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Neutron wall loading of Tokamak reactors

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Abstract

Neutron wall loading (Γ_n) is a key parameter for the selection of fusion power core component materials. It also impacts the economic, performance, design, safety and environmental aspect of the fusion power plant. This paper reports the determination of the range of Γ_n for economically competitive fusion power plants based on the analysis that couples the MHD stability physics results to a system design code. Cost of electricity (COE) was selected as the parameter to be minimized. For both normal conducting and superconducting coil options, at thermal efficiency of 46% and at the power output range of 1–2 GW(e) the average neutron wall loading is 4–7 MW/m^2. For a given power output, higher thermal efficiency will allow lower Γ_n. At the above range of Γ_n, in order to have economical fusion power reactors, for the solid first wall design option, high thermal efficiency of 46% to 57.5% requires the use of alloys like V and W-alloy, respectively. The corresponding COE can be projected to be in the economically competitive range of 62–54.6 mill/kWh.

1. Introduction

The goal for fusion research has been the production of economically and environmentally acceptable nuclear power. To minimize the capital cost of the fusion power core, high power density that translates to high neutron wall loading has been proposed [1]. This paper reports the assessment on the range of neutron wall loading that would be optimum for an economical Tokamak power reactor. To provide an integrated picture, this assessment includes the physics performance as a function of the aspect ratio A for normal conducting and superconducting magnet designs, and the effects from power output and thermal efficiency. The selection of structural material and blanket design will also directly impact the resulting thermal efficiency of the fusion power core system. This paper has the following outline: Section 2 presents the Tokamak equilibrium physics results, Section 3 summarizes the GA-system code, Section 4 presents the engineering assumptions and the results for normal and superconducting magnet designs, Section 5 presents the helium-cooled W-alloy, Li-breeder first wall blanket design as an example and Section 6 presents the conclusions of this assessment.

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2. Tokamak equilibrium physics

For a magnetically confined Tokamak system, equilibrium physics understanding is quite mature and the optimum physics performance has been projected. At the same time, based on the series of conceptual reactor point designs, the geometric constraints and technology limitations for the Tokamak system are also well understood. Ehst [2] has studied the influence of physics parameters on Tokamak reactor design and Stambaugh [3] 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. β_N is a key plasma parameter that can be used to estimate beta-toroidal (β_T) which shows the effectiveness in the use of confinement magnetic field and beta-poloidal (β_p) which is an indication on the need of additional current drive power to maintain the plasma current for steady state operation. On equilibrium physics, Miller [4] has found operation points that are stable (ballooning and low-n kink mode) at high bootstrap fraction of 99% with A varying from 1.2 to 3. We fitted the key plasma parameters of β_N, β_p, β_T, and plasma elongation (κ), with the inclusion of plasma temperature and density profiles as a function of A from 1.2 to 6 [5]. When compared to the Superconducting Coil (SC) ARIES-RS [6] ($A=4$) and the normal conducting coil (NC) ARIES-ST [7]

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($A = 1.6$) 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%.

3. General atomics (GA) – system code

For an integrated performance assessment of Tokamak reactors, we put together an iterative system design code. We started with the specification of physics parameters as a function of A. With the selection of A, central column conductor radius (R_c), major radius (R_o), and inboard coil stand-off 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 of 0.5 cm at mid-plane, the geometry of the plasma can also be specified. With the selection of the central toroidal magnetic field column current density and conductor radius, the toroidal magnetic field strength, plasma ion density and reactor reactivity can be calculated [3]. 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. The net output power or then can be determined by design iteration. The key difference between the SC and NC design is the stand-off distance of the inboard design. Similar to the ARIES designs [6,7], we selected a stand-off 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 radioactivity in order to maintain the Cu-alloy as class-C waste at the end of inboard blanket and central column lifetime of 4 yr. 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 [6]. The average Γ_n and maximum Γ_n at outer midplane can

Table 1
Key physics and engineering design input parameters for superconducting (SC) and normal conducting (NC) designs

	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
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
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 $\Gamma_{n\text{-max}} = 4$ MW/m^2	0.75	0.75
Costing assumptions	ARIES-RS(6)	ARIES-RS(6)

Table 2
Physics and engineering parameters of 2 GW(e) 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)	1.469	2.197
Major toroidal radius, R_o (m)	5.876	3.515
Plasma volume (m^3)	416.5	842
First-wall surface area (m^2)	501.9	609
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)	10.2	2.6
Plasma current (MA)	11.9	29.9
Plasma ion temperature (keV) peak	18	16.0
Peak plasma electron density, n_e (1020/m^3)	5.04	3.04
Peak plasma ion density (1020/m^3)	3.86	2.29
Energy confinement time (τ_E, s)	0.715	1.14
(τ_E-ITER98p(y), s)	0.717	0.55
Kr concentration (used to distribute transport power)	0.00092	0.00129
Effective plasma charge (Zeff)	2.36	2.827
Average fusion power density (MW/m^3)	10.89	6.12
Fusion power (MW)	4535	5148
Number of TF coils	16	12
Mass of coil set (tonne)	3236	2480
TF central column avg. current density (MA/m^2)	31	15
TF coil resistive power consumption [MW(e)]	0	311
Recirculating power [MW(e)]	330	610
Thermal conversion efficiency (%)	46	46
CD/heater [FWCD] power (MW)[a]	106.7	87.8
Plant Q	7.04	4.27
Total useful thermal power (MW)	5047	5670
Gross electrical output power [MW(e)]	2322	2608
Net electrical output power [MW(e)]	1992	1998
Average 14.06 MeV neutron load (MW/m^2)	6.93	6.62
Blanket energy multiplication	1.1	1.1
Average first wall heat flux (MW/m^2)	2.009	1.79
Divertor max. heat flux (MW/m^2)	2.244	3.72

[a] Fast wave current drive.

be determined by assuming a peaking factor of 1.4. Since the fusion power core component life will be a function of maximum Γ_n, 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 achieve an availability of 75% when the maximum Γ_n is at 4 MW/m^2. We also assumed that the material neutron fluence limit is 15 MW a/m^2 and a first wall and blanket change out time of three months. The variation of availability as a function of maximum neutron wall loading ($\Gamma_{n\text{-max}}$) can be represented by, Availability $= 288/(360 + 6\times\ \Gamma_{n\text{-max}})$. A more complete list of the key physics and engineering design-input parameters is presented in Tables 1 and 2.

For both NC and SC designs, similar to the ARIES-RS [6] and ARIES-ST [7] 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. Results

Based on the selected physics and engineering inputs parameters, we used the General Atomics-system code to estimate the COE for both SC and NC designs as a function of A, Γ_n, reactor output power and thermal power conversion efficiency. 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 lead to lower average Γ_n.

4.1. Normal conducting magnet design

For a NC Tokamak design, significant power consumption is associated with 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. Similar to the ARIES-ST [7] 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 $T_{out} = 50°C$.

Fig. 1 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

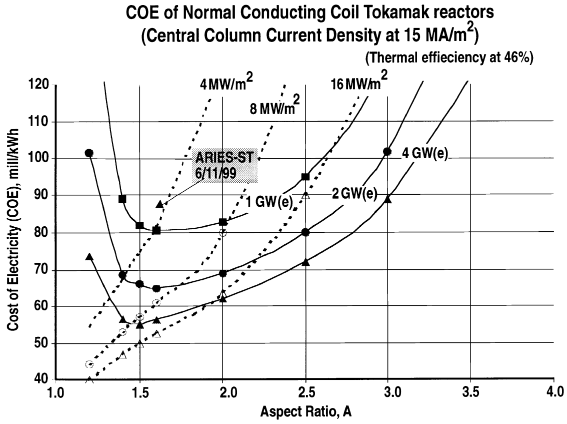

Fig. 1. Cost of electricity of normal conducting coil Tokamak reactor designs.

show that the COE has a minimum around $A = 1.5$–1.6. The minimum is broader at lower output power of 1 GW(e), and is more pronounced at 4 GW(e). 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. Fig. 1 shows that at an output power of 1–2 GW(e) and $1.5 < A < 1.6$, the minimum COE decreases from 82 to 65 mill/kWh with the average Γ_n increases from around 3–7 MW/m^2. Details of the design parameters of a 2 GW(e) NC design are given in Tables 1 and 2.

4.2. Superconducting coil design

Relatively, the evaluation of the SC design is much simpler. With the stand-off 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.

Fig. 2 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. Take $A = 4$ as an example, at the output power range of 1–2 GW(e), the COE decreases from 76 to 63 mill/kWh with the increase of average Γ_n from 4 to 7 MW/m^2.

At the higher output power of 4 GW(e), the COE minimizes at high average $\Gamma_n > 8$ MW/m^2, but due to the effect of the loss in availability at higher neutron wall loading, the COE has a flat minimum around $3 < A < 4$.

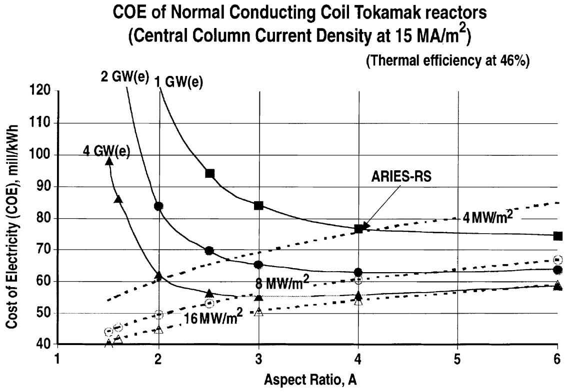

Fig. 2. Cost of electricity of superconducting coil Tokamak reactor designs.

4.3. Impacts from thermal efficiency

Ref. [1] presented a V-alloy helium-cooled first wall blanket design with an estimated gross – η_{th} thermal efficiency of $\sim 46\%$ [1]. To increase the gross thermal efficiency, a W-alloy helium-cooled design [8] shows the possibility of having a gross – η_{th} of 57.5%. We evaluated the impacts on Γ_n for these efficiencies. The conceptual design of the W-alloy blanket is summarized in the next section. Fig. 3 shows the COE, the plasma volume, the Greenwald density ratio and Γ_n variation with power output and gross thermal efficiency. The superconducting coil, $A = 4$ reactor is used as an example. As shown, the COE decreases with higher power output and higher thermal efficiency. At 2 GW(e) and $\eta_{th} = 57.5\%$, the COE is 54.6 mill/kWh, which would be competitive to advanced fission power plant. The volume of the plasma (P-vol) does not change much. The densities required at the temperature assumed are 1.23–1.3 times the Greenwald limit. The density can be reduced to acceptable value by increasing the plasma temperature to about 25 keV. Correspondingly, with the fixed power output of 2 GW(e), higher thermal efficiency of 57.5% would lead to lower average neutron wall loading of 5.6 MW/m^2.

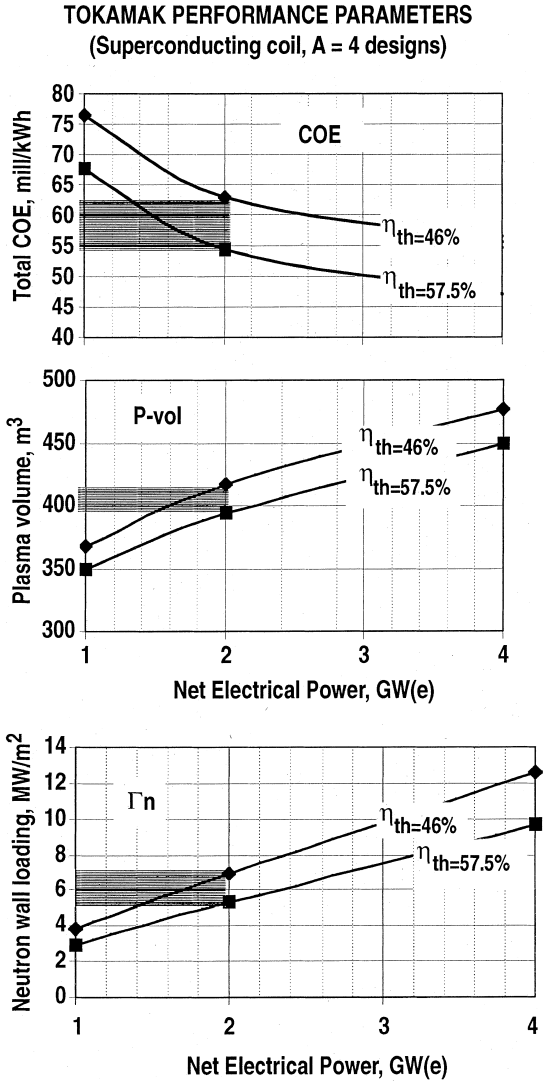

Fig. 3. Tokamak performance parameters.

5. Helium-cooled W-alloy, Li-breeder first wall blanket design

As an example, Fig. 4 shows the schematic of the helium-cooled W-alloy first wall blanket design [8]. The selected alloy is W–5Re. To meet the proposed design temperature range of W–5Re, $800^\circ < T < 1400^\circ$C, the helium coolant was selected to have an inlet temperature of 800°C and an outlet temperature of 1100°C. The first wall is made up of separate units, which in this case are connected to separate cooling manifolds at the back of each module. The first wall units consist of multiple parallel passages connected through an integral manifold to round inlet and outlet connections. The round connections should be easier to fit up and minimize thermal stresses at the interface. Identified critical issues for this design are the acquisition of the fusion neutron irradiated properties of W-alloy, and the development of the fabrication of W-alloy first wall and blanket components.

As shown in Fig. 4, the piping is routed in two circuits. The first circuit includes the first wall and part of the interior blanket tubing. Helium at 800°C enters the first wall through the supply manifold and exits into the first wall outlet manifold at 950°C. The helium is then routed inside the blanket zone, a container filled with

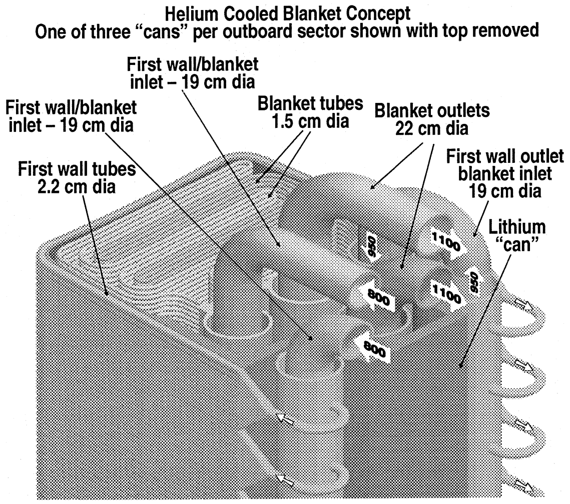

Fig. 4. W-alloy helium-cooled blanket concept.

lithium, to the first supply manifold for the tubes. The first tube circuit exits into a return manifold at 1100°C. The second helium circuit is also fed at 800°C and exits at 1100°C routing through the blanket zone only. The helium at outlet temperature of 1100°C can be coupled to a closed cycle gas turbine power conversion system and the corresponding projected η_{th} is 57.5%. For a 2 GW(e) reactor, the estimated average and maximum neutron wall loading are 5.6 and 7 MW/m^2, respectively. The corresponding estimated COE is 54.6 mill/kWh for this 2 GW(e) plant.

6. Conclusions

Neutron wall loading is a key parameter for the selection of fusion power core component materials. It impacts the economic, performance, design, safety and environmental impact of the fusion power plant. We determined the range of neutron wall loading for economically competitive fusion power plants based on the analysis that couples the MHD stability physics results to a system design code including the selected power output and the geometric impact of a toroidal reactor. Results show that normal conducting toroidal coil reactor COE optimizes at lower A in the range of 1.4–1.6. Superconducting coil reactor COE decreases with the increase of A. At selected net power output, normal and superconducting coil designs will optimize to similar COEs and similar range of neutron wall loading. For the Tokamak confinement concept and with optimum physics design, higher neutron wall loading designs will lead to lower COE, but are limited by the maximum power output to be acceptable to future utilities. For both normal conducting and superconducting coil options, and at the power output range of 1–2 GW(e) the range of neutron wall loading is from 3 to 7 MW/m^2. For the solid first wall design, in order to have economical fusion power reactors, high thermal efficiency of 46–57.5% requires the use of high temperature alloys like V- and W-alloys, respectively. The corresponding COE is projected to be in the economically competitive range of 62–54.6 mill/kWh.

Acknowledgements

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