GA–A23168

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JULY 1999

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This is a preprint of a paper to be presented at the 5th International Symposium on Fusion Nuclear Technology, September 19–24, 1999, in Roma, Italy, and to be published in *Fusion Engineering and Design.*

Work supported by
General Atomics IR&D Funds

GA PROJECT 4437
JULY 1999

ABSTRACT

This paper assesses the technical and economic potential of tokamak power plants which utilize superconducting coil (SC) or normal conducting coil (NC) designs as a function of aspect ratio (A). Based on the results from plasma equilibrium calculations, the key physics design parameters of β_N, β_p, β_T, and κ were fitted to parametric equations covering A in the range of 1.2 to 6. By using ARIES-RS and ARIES-ST as reference design points, a fusion reactor system code was used to project the performance and cost of electricity (COE) of SC and NC reactor designs over the same range of A. The principle difference between the SC and the NC designs are the inboard standoff distance between the coil and the inboard first wall, and the maximum central column current density used for respective coil types. Results show that at an output power of 2 GWe both NC and SC designs can project COE in the respectable range of 62 to 65 mill/kWh at gross thermal efficiency of 46%, with neutron wall loading (Γ_n) ~7 MW/m^2. More importantly, we have learned that based on the present knowledge of equilibrium physics and fusion power core components and system design we can project the performance and COE of reactor designs at least for the purpose of comparative assessment. Tokamak design points can then be selected and optimized for testing or commercial devices as a function of output power, A, and Γ_n for both SC and NC design options.

1. INTRODUCTION

The goal for fusion research has been the production of economically and environmentally acceptable nuclear power. For the magnetically confined tokamak system, equilibrium physics understanding is quite matured and indication of the optimum physics performance has been projected. At the same time, based on the series of conceptual reactor point designs, the geometric constrains and technology limitations for the tokamak system are well understood. Ehst [1] had studied the influence of physics parameters on tokamak reactor design and Stambaugh [2] presented the spherical tokamak path to fusion power. Both studies have used simple expressions to project the normalized beta, β_N as a function of A. To provide further evaluation, we made use of equilibrium physics results and developed a system design code to evaluate the performance of SC and NC designs as a function of A, Γ_n and output power of fusion power plants. This paper has the following outline: Section 2 presents the equilibrium physics results, Section 3 describes the GA-system code, Section 4 presents the engineering assumptions and Sections 5 and 6 present the results and conclusions of this assessment.

2. EQUILIBRIUM PHYSICS

For magnetically confined tokamak fusion reactor concepts, magnetohydrodynamics calculations provide the most robust prediction of plasma equilibrium stability behavior. Miller [3] has found operation points that are ballooning and low-n kink mode stable at high bootstrap fraction of 99% with A varying from 1.2 to 3. We fitted the key plasma parameters of normalized beta (β_N), beta-poloidal (β_p), beta-toroidal (β_T), and elongation (κ), with the inclusion of plasma temperature and density profiles as a function of A. Figure 1 shows the results covering the range of A from 1.2 to 6. We used the parabolic profile $(1–x^2)^\alpha$ for both density and temperature as a function of normalized distance x. With the density profile factor of α_n = 0.634 and the temperature profile factor of α_T = 0.702, we can see that our projection fits the DIII–D high equivalent D-T yield results [4] very well. When compared to the ARIES-RS [5] (SC, A=4) and ARIES-ST [6] (NC, A=1.6) physics projections, our results are more conservative. For the following calculations, in addition to the projected physics performance we also assumed a bootstrap fraction of 90%. The additional power needed for current drive is used to approximate the corresponding power needed for density and temperature profile control in order to maintain optimum physics performance.

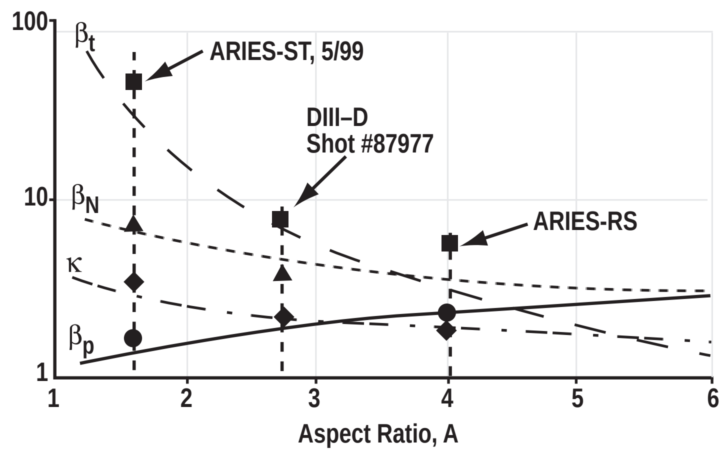

Fig. 1. Tokamak physics performance.

3. GA-SYSTEM CODE

For an integrated performance assessment of tokamak reactors, we put together an iterative system design code. Schematic of the design process of the GA-system code is given in Fig. 2. We started with the specification of physics parameters as a function of A. With the selection of the A, central column conductor radius (R_c), major radius (R_o), and inboard coil standoff distance (Δ_{IB}=shield+blanket+first wall), the geometry of the reactor plasma toroidal chamber can be specified. With the assumption of the plasma triangularity at 0.5 and a scrape off distance (0.5 cm) at mid-plane, the geometry of the plasma can also be specified. As shown in Fig. 2, with the selection of the central column current density (J_c) and conductor radius, and with the process presented in Ref. 2, the toroidal magnetic field strength, plasma ion density and reactor reactivity can be calculated. We have included the option of adding impurities into the core to enhance the radiation of transport power in order to reduce the maximum heat flux at the divertor. Energy balance can then be performed to account for the first wall and divertor heat flux, Γ_n, and re-circulating power of the reactor design. The net output power or the Γ_n can be determined by design iteration. Key design constraints like the stress limit (r) of the selected central column structural material and the water coolant velocity limit (V_{water}) of the Cu-coil design are used for the assessment. As shown in Fig. 2, the key difference between the SC and NC design is the standoff distance of the inboard design. Similar to the ARIES designs [5,6], we selected a standoff distance of 1.3 m for the SC design for superconducting magnet protection, and 0.25 m for the NC design. The latter choice is to minimize the amount of induced radioactive in order to maintain the Cu-alloy to be accepted as class-C waste at the end of reactor life.

Once the reactor geometry and power balance are defined, the costing of the reactor system can be estimated by using the accounting method similar to the ARIES-RS design [5]. It is observed that the fusion power core component life will be a function of maximum Γ_n, and frequent change out will have a negative impact on reactor availability. To account for this effect a simplified availability model is included. This model is based on the assumption that we can

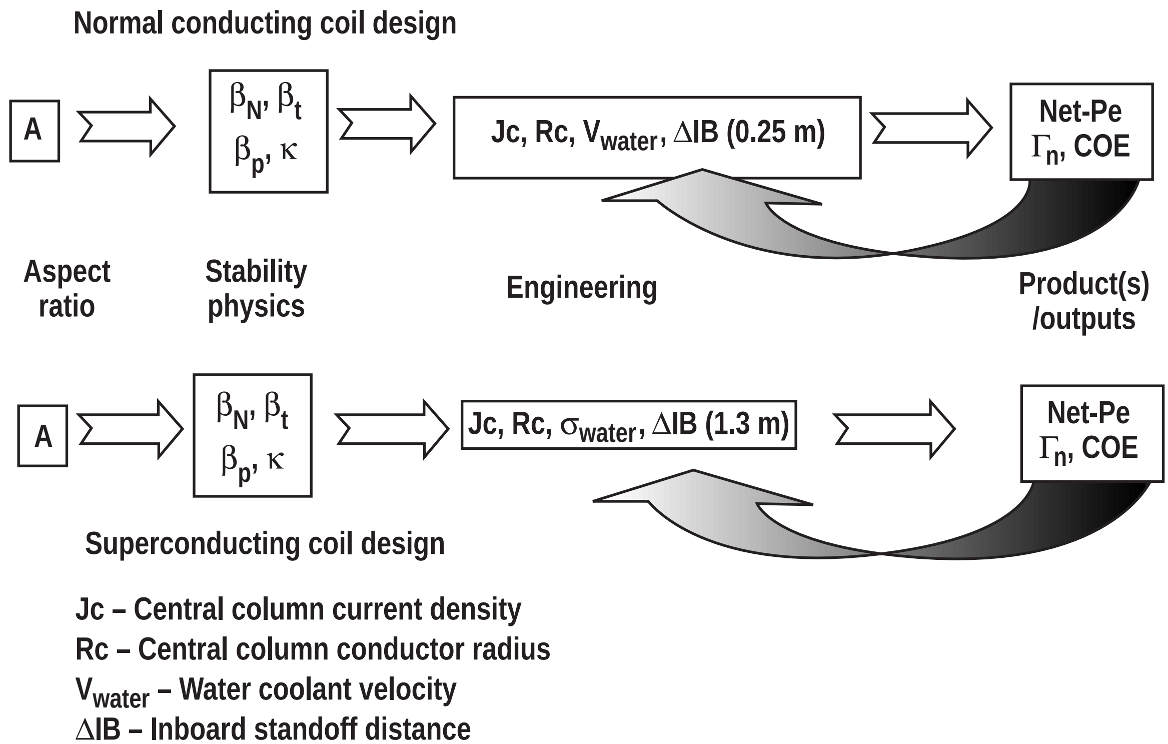

Fig. 2. Key system design process of a toakamak reactor.

achieve an availability of 75% when the maximum Γ_n is at 4 MW/m^2. The variation of availability as a function of maximum neutron wall loading (Γ_n-max) can be representation by,

Availability= 288/(360+6*$\Gamma_{n\text{-}max}$)

A more complete list of the key physics and engineering design-input parameters is presented in Section 4.

3.1. *Normal magnet design*

For a NC tokamak design, significant power consumption is the resistive power loss of the normal conducting toroidal and poloidal field coils. The resistive power loss is a function of coil current and electrical resistivity variation as a function of neutron radiation damage and coil temperature over the lifetime of the central column. The coolant channel design and power input from resistive power and volumetric power generated from high-energy neutrons determine the coil temperature. Similar to Ref. 2, these coupling effects are accounted for in our calculation. To minimize the recirculating power of the NC coils, a tapering factor of 2.5 is used for the central column design, with the narrowest radius located at the mid-plane. Similar to the ARIES-ST [6] design, the 0.5 m thick outboard TF-coil leg is also used as the vacuum vessel. To minimize the temperature of the central column, the water coolant is operated at low temperatures of T_{in}=30°C and T_{out}=50°C.

3.2. *Superconducting coil design*

Relatively, the evaluation of the SC design is much simpler. With the standoff inboard distance of 1.3 m, the required protection of the superconducting coil can be satisfied. Being superconducting, the recirculating power required is assumed to be zero and the power required to maintain the cryogenic system is assumed to be negligible.

For both NC and SC designs, similar to the ARIES-RS [5] and ARIES-ST [6] designs, the outboard coil thickness is assumed to be 0.5 m. The volume of the toroidal and poloidal coil set as dictated by the selected geometry is then used for the costing estimate for both designs.

4. ENGINEERING DESIGN ASSUMPTIONS

Reactor performance and COE are very sensitive to the design input parameters. Key physics and engineering design input parameters for the assessment of SC and NC designs are given in Table 1.

Table 1
Key physics and engineering design input parameters

	SC	NC
Inboard stand-off distance, m	1.3	0.25
Outboard coil thickness, m	0.5	0.5
Central column bore radius, m	1.775	0.0
Divertor vertical height, m	0.5	0.5
Bootstrap fraction, %	90	90
Denisty profile exponent, S_n	0.634	0.275
Temperature profile exponent, S_T	0.702	0.154
Max. ion temperature, keV	18	16
Helium concentration	0.1	0.1
Double null divertor	yes	yes
Water coolant speed limit, m/s	NA	10
TF coil central conductor current density, MA/m^2	31	15
Γ_n and first wall heat flux peaking factor	1.4	1.4
Material fluence life-time, MW.a/m^2	15	15
Thermal efficiency, %	46	46
Current drive	Fast wave	Fast wave
Assumed availability at Γ_n-max=4 MW/m^2	0.75	0.75
Costing assumptions	ARIES-RS(5)	ARIES-RS(5)

5. RESULTS

Based on the selected physics and engineering inputs parameters presented in Section 4, we used the GA-system code to estimate the COE for both SC and NC designs as a function of A, Γ_n, and reactor output power. It should be noted that based on the geometric constraints of the tokamak/toroidal configuration, at a constant output power, lower A would mean larger minor radius and larger first wall surface area which would then leads to lower average Γ_n.

5.1. *Superconducting coil designs*

Figure 3 shows the COE of SC designs as a function of A, reactor output power and average Γ_n at a gross thermal efficiency of 46%. The results show that at constant A, the COE decreases with higher average Γ_n, with correspondingly higher output power. At the output power of 1 GWe, the COE decreases with higher A to ~72 mill/kWh with the increase of average Γ_n to the range of 4 to 6 MW/m^2. At the higher output power of 4 GWe, the COE minimizes at high average Γ_n but due to the effect of the loss in availability at higher neutron wall loading, the COE has a flat minimum around 3<A<4. At the high but acceptable output power of 2 GWe, at A ~ 4 and average Γ_n around 7 MW/m^2 the COE is shown to be around 62 mill/kWh which is quite acceptable when compared to the cost projection of other energy sources in the future. Key physics, engineering and costing parameters of the 2 GWe SC design are listed in Tables 2 and 3.

5.2. *Normal conducting coil designs*

Figure 4 shows the COE of NC tokamak reactor designs as a function of A, reactor output power and average Γ_n at a gross thermal efficiency of 46%. The results show that the COE has a minimum around A = 1.5 to 1.6. The minimum is broader at lower output power of 1 GWe, and is more pronounced at 4 GWe. Due to the increase of re-circulating power at higher A, the COE increases for A > 2. For A<1.5, the physical size of the reactor gets much bigger and the average Γ_n gets lower for the same output power, therefore the COE increases. Figure 4 shows that at an output power of 2 GWe and 1.5<A<1.6, the minimum COE is at around 65 mill/kWh with the average Γ_n around 7 MW/m^2. Details of the design and costing parameters of a 2 GWe NC design is given in Table 2 and Table 3.

As shown in Figs. 3 and 4, for both SC and NC options, at constant A, higher average Γ_n can lead to COE < 65 mill/kWh, but the output power would have to be increased from 1 GWe to about 2 GWe. At the same output power, due to the effect from lower re-circulating power SC designs can have lower COE by about 3–8 mill/kWh than NC designs. At output power of 2 GWe both NC and SC designs can have COE in the respectable range of 62 to 65 mill/kWh and with average Γ_n ~7 MW/m^2.

Details of the design and costing parameters of a 2 GWe SC and NC designs are given in Table 2 and Table 3.

Table 2
Physics and engineering parameters of 2 GWe SC and NC reactor designs

	SC	NC
plasma aspect ratio, A	4	1.6
plasma vertical elongation, κ	1.769	2.799
minor plasma radius, a, (m)	1.463	2.175
major toroidal radius, R_o, (m)	5.851	3.48
plasma volume, (m^3)	411.2	817.2
first-wall surface area, (m^2)	497.5	597.1
radial profile exponent for density, s_n	0.634	0.275
radial profile exponent for temperature, s_T	0.702	0.154
toroidal beta, (%) volume averaged	2.8	37.5
poloidal beta, (%) volume averaged	2.29	1.51
on-axis toroidal field, (T)	10.1	2.56
plasma current, (MA)	11.7	29.2
plasma ion temperature, (keV) peak	18	16.0
peak plasma electron density, n_e, ($10^{20}/m^3$)	5.04	2.99
peak plasma ion density, ($10^{20}/m^3$)	3.77	2.22
energy confinement time (τ_E-ITER98p(y), s)	0.772	0.552
Kr concentration (used to distribute transport power)	0.00085	0.00129
effective plasma charge, (Z_{eff})	2.27	2.827
average fusion power density, (MW/m^3)	10.40	5.77
fusion power, (MW)	4274	4717
number of TF coils	16	12
mass of coil set, (tonne)	3236	3086
TF central column avg. current density, (MA/m^2)	31	15
TF coil resistive power consumption, (MWe)	0	242.6
recirculating power, (MWe)	314.7	526.8
thermal conversion efficiency, (%)	46	46
CD/heater [FWCD*] power, (MW)	100	82.37
plant Q	7.35	4.8
total useful thermal power, (MW)	5028	5499
gross electrical output power, (MWe)	2313	2530
net electrical output power, (MWe)	1998	2003
average 14.06-MeV neutron load, (MW/m^2)	6.588	6.188
Blanket energy multiplication	1.1	1.1
Average first wall heat flux, (MW/m^2)	1.853	1.68
Divertor max. heat flux, (MW/m^2)	3.32	3.53

*Fast wave Current Drive

Table 3
Costing parameters of 2 GWe SC and NC designs

Account Number	Account Title	SC M$ (1992)	NC M$ (1992)
20.	land & land rights	10.4	10.4
21.	structures & site facilities	635.6	695.2
22.	reactor plant equipment	1746	1820
22.1.1	FW/blanket/reflector	112.9	135.5
22.1.2	shield	191.6	99.1
22.1.3	magnets	300	204.3
22.1.4	supplemental-heating/CD systems	203	167.5
22.1.5	primary structure & support	60.9	73.2
22.1.6	reactor vacuum systems	181.6	218
22.1.7	power supply, switching & energy storage	87.35	93
22.1.8	impurity control	14.5	14.5
22.1.9	direct energy conversion system	0.0	0.0
22.1.10	ECRH breakdown system	4.6	4.6
22.1	reactor equipment	1169	1029
22.2	main heat transfer & transport systems	407.8	345.9
23.	turbine plant equipment	450.3	472.8
24.	electric plant equipment	175	185.9
25.	miscellaneous plant equipment	88.7	94.4
26.	special materials	21.26	23.26
90.	direct cost (not including contingency)	3115	3297
91.	construction services & equip.	373.8	395.7
92.	home office eng. & services	162	171.5
93.	Field office eng. & services	186.9	197.8
94.	owner's cost	576	609
96.	project contingency	767.5	812.5
97.	interest during constr. (IDC)	854.8	904.9
99.	total cost ($\$10^6$)	6035	6389
	unit overnight cost ($/kWe)	5181	5484
	capital return (mill/kWeh)	48.3	50.6
	plant availability	0.693	0.699
	decommissioning (mill/kWeh)	0.5	0.5
	fuel (mill/kWeh)	0.06	0.06
	LSA*=2, total COE† (mill/kWeh)	61.64	63.83

*Level of safety assurance

†COE includes replacement costs, fusion power core components operate to fluence of 15 MW.a/m^2

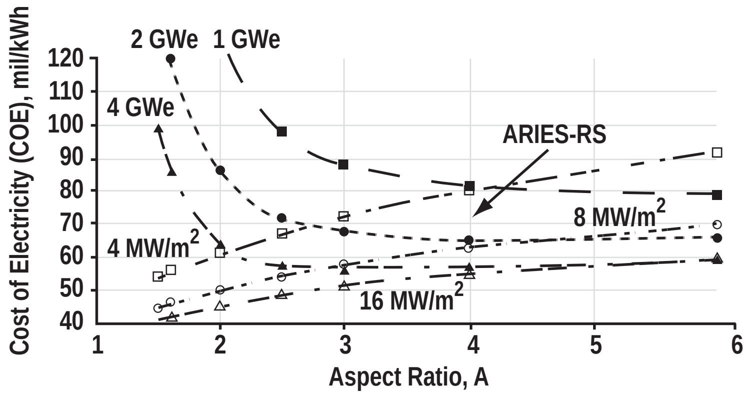

Fig. 3. COE of superconducting coil tokamak reactor designs.

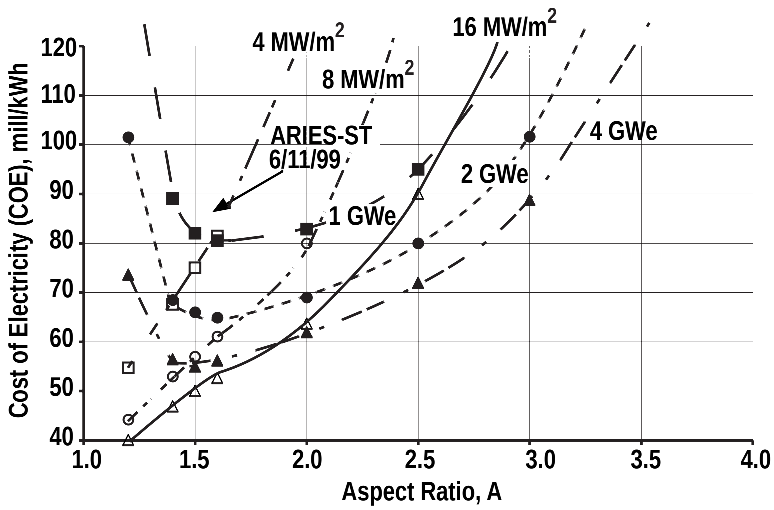

Fig. 4. COE of normal conducting coil tokamak reactor designs.

6. CONCLUSIONS

This paper projects the technical and economic potential of SC and NC tokamak designs as a function of A. Based on the results from plasma equilibrium calculations, utilizing system evaluation we projected the reactor performance and COE as a function of output power and average Γ_n. Results showed that due to the toroidal geometry of tokamak reactors, for the same output power, lower A would mean a larger machine with larger first wall surface area and therefore lower Γ_n. For the same A, higher Γ_n would mean lower COE but higher electrical output power. At a gross thermal efficiency of 46%, due to the lower re-circulating power, for the same output power and the engineering assumptions that we have made, the SC designs can have lower COE, by about 3–8 mill/kWh, than the NC designs. To minimize the re-circulating power while maintaining reasonably higher Γ_n, minimum COE of NC designs optimized to lower A in the range of $1.5 < A < 1.6$. SC coil designs have minimum COE for $A > 4$. At a gross thermal efficiency of 46% and an output power of 2 GWe both NC and SC designs can have COE in the respectable range of 62 to 65 mill/kWh with Γ_n ~7 MW/m^2. More importantly, we have learned that based on the present knowledge from equilibrium physics and the fusion power core components and system designs we can project the performance and COE of reactor designs at least for the purpose of comparative assessment. Design points can then be selected and optimized for testing or commercial devices as a function of output power, A, Γ_n for both SC and NC tokamak design options.

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ACKNOWLEDGMENT

Work supported by General Atomics IR&D funds.