.2
PF coils	0.6	0.7	1.4
Vacuum vessel	0.3	0.4	0.6
Current drive	4.8	5.3	17.4
Heating power ^a	0	0	0
Support structure	1.1	1.6	3.1
Cooling	0.4	0.4	0.7
Divertor	1.9	2.4	3.6
Fixed	2.2	2.2	2.2
Total	13.6	16.5	35.4

*Cost credit is taken for 2 MW.

^aThis is the power used only for heating and does not contribute toward current drive.