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POTENTIAL MINIMUM COST OF ELECTRICITY OF SUPERCONDUCTING COIL TOKAMAK POWER REACTORS*

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ABSTRACT

The potential minimum cost of electricity (COE) for superconducting tokamak power reactors is estimated by increasing the physics (confinement, beta limit, bootstrap current fraction) and technology [neutral beam energy, toroidal field (TF) coil allowable stresses, divertor heat flux, superconducting coil critical field, critical temperature, and quench temperature rise] constraints far beyond those assumed for ITER until the point of diminishing returns is reached. A version of the TETRA systems code, calibrated with the ITER design and modified for power reactors, is used for this analysis, limiting this study to reactors with the same basic device configuration and costing algorithms as ITER. A minimum COE is reduced from >200 to about 80 mill/kWh when the allowable design constraints are raised to 2 times those of ITER. At 4 times the ITER allowables, a minimum COE of about 60 mill/kWh is obtained. The corresponding tokamak has a major radius of approximately 4 m, a plasma current close to 10 MA, an aspect ratio of 4, a confinement H-factor ≤ 3, a beta limit of approximately 2 times the first stability regime, a divertor heat flux of about 20 MW/m^2, a $B_{max} \leq 18$ T, and a TF coil average current density about 3 times that of ITER. The design constraints that bound the minimum COE are the allowable stresses in the TF coil, the neutral beam energy, and the 99% bootstrap current (essentially free current drive).

INTRODUCTION

This study is motivated by the following reasons:

1. Recent studies of commercial reactor concepts assume varied advances in physics and technology, which lead to a wide range of COE estimates. To assess the importance of various scientific and technical advances in reducing COE, COE minimization is systematically approached[1] via constrained variation of major design variables.
2. Recent studies of ITER-like reactors[2] and advanced-technology future reactors [i.e., ARIES-1 (Ref. 3)] have shown interesting correlations between COE and design assumptions. This study aims to clarify the needed advances in physics and technology beyond ITER to attain the ultimate potential minimum COE for the superconducting tokamak reactors, for which ITER aspires to be the first experimental reactor.

This study of tokamak power reactors, then, (1) determines the minimum COE and design parameters for reactors consistent with present ITER physics, technology, and engineering constraints and (2) determines the limiting COE as the constraints are systematically relaxed up to a point of diminishing returns. In each part of the study, the physics and engineering parameters consistent with the minimum COE are found subject to the imposed constraints of divertor heat flux, maximum TF, beta limit, confinement time, beam energy, magnet stress, superconducting magnet critical field and temperature, etc. Critical trade-offs in key design issues, such as aspect ratio, maximum TF, and confinement scaling, are also clarified for reactors.

This paper addresses (1) modifications to the TETRA code[1] in order to model power reactors, (2) benchmarking the technology phase of the ITER design with the modified code (called TETRA-R), (3) benchmarking two previous power reactor studies (GENEROMAK,[4] TPSS[5]) with the TETRA-R code, and (4) determining the minimum COE as a function of the degree of relaxation of the ITER limits and constraints. The paper ends with conclusions and discussions.

CODE MODIFICATIONS

The TETRA code was written specifically to model an experimental power reactor. The thermal power was dumped to the atmosphere by way of a cooling tower, and no provision was made in the code for specifying the use of a turbine generator. In order to complete this study, TETRA was modified to include models of power blankets; turbine generators; an economics package to compute COE; adjustments of unit costs assuming a mature reactor economy; and treating replaceable tokamak components (e.g. blanket and divertor) as recurring costs similar to fuel costs in a fission reactor.[4] In addition, the code was upgraded with a recent fixed boundary MHD equilibria package so as to more accurately model the poloidal field (PF) system[6] and to provide for consistency between the value of beta used

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in the power balance and the value of beta used in the plasma equilibrium determination.

Blanket

The blanket model supplies a consistent set of input parameters based on blanket materials and temperatures desired. These parameters are taken from previous detailed analysis of candidate blanket configurations such as those presented in Ref. 7. These include inboard thickness, outboard thickness, average density consistent with volume fractions of materials used, average unit cost, energy multiplication factor, and thermal conversion efficiency.

Capital costs

The capital cost computations in the TETRA code are modified for power reactor application. The elements of the tokamak are costed based on ITER unit cost and modified for learning curve effects, and the costs of the balance of plant (BOP) and heat transport systems are scaled functions of thermal or electrical power as suggested in Ref. 4 and updated according to Ref. 8. Building costs are modified to include the cost of a turbine building. The costs of the first wall, blanket, divertor, and 25% of the current drive system are treated as recurring costs (similar to fuel costs in a fission reactor) and are, therefore, not included in the total capital cost. Indirect costs and interest during construction are accounted for.

As previously indicated, the unit costs for developmental portions of the reactor are corrected consistent with a mature reactor economy (tenth-of-a-kind) based on the cost data base accumulated at the Fusion Engineering Design Center.[9,10] Representative values used in the systems code for ITER and for a reactor version of ITER are shown in Table I.

Economics model

A package to compute the COE, similar to that found in Ref. 4 and updated according to Ref. 11, was added to the systems code. The capital investment computed in the code is converted to an annual cost based on an input constant-dollar fixed charge rate (which is a function of plant life, interest rate, inflation rate, tax rate, etc.).

The net electric power, used in computing the COE, is based on converting the available thermal power to electric power using an input thermal conversion efficiency and subtracting the current drive power and BOP recirculating power.

The costs of the replaceable components, such as the blanket and divertor, are treated as fuel-type charges and are amortized over the lives of these components. The blanket life is based on the neutron wall loading, an input material fluence limitation, and the plant capacity factor. The divertor life is based on a limiting divertor heat flux, a divertor material fluence limitation, and the plant capacity factor. A peaking factor (peak-to-average divertor heat flux) is also input to the code. The divertor plate angle is optimized based on heat flux and vertical build considerations.

Table I. Unit cost for tokamak systems

	ITER base	Reactor base
First wall ($/m^2)		
Inboard	166,000	60,000
Outboard	61,000	60,000
Divertor ($/m^2)	700,000	100,000
Blanket ($/kg)		
Steel		50
Vanadium		400
Lithium		45
Shield (steel)($/kg)	28	28
Magnets		
TF coils OH solenoid ($/kg)		
Nb_3Sn wire	(650–600)x^a	(400–350)x
Case	80	50
Winding process	130	50
EF coils ($/kA·mT)		
Nb_3Sn conductor	2.3	1.6

$^a x$ is the copper-to-superconductor ratio.

BENCHMARK TO ITER

The TETRA-R code is used to simulate the current-driven, steady-state (major radius of 5.5 m, plasma current of 18 MA) version of ITER. Global parameters for this device are shown in Table II. Table III gives the

Table II. TETRA-R code simulation of ITER global parameters

Major radius, m	5.50
Aspect ratio	3.06
Neutron wall load, MW/m^2	1.02
Plasma current, MA	18
Plasma temperature, keV	17.6
Maximum toroidal field, T	11.7
Current drive power, MW	114
Bootstrap current fraction	0.28
Beta, %	5.7
Divertor heat flux, MW/m^2	10
Neutral beam energy, keV	1000.0
Current drive efficiency	
Neutral beam	0.11
Lower hybrid	0.11

Table III. Capital cost for ITER configuration and a reactor based on ITER

	Cost (millions of dollars)	
	ITER	Reactor
Buildings	254	280
First wall	69	40
Blanket and shield	398	398
Structure	86	86
Divertor	90	13
TF magnets	500	291
PF magnets	304	146
Cryostat	17	17
Current drive system	379	379
Vacuum system	61	15
Power conditioning	147	147
Heat transport	104	94
Fuel handling	116	116
I & C	100	21
Maintenance equipment	60	35
Turbine plant	0	75
Electrical plant	28	68
Misc. plant equipment	25	22
Heat rejection	30	25
Plant direct cost	2768	2268
Process contingency	487	0
Indirect cost	1139	850
Project contingency	439	468
Interest during construction	0	623
Total investment	4833	4209

ITER unit cost values and the unit cost values modified for power reactor application. For this comparison, the first wall, blanket, divertor, and all the current drive system costs are included as capital costs, instead of being treated as recurring costs (as in the reactor study). In addition, the water-cooled blanket for ITER is treated as a shield for this comparison.

The direct capital cost for ITER is \$500 million lower when the reactor costing methodology is used; the difference is in the lower tokamak cost (torus, magnets, etc.), higher building cost because of the turbine building, cost of a turbine-generator, and reduced cost of maintenance equipment, I&C equipment, etc., assuming a mature reactor economy as opposed to using development equipment. The total capital investment for ITER is \$600 million lower for reactor costing than for experimental costing. Going from direct capital cost to total capital investment, the reactor costing methodology has no process contingency and has a lower indirect cost percentage but is required to account for interest during construction. The COE in the reactor study is based on the total capital investment, including the interest during construction.

BENCHMARK TO GENEROMAK AND TPSS STUDIES

As part of this study, the TETRA-R code was used to simulate the configuration and performance for power reactors from previous studies to determine the values of design constraints (such as allowed stress in the magnets) required in these designs relative to the ITER values. Cost values for these power reactors were also estimated with the TETRA-R code by applying the unit cost values for ITER.

Generomak

The various elements of the radial build and performance parameters for GENEROMAK are taken from Ref. 4. The simulation indicated that the GENEROMAK beta limit must be a factor of 1.76, the divertor heat flux a factor of 2.0, and the TF coil stress a factor of 1.45 of the ITER limits and constraints. The TETRA-R code gives an estimated COE of 85 mill/kWh, as opposed to the GENEROMAK published value of 53 mill/kWh. Most of the difference lies in the cost of the reactor plant equipment.

TPSS

The performance values and radial build for the TPSS study were taken from Ref. 5. TPSS is characterised as a low-current, high-beta (second stability) tokamak. Accordingly, the simulation shows that the beta limit must be a factor of 8.6 and the divertor heat flux a factor of 1.33 of the ITER limits and constraints. Also, the TPSS confinement enhancement (H-factor) is 5 as opposed to an ITER value of 1.7. The TETRA-R code gives an estimated COE of 87 mill/kWh as opposed to the TPSS published value of 32 mill/kWh (the TPSS published cost is in 1986 constant dollars).

POWER REACTOR PARAMETERS AND COST STUDY

The objective of this study was to determine reactor parameters consistent with a minimum COE for tokamak power reactors. The optimizer feature of the TETRA-R code was used to find the minimum COE with the constrained optimization process described in Ref. 1. The minimum COE was determined for power reactor configurations sized at ITER limits; at 1.5 times ITER limits; at 2 times ITER limits; at 3 times ITER limits; and at 4 times ITER limits. The base ITER limits are shown in Table IV. Note that for 2 times the ITER limits, it is implied that the TF stress limit is raised by a factor of 2 (from 600 to 1200 MPa), the

Table IV. Base ITER limits

Troyon beta coefficient	3.0
Confinement H-factor (Goldston, JAERI, T10, Rebut-Lallia)	2.0
Neutral beam energy, keV	1000
Divertor heat flux, MW/m^2	10
Magnet stress, MPa	800
Fraction TF critical current (at operating field, temperature, and strain)	0.6
Fraction TF current density (for allowed quench temperature rise)	1.0
Minimum TF temperature margin	0.5
Bootstrap current fraction	0.3
Critical toroidal field, T	28
Critical TF coil temperature, K	18
TF coil quench temperature rise, K	150

allowable neutral beam energy is increased by a factor of 2 (from 1000 to 2000 keV), etc. The exception to this procedure is the TF current density, which is taken to be the lowest of the base value of 0.6 times the critical current density at the operating temperature, field, and strain; the current density consistent with a given coil temperature rise during a coil quench; or the current density consistent with a given temperature margin. The fundamental superconducting properties themselves (B_{crit}, T_{crit}, quench temperature rise) are raised by the incremental increase in ITER limits. The bootstrap current is varied from an initial value of 0.3 to a maximum value of 0.99, and the coil temperature during quench is varied from an initial value of 150 K to a maximum value of room temperature as the ITER limits are increased up to a factor of 4.

The net electric power was limited to 1000 MW for this study. It is well known that the COE decreases as the net electric power increases. However, a value of 1000–1300 MW(e) is the power level favored by most electrical utilities. Other fixed parameters are listed in Table V.

Table VI is a concise summary of the results of this study. The minimum COE achievable for each of the five ITER limit multipliers (1, 1.5, 2, 3, and 4) is given; these COEs are based on an "F-factor," the ratio of the actual limit to the allowable limit for each constraint. For a power reactor configured at the ITER limits (multiplier of 1), all of the allowable limits are reached except for the confinement H-factor (2.0 allowed, 1.87 selected) and the net electric power (F = 1.0 allowed, F = 0.48 selected). The divertor heat flux is the primary constraint that keeps the net electric power from reaching the allowable limit of 1000 MW for minimum COE (205 mill/kWh) in this configuration. For a power reactor configured at twice the ITER limits (multiplier of 2), the net electric power, beta, beam energy, TF coil case stress, divertor heat flux, and bootstrap current fraction all reach the allowable limits, and the

Table V. Fixed parameters for this study

Plasma elongation to X-point	2.278
Plasma safety factor at 95% flux	3.16
Blanket	
Inboard thickness, m	0.6
Outboard thickness, m	0.7
Energy multiplication	1.27
Shield	
Inboard thickness, m	0.6
Outboard thickness, m	1.0
Current drive electrical efficiency, %	75
Thermal efficiency, %	40
BOP circulation power, %	6
First wall fluence, MW·yr/m^2	20
Availability, %	75
Plant life, yr	30
Construction time, yr	6
Fixed charge rate	0.0986
Effective cost of money, %	9.57
Inflation rate, %	5.0
Effective tax rate, %	36.6

Table VI. F-factors (ratios of actual limits to allowable limits), H-factor, bootstrap current fraction, and minimum COE for reactors based on ITER limit multipliers of 1, 1.5, 2.3, and 4

	Limit multiplier				
	1	1.5	2	3	4
F-factor					
Net electrical power[a]	0.48	0.91	**1.0**[b]	**1.0**	**1.0**
Beta limit	**1.0**	**1.5**	**2.0**	2.09	1.87
Beam energy	**1.0**	**1.5**	**2.0**	2.96	**4.0**
Current density over TF coil[c]	0.53	1.17	1.23	3.50	3.21
TF coil case stress	**1.0**	**1.5**	**2.0**	2.68	**4.0**
TF coil stress	**1.0**	**1.5**	1.76	2.51	**4.0**
Divertor heat flux	**1.0**	**1.5**	**2.0**	2.02	2.02
H-factor[d]	1.87	2.77	2.32	2.41	2.81
Bootstrap current fraction	0.30	0.45	0.60	0.90	0.99
Minimum COE, mill/kWh	204.8	103.7	79.1	65.0	61.4

[a]Limited to ≤1000 MW.
[b]Bold values are at the allowable limits.
[c]Relative to the ITER value. The allowable value is determined by raising the critical field, critical temperature, and temperature during a coil quench from the ITER values (28 T, 18 K, and 150 K) by the limit multiplier.
[d]The base H-factor at the ITER limit is 2.

COE drops to 79 mill/kWh. At the maximum limits addressed in this study (multiplier of 4), the COE is 61 mill/kWh, with the bootstrap current fraction, TF coil stress, beam energy, and net electric power all at their allowable limits. Table VI shows that the confinement H-factor need not be greater than 3 and that the beta limit and the divertor heat flux limit need not be greater than a factor of about 2.0. The major reduction in COE, as seen from Table VI and Fig. 1, occurs when the ITER limits are multiplied by 2.

The COE, as shown in Fig. 1, is composed of three elements: the capital investment (by far the greatest contributor), operating and maintenance costs (O&M), and the costs of the replaceable components, such as blankets and divertor, which are similar to fuel charges in fission plants. The O&M costs are scaled as a function of power level. The fuel-type costs are determined

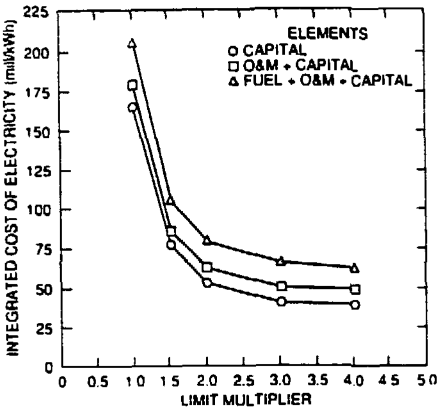

Fig. 1. COE vs limit multiplier.

as a function of the initial costs of the blanket and divertor and a portion of the current drive system cost and amplified by the number of replacements over the plant life, which varies inversely with neutron wall loading. The direct capital cost portion of the COE is shown in Fig. 2 as a function of the limit multiplier. The major reduction in capital cost is associated with the cost of the nuclear island (shield, structure, TF and PF magnets, cryostat, and current drive system). The cost of the BOP, buildings, and other reactor plant equipment is relatively insensitive to variation in the constraint multiplier.

Global parameters of interest for this study are presented in Table VII. For the limit multipliers used, the major radius decreases from 7.51 to 3.97 m, the net electric power increases from 481 to the maximum value of 1000 MW, the plasma current decreases from 19.7 to 9.2 MA, and the neutron wall load increases from 0.99 to 6.44 MW/m^2 with an accompanying decrease in blanket life from 27 to 4.1 years. The current drive power decreases from 128 to 4.8 MW because of the allowable increase in the bootstrap current fraction from 0.3 to 0.99. The required H-factor for confinement increases from 1.87 to 2.81. The maximum toroidal field increases from 11 to 17 T; however, very broad minima exist as a function of maximum field for each constraint multiplier, as shown in Fig. 3. Aspect ratio, over the range of this study, varied from about 3 to 4 for minimum COE. Further increases in aspect ratio would result in a higher COE, as shown in Fig. 4.

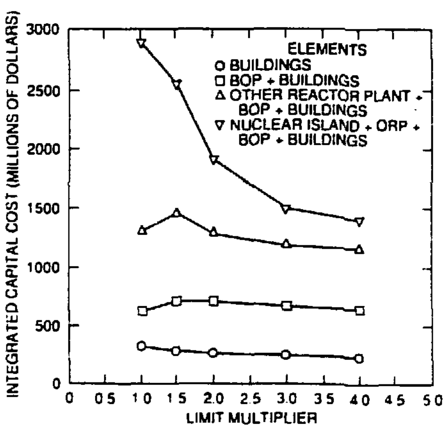

Fig. 2. Direct capital cost vs limit multiplier.

Table VII. Reactor parameters

	Limit multiplier				
	1	1.5	2	3	4
Major radius, m	7.51	6.70	5.11	4.29	3.97
Aspect ratio	3.54	3.83	3.21	3.62	4.06
Net electric power, MW	481	910	1000	1000	1000
Plasma current, MA	19.7	19.6	15.6	11.1	9.2
Beta	0.047	0.056	0.106	0.097	0.076
T_e, keV	21.0	27.0	20.6	13.6	8.4
Fusion power, MW	1321	2190	2327	2231	2197
Neutron wall load, MW/m^2	0.99	2.24	3.38	5.09	6.44
Blanket life, yr	27.0	12.0	7.9	5.2	4.1
Confinement H-factor	1.87	2.77	2.32	2.41	2.81
Current drive power, MW	128	95.2	67.0	21.0	4.7
Bootstrap current fraction	0.3	0.45	0.60	0.90	0.99
B_{maxtf}, T	11	13	13	15	17
J_{wp}, MA/m^2	15.3	33.8	35.8	102	93
J_{oh}, MA/m^2	9.0	17	22.6	47	55

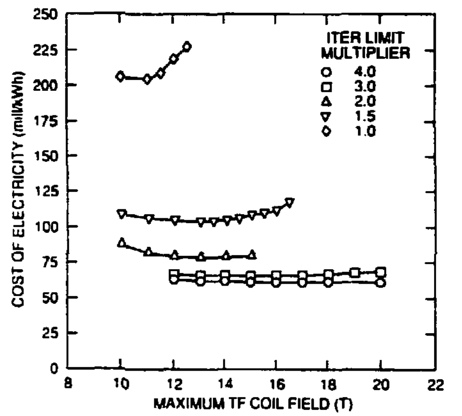

Fig. 3. COE vs maximum TF coil field.

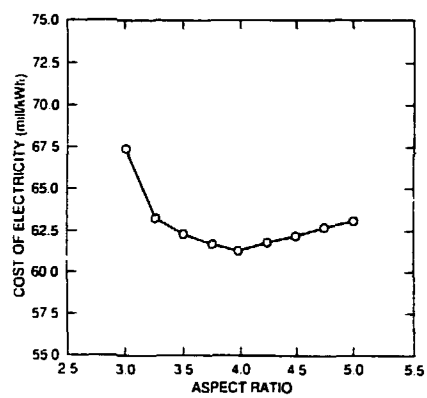

Fig. 4. COE vs aspect ratio for a limit multiplier of 4.

CONCLUSIONS AND DISCUSSION

This study assumed constant kappa, safety factor, blanket thickness, shield thickness, gaps, costing algorithms, component design models, and net electric power. The following conclusions are drawn.

The potential minimum COE is approximately 60 mill/kWh when the design constraints are increased to 4 times the ITER limits. This minimum COE is bounded by the stress in the TF coil, the neutral beam energy, and the 99% bootstrap current fraction (i.e., essentially free current drive).

The largest decrease in the COE is obtained when the design constraints are increased to 2 times those for ITER (from 205 to 79 mill/kWh).

The confinement H-factor need not be >3 to achieve this potential minimum COE. This corresponds to the super-H-mode confinement regime (Ref. 12).

The beta limit need not be more than 2 times that for the first stability regime (Ref. 13).

The divertor heat flux need not exceed the present ITER design limit by more than a factor of 2.

The maximum field at the TF coil need not exceed 18 T, despite an assumed critical field of 112 T, a critical temperature of 72 K, and a quench temperature rise of 300 K for the superconductor in a forced-flow, cable-in-conduit design.

The aspect ratio consistent with the minimum COE lies between 3 and 4 for all cases considered.

These conclusions are valid subject to the following limitations and uncertainties.

1. The reactor concepts are limited to those with a tokamak power core similar to ITER in configuration, in the basic design of major components, and in the cost estimating approaches of size, volume, and weight. Breakthroughs in science and technology that fundamentally alter the basis for these assumptions would lead to different configurations, component designs, and costing approaches that would render our COE results pessimistic.
2. The unit costs per weight or area are assumed, in this study, to be independent of the advances in technology as scaled by the F-factors. This renders our COE results potentially optimistic.
3. The approach of constrained minimization of COE and the rather uniform application of advances in key physics and technology areas have led to a clarification of the "ultimate" potential minimum COE in this "ITER class" of reactors. The actual advances may well be nonuniform in F-factors used and lead to COEs higher than those indicated here. The ability of advancement in one area to substantially reduce COE depends on the relative advances in several of the other key areas considered here.

A detailed report of the findings in this paper will be provided in a separate article (Ref. 14).

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