Toroidal Reactor Designs as a Function of Aspect Ratio

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Abstract. This paper presents a "common basis" systems study of superconducting (SC) and normal-conducting (NC) DT-burning fusion power and materials testing reactor designs. Figures-of-merit for power and materials-testing reactors are respectively; projected cost-of-electricity (COE) and direct cost (DC). A common 0-D plasma modeling basis is used and the plasma geometry and engineering aspects of the SC and NC designs are treated in an equivalent manner that is consistent with the limitations of their respective magnet technologies and other design constraints. Aspect ratios A in the range $1.2 \leq A \leq 6$ and plasma elongations in the range $1 \leq \kappa \leq 3$ are explored and a MHD stability (beta limit) physics basis that accurately describes the increase of normalized beta β_N and toroidal beta β_T with a decreasing A and/or increasing κ is incorporated. With this MHD basis taken into account and with the usual reactor geometry, physics and engineering constraints and costing bases applied, the results of the study show that for power reactors the minimum COE is pointing towards lower A~2 than generally found in previous studies. The minimum is broader with higher κ. For test reactors with similar fusion power output, the direct cost for NC options is significantly lower than for SC coil options. With the NC category, testing designs that combine intermediate A and higher elongation show promise as a D-T burn next step device that could provide scientific and testing data to support future SC and NC reactors. For example, a NC coil design with A~2, κ=3 could produce 200 MW fusion power at 1.23 MW/m^2 average neutron wall loading at a total direct cost of about $643 M. This NC design with a fissile blanket could also convert ~1270 kg of fission reactor waste per full power year.

1. Introduction

Attainment of economically and environmentally-acceptable nuclear power has been a long-standing goal for fusion development, and numerous systems studies and specific concept design studies have attempted to identify adequate or optimal concepts for achieving either this goal of power production or the related enabling goal of testing materials for a fusion reactor. Comprehensive studies for ITER [1] with plasma aspect ratio A~3, indicate that the development cost for fusion based on conservative application of the present physics and engineering bases for an inductively-sustained SC tokamak system will be relatively high. Possibilities for obtaining better economics by means of improved ("advanced") plasma performance plus full non-inductively sustained steady-state operation in a SC system at higher A (~4) have been studied in the context of various future commercial reactor concepts [2] (ARIES-AT). A lower-A normal-conducting NC magnet option [3] based upon the 'spherical torus' configuration has been proposed as an alternate approach. But these studies each apply a physics basis that is selected (or projected) to best describe the plasma aspect ratio and magnetic field regimes that the respective designs aspire to. It is therefore of great interest to map the performance of both SC and NC power and test reactors in a self-consistent manner as a function of A and κ.

Ehst [4] had studied the influence of physics parameters on tokamak reactor design and Stambaugh [5] presented the spherical tokamak path to fusion power. Both studies have used simple expressions to project the plasma performance as a function of A. Our earlier paper [6] on the subject employed a dependence of κ as a function of A in the form of $\kappa = (0.277+9.129/A-5.748/A^2)$. Results show that NC power reactor designs have minimum for $1.4<A<1.6$, and SC designs have flat minimum for $A>3$. For this paper we evaluate the test and power reactor designs as a function of κ = 1.5, 2 and 3 and for $1.2 \leq A \leq 6$.

2. MHD Stability Basis

The plasma equilibrium and corresponding ideal MHD stability (beta) limit for a tokamak system is well understood. For the range of A from 1.2 to 3, Miller [7] found plasma equilibria that are stable to infinite-n ballooning and with assumed wall stabilization of low-n kink modes at a bootstrap fraction of 99% for κ=1.5, 2 and 3. We parametrized these results to perform parametric investigations of A and κ variations with our systems modeling. Equations (1) and (2) show β_N, β_T as a function of A with the effect of the plasma temperature and density form factors taken into account. The Eq. (3) relationship of β_p and β_T with peakfactoring is based on the DIII–D high equivalent D-T yield results [8]. For our calculations, we used a bootstrap fraction of 90% to further optimize β_N and β_T, and approximate the power needed for direct (non-bootstrap) current drive and plasma profile control.

$$\beta_N = \frac{\left(3.09 + \frac{3.35}{A} + \frac{3.87}{A^{0.5}}\right) \cdot \left(\frac{\kappa}{3}\right)^{0.5}}{\text{peakfactor}^{0.5}} \tag{1}$$

$$\beta_T = \frac{25}{\beta_p} \cdot \left(\frac{1+\kappa^2}{2}\right) \cdot \left(\frac{\beta_N}{100}\right)^2 \tag{2}$$

where β_p is given by,

$$\beta_p = f_{bs} \cdot \frac{\sqrt{A}}{(0.663 + 0.019 \cdot \kappa) \cdot \left[\int_0^1 \left(1-x^2\right)^{S_t} \cdot \left(1-x^2\right)^{S_n} dx\right]^{-0.25}} \tag{3}$$

where x is the normalized minor radius and S_t and S_n are temperature and density form factors, and f_{bs} is the bootstrap fraction.

Equations (1) and (2) show that β_N and β_T increase with a decreasing A and/or increasing κ.

3. Method of Calculation

We employ the usual tokamak systems modeling approach, with the inter-relationships among physics, plasma geometry and engineering parameter selection based on the SC or NC design concept being examined and the satisfaction of the corresponding technology limits. Details of our systems design approach and parametrization methodology are given in Ref. [6]. In summary, we start with the inboard toroidal field (TF) coil radial build. With the central column current density, A, κ, plasma profiles, and the standoff distance between the coil and the plasma given, β_N is calculated or specified. The necessary toroidal magnetic field strength, toroidal beta β_T, and reactor geometry are then calculated using Eq. (1). The plasma density can then be determined and the fusion power can be calculated [5,6]. Key specified parameters such as power output or Γ_n are determined iteratively. With further specification of reactor component costs and power conversion efficiencies, the direct and total cost and COE are estimated. The power reactor costing assumption is based on the ARIES systems code [9], based on 1992$, and the test reactor costing is approximated from the ITER-FDR design [1], based on 1999$. Input parameters for the SC and NC coil steady state power and test reactors are given in the next section.

4. Input Assumptions and Parameters

For evaluations in this paper, unless specified otherwise, the common input parameters are: TF coil resistive power conversion efficiency 90%, blanket energy multiplication 1.1, plasma chamber neutron fluence life 15 MW-yr/m^2, plasma triangularity 0.4, density form factor S_n = 0.25, temperature form factor S_t = 0.25, mid-plane scrape-off-layer thickness 0.05 m, peak plasma temperature 20 keV, bootstrap fraction 90%, additional vertical height for double-null divertor chamber 0.5 m each end. Other input parameters for the different design options are given in Table I.

TABLE I: INPUT PARAMETERS FOR SC AND NC POWER AND TEST REACTORS

Reactor application	Power	Test	Power	Test
TF magnet type	SC	SC	NC	NC
TF central column current density, MA/m^2	40	40	15.4	15.4
Number of TF coils	16	16	12	12
Outboard TF coil leg thickness, m	0.2	0.2	0.7	0.7
Inboard coil to first wall stand-off distance, m	1.04	1.04	0.25-0.5[1]	0.1
Water coolant volume fraction	NA	NA	0.15	0.15
Water temperatures: T_{in}/T_{out} (°C/°C)	NA	NA	30/50	30/50
Central column top to bottom taper	1	1	1.5	1.5
Thermal conversion efficiency, %	46	33	46	33

[1]To account for the need for inboard tritium breeding at higher A, the standoff distance is increased linearly from 0.25 m to 0.5 m for A increasing from 1.2 to 4

Geometrical constraints appropriate to the magnet technology being invoked are applied. SC designs have central bores to accommodate OH-coils and a bucking cylinder structure. To take structural loading limitations into account, the bore radius is chosen such that the product of average current and average B-field of the central column is a constant, equal in value to that of ARIES-AT [2], but the minimum core radius is also taken to be ≥ 0.5 m. NC designs have no central bore. For the NC central column, the Von Mises stress is limited to < 138 MPa [10].

To account for the effect of Γ_n, a simple availability model is included in our COE calculation. The assumption is that the availability will be 75% when the Γ_n is 4 MW/m^2. With an assumed plant life of 30 years and a total blanket change-out time of 3 months, the variation of reactor availability as a function of maximum neutron wall loading ($\Gamma_{n,max}$) is approximated by availability = $(294–6*\Gamma_{n,max})/360$. We also limit the average neutron wall loading to ≤ 8 MW/m^2.

5. One GWe SC and NC Designs

Figure 1 shows, as a typical example, the design parameters of 1 GWe SC and NC reactor designs as a function of κ and A. Since plasma performance decreases with higher A and/or lower κ, in order to maintain the same power, reactor size must increase and the neutron wall loading falls. This leads to higher COE. On the other hand, as A falls below 2, due to geometry the first wall area increases. These effects lead to a maximum in Γ_n and a minimum in COE as a function of A. As Fig. 1 shows, COE is near-minimum for A~2 and the minimum becomes broader for higher κ. Also, as is clearly shown in Fig. 1, plasma densities required normalized to the Greenwald density $n_{GW}(10^{20}\ m^{-3}) = I(MA)/a^2(m)$ decrease for higher κ. Experiments will be needed to show the possibility of the design cases with n ≥ n_{GW} shown in Fig. 1.

6. More Detailed Comparison of SC and NC Designs

Figure 2 shows more complete data for the SC and NC designs, with various combinations of SC and NC power and test reactor designs at different A and κ. For each of the six figures, the three upper curves show the direct cost ($M) of test reactors as a function of A at 0.1, 0.2 and 0.5 GW-fusion power output. The lower three curves show the COE (mill/kWh) of power reactors as a function of A at 0.5, 1 and 2 GWe net electrical power output. The following observations can be made: first, power reactor designs have minimum COE for A~2. Higher κ results in a broader minimum. Second, within uncertainity of our calculation, the minimum COEs for SC and NC power reactors are similar. Third, for test reactors, NC design direct cost is lower than SC design direct cost, but NC designs have higher recirculating power (not shown in Fig. 2).

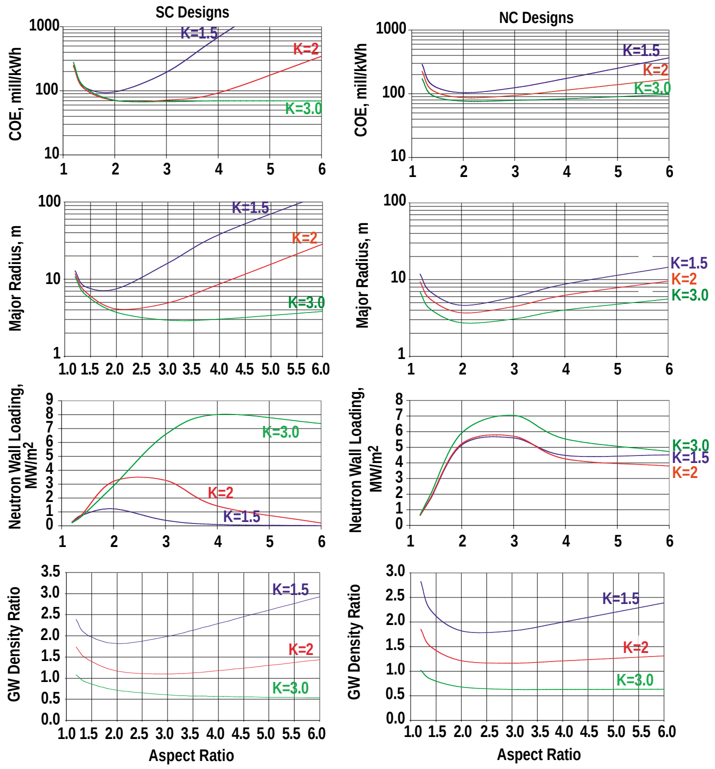

FIG. 1. 1 GWe SC and NC reactor design parameters.

7. Testing Device

Seperate application of the modeling basis described herein shows that at a direct cost of <\$650M, a 200 MW fusion power design at A = 2 and κ = 3 with an average Γ_n of 1.2 MW/m^2 is an interesting option for a next-step D-T burn device. Given that our studies show that both SC and NC designs optimize at A ≅ 2, the scientific knowledge from this design could be applicable for SC and NC power reactor development and the resulting wall loading will be adequate for fusion components nuclear testing.

This A = 2 NC coil device could also be used for the demonstration of other applications of fusion neutrons, e.g. for burning of fission reactor waste materials [11]. With a fissile blanket energy multiplication of 20 and at a thermal efficiency of 33%, this device could produce 977 MWe net electrical power, at a COE of 62 mill/kWh and convert about ~1270 kg of fission reactor waste per full power year. The COE estimated here does not include costs for the handling of fissile materials and the higher blanket power densities and safety considerations associated with the incorporation of fissile material. However, the revenue generated from the disposal of fission reactor wastes may compensate for the extra costs incurred.

8. Conclusions

Designs of steady state SC and NC power and test reactors have been studied as a function of A and κ. The effect of MHD stability as a function of A and k is included. Power reactor designs have minimum COE at low A~2. Higher κ results in a broad minimum in A. Direct cost of NC test reactors, for similar fusion power, is significantly lower than equivalent SC designs. An intermediate-A NC design with high elongation could provide scientific and testing data to support future SC or NC reactors. This may be a device that could lead to a reduced development cost for fusion. A NC test reactor at A~2 and κ ~ 3 could produce 200 MW fusion power at 1.23 MW/m^2 average neutron wall loading at a total direct cost of ~\$643 M. This device could also be a application demonstration for non-electrical generation of fusion neutrons and could convert ~1300 kg of fission reactor waste per full power year.

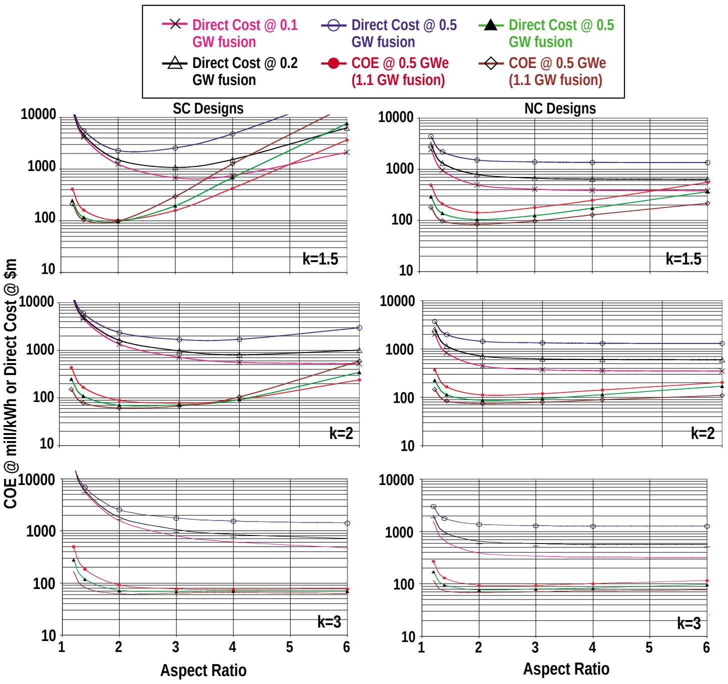

FIG. 2. SC and NC, power and test reactors (T-peak at 20 keV, f_{bs} =90% and S_n = 0.25, S_t =0.25).

Acknowledgement

The authors would like to acknowledge the continuing support and advice provided by Dr. P. Politzer and Dr. E. Reis of General Atomics. Work supported by U.S. Department of Energy under Contract No. DE-AC03-98ER54411 and General Atomics for Systems Design Code development.

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