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ECONOMIC CONSIDERATIONS OF COMMERCIAL TOKAMAK OPTIONS

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ABSTRACT

Systems studies have been performed to assess commercial tokamak options. Superconducting, as well as normal, magnet coils in either first or second stability regimes have been considered. A spherical torus (ST), as well as an elongated tokamak (ET), is included in the study. The cost of electricity (COE) is selected as the figure of merit, and beta and first-wall neutron wall loads are selected to represent the physics and technology characteristics of various options. The results indicate that an economical optimum for tokamaks is predicted to require a beta of around 10%, as predicted to be achieved in the second stability regime, and a wall load of about 5 MW/m^2 , which is assumed to be optimum technologically. This tokamak is expected to be competitive with fission plants if efficient, noninductive current drive is developed. However, if this regime cannot be attained, all other tokamaks operating in the first stability regime, including spherical torus and elongated tokamak and assuming a limiting wall load of 5 MW/m^2 , will compete with one another with a COE of about 50 mill/kWh. This is 40% higher than the COE for the optimum reactor in the second stability regime with fast-wave current drive. The above conclusions pertain to a 1200-MW(e) net electric power plant. A comparison was also made between ST, ET, and superconducting magnets in the second stability regime with fast-wave current drive at 600 MW(e). The result indicates that the COE of an ET or ST device operating with the same technology requirement (wall load) as a superconducting magnet tokamak is about the same as the COE of a superconducting magnet tokamak in the second stability regime. Therefore, a device like ST or ET appears to look more attractive in lower ratings than 1200 MW(e).

I. INTRODUCTION

Scoping studies have recently been performed [1] for commercial tokamaks in the second stability regime with superconducting magnets. An economical optimum range was predicted to have beta between 10 and 20% and neutron wall loading between 6 and 8 MW/m². In this paper, the study is extended to cover other tokamaks that either operate in the first stability regime or utilize normal magnets. Table I shows the options considered for this study. Five options were considered for the first stability regime, three utilizing normal magnets and two with superconducting magnets. The normal magnet options are spherical torus (ST) [2] with F- θ pumping for current drive, elongated tokamak (ET) [3], and standard tokamak with inductive current drive. The two superconducting options are one with normal q at the edge ($q = 2.4$) and the other with $q = 1.1$. Tokamaks have operated without disruption with q as low as 1.3 [4]. Lower q operation may increase beta, which results in higher power density reactors [5]. Both of the reactors will be driven by fast-wave current drive with minimum power requirements [6]. Four options were studied

TABLE I
Commercial Tokamak Options Considered

First Stability Regime	Second Stability Regime
Normal magnet	Normal magnet (inductive current drive)
Spherical torus (oscillating field current drive/F- θ pumping)	
Elongated tokamak (inductive current drive)	
Conventional tokamak (inductive current drive)	
Superconducting magnet (fast-wave current drive)	Superconducting magnet
$q = 2.4$	Inductive current drive
$q = 1.1$	Lower hybrid current drive
	Fast-wave current drive

for the second stability regime: three utilizing superconducting magnets and one using normal magnets with inductive current drive. The differences among the superconducting options are the current drive techniques. They are lower hybrid, slow wave used in STARFIRE [7]; fast-wave current drive with the minimum power requirement [6]; and inductive current drive.

The scoping study was focused on comparing 1200-MW(e) net electric power plants. The smaller reactors with lower ratings [200–600 MW(e)] are discussed in another paper dealing with multiplexing [8]. However, a comparison between ET, ST, and superconducting magnets operating in the second stability regime with fast-wave current drive for 600 MW(e) net was made in this paper to see if there is a merit for devices like ET or ST at lower ratings.

Finally, it is important to note that different options might have different engineering parameters for the optimum design. For example, neutron energy multiplication might be different for the optimum design of an option. There has not been any attempt in this paper to design an optimum reactor for each option. Therefore, some of the engineering parameters have been assumed to be the same for all the options.

II. METHOD OF ANALYSIS

A system code was developed at the Fusion Engineering Design Center (FEDC) for scoping studies of commercial tokamak reactor options. The model used in this code is generic [9] in that the first wall, blanket, and shield conform to the elliptical plasma shape with minimal gaps. The toroidal field (TF) coils are located tightly around the shield, and a nominal poloidal field (PF) coil set is input. The code does not perform any geometrical checks for access clearances or torus-penetration requirements. The ideal tokamak is coupled with realistic costing algorithms. Key features of the code are simplicity of operation, rapid analysis of many cases, and the flexibility to compare a wide range of options. Major options include the selection of copper or superconducting TF coils, scaling laws (neoclassical, neo-Alcator, Mirnov, and H-mode), current-drive mechanisms, and the standard- or

multiplex-plant arrangements. The ohmic heating (OH) coil could be placed either outside or inside the TF coil.

The plasma parameters are calculated using the standard zero-dimensional equations, with the inputs being power (fusion or net electric), average plasma temperature, maximum magnetic field, plasma elongation, safety factor at the edge, impurity levels, plasma temperature and density profiles, and two of the following parameters: average neutron wall loading, beta, aspect ratio, or field on axis. The code can also evaluate a reactor with beta equal to critical beta with different Troyon coefficients. The plasma current is approximated by [2]:

$$I = 2.5(1 + \kappa^2) \frac{aB}{q} \epsilon \frac{1.22 - 0.68\epsilon}{(1 - \epsilon^2)^2} \text{ MA} , \quad (1)$$

where

- κ = elongation;
- a = minor radius, in m;
- B = field on axis, in tesla;
- q = safety factor at edge;
- ϵ = inverse of aspect ratio.

Although this equation was derived for aspect ratios up to about 4, it is a fair approximation for higher-aspect-ratio devices [10]. The power required to drive the plasma current can be calculated by the lower hybrid, slow-wave model used in STARFIRE [7] or by the fast-wave model. Using the best theoretical efficiency in the fast-wave model results in the ideal current drive, with minimum power requirement P estimated [6] to be:

$$P = \frac{n_e R I}{1.4 \times 10^{20}} \text{ MW} , \quad (2)$$

where

n_e = electron density, in m^{-3} ;

R = major radius, in m;

I = plasma current, in MA.

The electrical efficiency of the current drive is assumed to be 70% for all cases [6].

The ignition margin is calculated by the definition:

$$M = \frac{f_\alpha P_\alpha + P}{P_L + P_R} \quad (3)$$

where

f_α = fraction of α particles trapped,

P_α = fusion α power,

P_L = energy transport losses,

P_R = total radiation losses.

The confinement, depicted by the energy confinement time τ_E , is calculated by

$$\tau_E = \frac{W_p}{P_L} \quad (4)$$

where W_p is the plasma energy. Because the applications of the empirical scaling laws for the cases considered here are uncertain, four options are included in the code. These are the neoclassical, neo-Alcator, H-mode, and Mirnov scalings for electrons, and neoclassical scaling for the ions.

The maximum line-averaged plasma density attainable for stable plasma operation is given by the Murakami Limit,

$$n_e = \nu_{Mu} B \kappa^{1/2} R^{-1} q^{-1} \quad (5)$$

The value of ν_{Mn} is in the range of 1 for ohmically heated experiments but could be as high as 3 for systems with auxiliary heating. A value of 3 was assumed for the system study.

The first wall and blanket are modeled based on a design using HT-9 steel and LiPb breeder and coolant. Although this design is not one of the primary choices of the blanket comparison and selection study [11] for tokamaks, it was chosen as a representative blanket for the cost model. The shield material is assumed to be water-cooled Fe-1422. The outboard blanket and shield (no outboard shield for copper magnet option) are assumed to be 0.6 and 0.4 m thick, respectively. (The breeder is 90% enriched ^6Li .) The inboard shield (no inboard blanket is assumed) is sized to limit the nuclear heating in the TF coil, the neutron damage to the TF coil insulator, or the time period between winding-pack annealing. The neutron energy multiplication is assumed to be 1.15, and the thermal-to-electric efficiency is assumed to be 40%. It should be noted that the COE will increase around 5% for the same net electric output if this efficiency is brought down to 35%. (The exact value of the COE change depends on recirculating power.)

The current density in the superconducting TF coils is approximated by the following equation, which is consistent with values in Ref. 9:

$$J = (6.5 - 0.25B_m)10^7 \text{ A/m}^2, \quad (6)$$

where B_m is the maximum field in the winding in tesla. The magnet case thickness and the winding-pack copper fraction are specified as input and are typically assumed to be 0.07 m and 0.80, respectively. The code calculates the stresses in the TF and OH coils and in the bucking cylinder. The PF coil system is assumed to be composed of four superconducting coils, two located close to the inboard side of the plasma and the other two located close to the outboard side of the TF coil. The current in each PF coil is assumed to be equal to the plasma current, as suggested by magnetohydrodynamic (MHD) calculations of plasma equilibria, except for ST and ET, which are described in Sec. III.

The plant power balance accounts for the current-drive power required, the cryogenics power required due to the nuclear heating in the TF coil, and the auxiliary power, which is taken to be 6% of the gross electric power.

The direct cost of the reactor and associated equipment is calculated based on algorithms developed at the FEDC or scaled from STARFIRE values. The direct cost of the balance-of-plant equipment is based on the Energy Economic Data Base [12] and STARFIRE and is shown in Tables II and III. Some of the algorithms in Table II could have been based upon thermal power rather than gross electric power; however, since the thermal efficiencies of power plants are within a narrow range, all the algorithms were

TABLE II
Cost Algorithms for Direct Cost

Cost account	Millions of dollars
Land and land rights	5
Site improvement and facilities	20
Reactor building	$180 (P_G/1200)^{0.5a}$
Turbine building	$\begin{cases} 32 (P_G/1200)^{0.5b} \\ 48 (P_G/1200)^{0.5} \end{cases}$
Reactor maintenance building	$70 (P_G/1200)^{0.5}$
Tritium building	$10 (P_G/1200)^{0.5}$
Heat transport	$70 (P_G/1200)^{0.6}$
Fuel handling	$40 (P_G/1200)^{0.6}$
Maintenance equipment	$40 (P_G/1200)^{0.6}$
Instrumentation and controls	23
Turbine plant equipment	$\begin{cases} 250 (P_G/1200)^{0.8} \\ 500 (P_G/2400)^{0.8} \end{cases}$
Heat rejection system	$30 (P_G/1200)^{0.8}$
Electric plant equipment	$100 (P_G/1200)^{0.8}$
Miscellaneous plant equipment	$35 (P_G/1200)^{0.3}$
Electrical equipment building	$13 (P_G/1200)^{0.5}$
Miscellaneous buildings	$25 (P_G/1200)^{0.5}$

^a P_G is the total gross electrical power of the plant.

^bIt is assumed that there will be two turbines if P_G is greater than 1200 MW(e) but only one turbine building. The cost of the turbine building housing two turbines is assumed to be 1.5 times the cost of the building housing one turbine.

TABLE III
Plant Unit Cost Factors for Various
Reactor Components (1985 dollars)

First wall, \$/kg	70
Blanket, \$/kg	70
Reflector, \$/kg	16
Stainless steel shield, \$/kg	20
Magnet conductor, ^a \$/kA-m-T	2.5
TF magnet winding, ^b \$/kg	80
PF and OH magnet winding, ^c \$/kg	175
Magnet case, ^d \$/kg	40
Copper magnet, ^e \$/kg	85
ICRF power, \$/W	1.1
Power supply, \$/W	0.1

^aNbTi conductor.

^bPool-boiling-type winding.

^cForced-flow-type winding.

^dMinimally machined welded stainless steel case.

^eSingle-in-hand, single-pancake winding of internally cooled copper conductor.

based on gross electric power. Indirect costs are assumed to be 35% of direct costs. This indirect cost is less than the short-term projection for fission reactors but is higher than that projected for coal-fired plants [13]. Interest during construction is calculated based on an inflation-free effective cost of money of 4.2%. The levelized-revenue-requirements method is used to calculate the COE with a fixed charge rate of 8.3% [13,14]. These cost algorithms were developed specifically for this study. A more general cost data base is in preparation as an extension to the FEDC cost accounting system [14].

The overall plant availability is taken to be 75% up to a neutron wall loading of about 13 MW/m². This is based on the assumptions that half a torus can be replaced during a scheduled annual plant maintenance period of 30 days, that the unscheduled maintenance period is 61 days, and that the wall life is 20 MW·year/m². The assumption of 30 days for replacement of half the torus is consistent with STARFIRE [7] and INTOR [15]. If

the wall loading divided by the wall life exceeds $2/3 \text{ year}^{-1}$, then the availability is reduced to accommodate more frequent torus replacement by

$$\text{Availability} = 0.83 / [(1 + 0.164) \text{wall load (MW/m}^2\text{)} / \text{wall life (MW-year/m}^2\text{)}] .$$

III. TOKAMAK DESCRIPTIONS

In this section, nonconventional tokamaks are briefly described. They are ST, ET, and a tokamak with q at the edge equal to 1.1.

III.A Spherical Torus (ST)

The spherical torus (2) is a very small aspect ratio ($A < 2$) confinement concept obtained by retaining only the indispensable components inboard to the plasma torus. MHD equilibrium calculations show that spherical torus plasmas with a safety factor $q > 1$ are characterized by high toroidal beta ($\beta_t > 0.2$), low poloidal beta ($\beta_p < 0.3$), naturally large elongation ($\kappa \geq 2$), large plasma current, strong paramagnetism, and strong plasma helicity. These features combine to permit a compact reactor at low field. Because of its strong paramagnetism and helicity, the spherical torus plasma shares some of the desirable features of spheromak and reversed-field pinch (RFP), but with tokamak-like confinement and safety factors. Although current drive via ac helicity injection was first introduced for the RFP [16], its application to tokamak plasmas was also suggested recently [17,18]. Because there is no data base for spherical torus, there are considerable uncertainties about the effects of strong paramagnetism on achievable plasma beta, plasma energy, confinement at low aspect ratio and high current, and oscillating-field current-drive approaches.

The concept of a spherical torus was selected for study as a reactor. Due to its large natural elongation, an elongation of 3 was not considered excessive and is consistent with

the MHD calculation. The beta is assumed to be equal to the critical beta with the following equation [19]:

$$\beta_c = 0.035 \frac{I}{aB} \quad (7)$$

It is assumed that $F\theta$ pumping is used to drive the current and that it does not require an appreciable amount of recirculating power. (It is important to note that this type of current drive requires large amounts of reactive power.) It is further assumed that there is no requirement in the form of an inductive mechanism for startup. Therefore, the reactor will not have an OH coil.

The use of a superconducting magnet will result in a relatively large size due to geometrical constraints (low aspect ratio). In order to minimize the inboard shield thickness for the copper coil design, it is possible to use a demountable coil so that the inner portion of the TF coil could be replaced annually. The reference design for the scoping study is a TF coil that lasts the plant lifetime, through increasing the inboard shield thickness. The results of the trade-off study show that the COE will not increase significantly due to the latter design. The bucking cylinder has been eliminated in this design, and the inboard radial forces will be taken by wedging the TF coils in the inboard side. There are four superconducting EF coils, with the total current (I_{EF}) equal to:

$$I_{EF} = [0.3 + (0.8/\epsilon - 0.9)]I \quad (8)$$

This is consistent with the MHD calculation performed for the plasma [2].

III.B Elongated Tokamak (ET)

The ET concept is based on taking full advantage of plasma elongation and shaping to improve the plasma confinement [3]. The ET concept achieves the large elongation by

inserting a straight section into the middle of a circular plasma, using straight, solenoidal PF coils. The total PF coil current is approximately equal to the plasma current. It is possible to reach ignition on ohmic heating alone, provided ET is long enough [3]. Evaluation of the plasma energy balance indicates that $\kappa = 6$ to 10 is adequate for ignition in fairly small devices, depending on the plasma temperature profile during startup.

Although ET relies on "conventional" tokamak physics, it is clearly extrapolating well beyond the region of any experimental basis. A number of critical issues must be resolved if the ET concept is to be pursued. For example, the most important question is, "Can a highly elongated, high-beta plasma configuration be created, sustained, and controlled for tokamak time scales?"

The ET concept was selected for study as a reactor. An elongation of 9 was assumed for the reactor. The beta is assumed to be equal to the critical beta, which is given in Eq. (7).

It is assumed that the reactor does not require any auxiliary heating. The OH coil is inside the TF coil for better coupling and is used for both startup and burn.

A normal magnet is utilized for TF coils. Our reference point design for the scoping study is a TF coil that lasts the plant lifetime rather than demountable coils that require annual replacement of the inner part. The bucking cylinder has been eliminated, and the inboard radial forces will be taken by wedging the TF coils in the inboard side. The total PF current is assumed to be equal to the plasma current.

III.C Tokamak with $q = 1.1$

One possible way to increase average beta is to operate the tokamak at low q . Low q implies a relatively low toroidal-field requirement and high plasma current, which appears beneficial for transport and stability. Tokamaks have operated without disruption with q as low as 1.3 [4]. However, no attempt was made in these machines to raise beta; thus the physics of low- q , high-beta tokamak reactors must be considered uncertain.

Recent experimental tokamak data have shown the following equation to be valid for maximum beta in the first stability regime [20]:

$$\beta_c = \frac{0.27}{q^{1.1}} \left(\frac{a}{R} \right)^{1.3} \kappa^{1.2} (1 + 1.5\delta) , \quad (9)$$

which includes q and plasma shaping (δ) explicitly. The favorable benefit of low q is apparent from the equation, but there are no experimental or theoretical data to support this formula in the low- q regime.

The concept of a low- q tokamak was selected for study as a reactor. The q was assumed to be equal to 1.1, a significant extrapolation toward lower q , to evaluate the effect of low- q tokamak operation. It is assumed that the power required to drive the plasma current is given by the high-speed drive model [Eq. (2)]. The input parameters for this reactor are discussed in Sec. IV.

IV. RESULTS

The systems code results were compared with separate existing codes developed elsewhere specifically for ST [21], ET [22], and conventional normal magnet tokamaks [23]. The agreement between our systems code and others is fairly good, in view of the generality of the systems code. The design point close to optimum (minimum COE) for each reactor was found and selected for final comparison of the various reactors. Steady-state operation was assumed for all the reactors except the ones inductively driven, in which long pulses (approximately 0.5 h) were considered. No optimization was performed as a function of the pulse length. The input parameters for each reactor are discussed in Sec. IV.A, followed by the output parameters in Sec. IV.B.

IV.A Input Parameters

The input parameters for the nine reactors are shown in Tables IV and V for superconducting and normal magnets, respectively. The input parameters were selected in such a way as to attain a reactor close to the optimum point. An aspect ratio of 6 and a moderate neutron wall loading of 5 MW/m^2 were chosen for all superconducting magnet cases operating in the second stability regime with moderate $B_{\text{max}} = 8 \text{ T}$. A moderate wall loading of $3\text{--}5 \text{ MW/m}^2$ is desirable to sustain loss-of-coolant/flow accidents without material damage (assuming low-activation material is not utilized here) that would otherwise result from afterheat [24]. A magnetic field on axis of 4 T for superconducting magnets was chosen for the first stability regime (Table II).

The input parameters for normal magnet cases (Table V) are close to those chosen by other investigators [21–23] to obtain a reactor close to the optimum point. For the normal

TABLE IV
Power Plant Input Parameters (Superconducting Magnet)

	First Stability		Second Stability		
	$q = 1.1$	$q = 2.4$	Ohmic Drive	Lower Hybrid	Fast Wave
T, keV	20 ^a	20	15	20	20
B_{max} , T	8 ^b	8	8	8	8
Elongation	2.2 ^c	2.2	1.6	1.6	1.6
Safety factor	1.1	2.4	2.4	2.4	2.4
Aspect ratio	d		6	6	6
Wall load, MW/m^2			5	5	5
B_0 , T	4	4			
OH coil current density	1	1	1	1	1

^aTo reduce current drive power requirement.

^bTo reduce magnet stresses.

^cTo increase β_{crit} .

^dOutput variables.

TABLE V
Power Plant Input Parameters (Normal Magnet)

	First Stability			Second Stability
	ST	ET	Conventional Tokamak	Conventional Tokamak
T , keV	15	10	10	10
Elongation	3	9	2.2	1.6
Safety factor	2.7	2.4	2.4	2.4
B_0 , T	3	4	5	<i>a</i>
Aspect ratio				6
Beta				0.15
Paramagnetism	1.57	1	1	1
TF coil current density, (A/m ²) $\times 10^7$	2	1	1	1
OH coil current density, (A/m ²) $\times 10^7$	NA ^b	1	1.0	0.25
Burn time, h	NA	0.5	0.5	0.5
Bore radius, m	0.2	0.6	0.6	3.7

^aOutput variables.

^bNA = not applicable.

magnet tokamak operating in the second stability regime, the value of beta cannot be greater than 15% if an ignition margin of one based on Mirnov scaling is required. A beta value of 15% was chosen to minimize COE. The value of paramagnetism for ST was derived from the equilibrium calculation.

IV.B Output Parameters

The output parameters for the nine reactors, each producing 1200-MW(e) net, are shown in Tables VI and VII for superconducting and normal magnets, respectively. The plasma parameters resulting from the systems code are very close to those obtained by other investigators for ST [21], ET [22], and normal magnet tokamaks operating in the first stability regime [23]. The major radius of the ET devices is 3.12 m vs a reported 2.93

TABLE VI
Power Plant Output Parameters [Superconducting Magnet, $P_{\text{net}} = 1200 \text{ MW(e)}$]

	First Stability Regime		Second Stability Regime		
	$q = 1.1$	$q = 2.4$	Ohmic Drive	Lower Hybrid	Fast Wave
Major radius, m	5.35	7.2	6.9	7.7	7.04
Beta, %	15.7	8.3	10	10	10.9
Wall load, MW/m ²	4.4	2.0	<i>a</i>		
Plasma current, MA	27.5	25	4.5	5	4.6
Field on axis, T			5.3	5.5	5.4
Available volt-seconds	180	580	1260	1900	1360
Plasma density, 10 ²⁰ /m ³	1.6	0.85	2.4	1.9	2.0
Murakami limit, 10 ²⁰ /m ³	3.0	1.2	1.2	1.1	1.2
Recirculating power fraction, %	22.4	18	7	29	11.8
Engineering Q	4.4	5.6	14.4	3.5	8.5
Mass power density, kW/tonne	220	125	190	220	260
Unit direct cost, \$1000/kW	1.6	1.7	1.4	1.7	1.24
COE, mill/kWh (constant year dollars)	45.1	48.0	40.5	48.1	36.4

^aInput variables.

m [22]. This difference is due to large volt-second requirements assumptions in our study for 1800-s burn time vs the 1100-s reported time [22]. As Table VI indicates, the beta for all superconducting magnet cases operating in the second stability regime is about 10–11%, with a plasma current in the range of 4–5 MA. On the other hand, the plasma current in the first stability regime is in the range of 16 MA for ET and 27 MA for ST, which may be more difficult for ST current drive. The beta (35–37%) and wall load (8–10 MW/m²) are both high for ST and ET for minimum COE, as indicated in Table VII. As is discussed in Sec. V, they can, however, be reduced, with a resulting increase of COE. In some

TABLE VII
Power Plant Output Parameters [Normal Magnet, $P_{net} = 1200$ MW(e)]

	First Stability			Second Stability
	ST	ET	Conventional Tokamak	Conventional Tokamak
Major radius, m	2.76	3.12	4.9	7.7
Beta, %	35	37	8.8	<i>a</i>
Aspect ratio	2.1	8.9	2.9	
Wall load, MW/m ²	8.1	9.0	5.6	5.1
Plasma current, MA	27	15.7	22.8	3.8
Plasma density, 10 ²⁰ /m ³	2.7	7.6	2.8	3.2
Murakami limit, 10 ²⁰ /m ³	2.1	4.8	1.9	0.85
Recirculating power fraction	0.26	0.22	0.39	0.27
Engineering <i>Q</i>	3.8	4.5	2.6	3.7
Mass power density, kW/tonne	560	425	170	140
Unit direct cost, \$1000/kW(e)	1.2	1.2	2.0	1.7
COE, mill/kWh (constant year dollars)	35.5	35.8	50.6	48.5

^aInput variables.

of the reactors, such as the second stability regime and copper coil design, the plasma density exceeds the Murakami limit. Future experimental developments would be beneficial for clarification.

Table VIII presents the output parameters for ST, ET, and a superconducting magnet tokamak with fast-current drive, each producing 600 MW(e) of net electric power. The input parameters for this table are the same as those shown in Tables IV and V except that the field on axis in ST has been increased to 3.5 T from 3 T to minimize the COE, and the maximum field in the superconducting magnet tokamak has been increased to 10 T from 8 T to keep beta around 10%.

TABLE VIII
Power Plant Output Parameters [$P_{net} = 600$ MW(e)]

	First Stability, Normal Magnet		Second Stability, Superconducting Magnet
	ST	ET	Fast wave
Major radius, m	2.4	2.84	5.04
Beta, %	28	34	10
Wall load, MW/m ²	7.0	6.5	5 ^a
Aspect ratio	2.38	9.7	6 ^a
Plasma current, MA	19.0	12.2	3.7
Plasma density, 10 ²⁰ /m ³	2.9	7	2.4
Recirculating power fraction	0.35	0.29	0.14
Engineering Q	2.9	3.4	7.2
Mass power density, kW/tonne	370	275	210
Unit direct cost, \$1000/kW(e)	1.7	1.7	1.6
COE, mill/kWh (constant year dollars)	50	50.5	49.2

^aInput variables.

V. DISCUSSION OF RESULTS AND CONCLUSIONS

Figure 1 presents the COE for various concepts with the corresponding wall loads. The COE for ST and ET falls within the COE of a superconducting magnet tokamak operating in the second stability regime with fast-wave current drive, but with the higher wall load. Changing the current drive of the reference superconducting case to ohmic drive will increase COE by 10%, and switching to lower hybrid current drive will increase the COE by 30%. The COE of a superconducting magnet tokamak operating in the first stability regime with $q = 2.4$ and fast-wave current drive is 30% higher than that of the reference tokamak operating in the second stability regime. Reducing q to 1.1 will improve COE by 6%, with respect to the $q = 2.4$ case. The COE of a copper magnet tokamak operating in the first and second stability regimes is 40 and 33% higher than that of the

Fig. 1. Comparison of the COE of various tokamak options ($P_{net} = 1200$ MW).

reference superconducting case with fast-wave current drive at the same wall load, respectively. However, if the wall load for the normal-magnet tokamak operating in the second stability regime increases from 5.1 to 9.0 MW/m², the COE drops 15% from 48.5 to 41.4 mill/kWh(e). The COE of a reactor with STARFIRE parameters is also shown in Fig. 1. It should be noted that the reported value of beta for STARFIRE (beta = 6.7%) falls in the second stability regime [the Troyon limit for STARFIRE is about 3%, according to Eq. (7)]. The STARFIRE plasma heating power requirement [90 MW(e)] is closer to that predicted by the fast-wave model than to the lower hybrid, slow-wave model.

Figure 2 shows the beta of each concept vs the corresponding wall load. Beta and wall load are assumed to qualitatively represent the physics and technology requirements of the reactors, respectively. Note that the COEs for ST and ET and the superconducting magnet tokamak operating in the second stability regime with fast-wave current drive are comparable and most attractive. However, the physics and technology requirements of these reactors are judged to be significantly higher than the first stability, ohmic-drive reactor, but it is possible to reduce these requirements. This results in a higher COE and is discussed below. The reactor with the lowest requirement corresponds to a superconducting magnet operating in the first stability regime with fast-wave current drives with $q = 2.4$.

Table IX shows the effect of reducing beta and the wall load on COE for ET and ST via reducing field on axis only. In both reactors, COE increases as wall load or beta decreases. For example, the COE of an ET or ST device, operating with the same physics and technology requirements as a superconducting magnet tokamak in the second stability regime with a fast-wave current drive, is about 50% higher. It is noted that COEs given in this table for lower wall loads and betas are not based on optimum design points; therefore, this table only represents a trend in the COE change. The values for COE for ET in Table IX are for constant availability volt-seconds. Since the plasma current increases with the decrease of the Troyon coefficient, the burn time is not constant and decreases as the Troyon coefficient decreases.

The results of parametric studies on 600-MW(e) power plants (Table VIII) indicate that the COE of ST and ET is comparable to that of a superconducting magnet tokamak

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Fig. 2. Comparison of the beta of various tokamak options ($P_{net} = 1200$ MW).

TABLE IX
The Effect of Beta and Wall-Load Variations on COE for ET and ST

ET				ST			
Wall load (MW/m ²)	Beta (%)	β/β_{crit}	COE (mill/kWh)	Wall load (MW/m ²)	Beta (%)	β/β_{crit}	COE (mill/kWh)
9.0	37	1.0	35.8	8.1	35	1.0	35.5
5.0	57	1.0	38.2	5.0	53	1.0	36.8
5.0	47	0.8	39.1	5.0	41	0.8	38.3
5.0	36	0.6	40.5	5.0	30	0.6	40.7
5.0	25	0.4	43.0	5.0	19	0.4	45.6
5.0	13	0.2	49.4	5.0	9	0.2	60.0

in the second stability regime with a fast-wave current drive. Since wall loads in the three concepts are approximately the same, it could be concluded that the COE of an ET or ST device operating with the same technology (not physics) requirements as a superconducting magnet tokamak is about the same as the COE of a superconducting magnet tokamak in the second stability regime. Therefore, devices such as ST or ET appear to look more attractive in power ratings less than 1200 MW(e) net.

In conclusion, an economic optimum for tokamaks with 1200-MW(e) net electric power is predicted to require a beta of around 10%, as predicted to be achieved in the second stability regime, and a wall load of about 5 MW/m², which is assumed to be optimum technologically. This tokamak is expected to be competitive with fission plants if an efficient, noninductive current drive is developed. However, if this regime cannot be attained, all other tokamaks operating in the first stability regime with a wall load around 5 MW/m², including ST and ET, will compete with one another with a COE of about 50 mill/kWh(e). This is 40% higher than the optimum reactor in the second stability regime with fast-wave current drive. At lower ratings, ST and ET look more attractive.

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