Physics and Engineering Basis of Multi-functional Compact Tokamak Reactor Concept

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Abstract

An important milestone on the Fast Track path to Fusion Power is to demonstrate reliable commercial application of Fusion as soon as possible. Many applications of fusion, other than electricity production, have already been studied in some depth for ITER class facilities. We show that these applications might be usefully realized on a small scale, in a Multi-Functional Compact Tokamak Reactor based on a Spherical Tokamak with similar size, but higher fields and currents than the present experiments NSTX and MAST, where performance has already exceeded expectations. The small power outputs, 20-40MW, permit existing materials and technologies to be used. The analysis of the performance of the compact reactor is based on the solution of the global power balance using empirical scaling laws considering requirements for the minimum necessary fusion power (which is determined by the optimized efficiency of the blanket design), positive power gain and constraints on the wall load. In addition, ASTRA and DINA simulations have been performed for the range of the design parameters. Our studies show that increased toroidal field in a spherical tokamak can be possibly achieved by use of commercially available high temperature superconductors. This multi-functional compact reactor will also contribute to the mainstream GW Fusion power concept by providing data on burning plasma, test of diagnostics, remote handling, blanket design and operation, reactor integration etc. In this paper, the motivation for the concept as well as physics and technological challenges of the multi-functional compact reactor are discussed.

1. Introduction

The Fast Track path to Fusion Power aims to demonstrate commercial use of fusion to produce electricity in 2030-2050. ITER construction and exploitation, materials studies on IFMIF and early construction of a DEMO reactor are essential steps of this programme. For its support, it will be necessary to continue the research on present facilities and to add critical facilities for component testing and material studies. Validation for first wall materials in high power reactors is a challenge. Here we show that in a compact fusion pilot plant the wall load may be reduced to levels acceptable for existing materials by lowering the fusion output by a factor of 10 or even 100 compared to a conventional GW power fusion reactor. This can be achieved in a compact spherical tokamak (ST) with increased toroidal field and plasma current (from $0.3< B_t <0.6$T and $0.5< I_p < 1.5$MA to $3< B_t <4$T and $4< I_p < 6$MA) and a moderate auxiliary heating power from available sources. We show that recent confinement studies in STs suggest very favourable confinement scalings with the toroidal field and weaker dependence on plasma current. The extrapolation of physics parameters from current experiments to reactors with Q>1 is modest. A promising approach to increasing the toroidal field, always a challenge due to limited space for the central stack, is the use of a high temperature (20-77K) superconductors (HTS-II) for the toroidal magnet.

The value of these small fusion power outputs becomes significant when used as a fusion core to drive (a) Sub-critical fission reactor blankets, including higher Actinides, to generate electricity. (b) Destruction of difficult Fission Product wastes. (c) Efficient breeding of essential fissile fuel for Uranium or Thorium cycle reactors. (d) Efficient breeding of Tritium in all machines or dedicated versions. (e) Many aspects of fusion nuclear technology development including diagnostics, the behavior of advanced materials and physics of burning plasmas. Multiple MF-CTR units may offer benefits of reliability and ease of maintenance at low capital cost. Secondary technologies with quite accessible goals, such as remote handling, repair and maintenance, reactor management and monitoring, safety and security systems, and handling of nuclear materials, are all essential to deployment of MF-CTRs and, indeed, to fusion as a whole. Although demonstration of the electricity generation will be very beneficial, the goal of a compact pilot plant is to demonstrate production, not to sell electricity. At this stage, MF-CTR will contribute to the electricity generation by supporting the nuclear industry mainly with production of fuel. The efficiency of the fissile fuel production in a fusion breeding device can be much higher than that in a fission breeder and a complementary fusion-fission power plant can be a good solution to resolve the problem of lack of fission fuel needed for the fast expanding nuclear industry [1].

The MF-CTR fusion output increases to high power just by increasing the linear dimensions or/and the input power. The ST approach for this path to fusion power benefits from the possibility of progressing from a pilot plant to a power plant just by increasing the linear dimensions of the device without changing the technology [2]. However, the simple wall and heating technologies do not and will depend on the successful outcomes of the ITER project.

2. Motivation.

Oil decline and the need to reduce CO_2 emissions by 65-80% by 2050 require a large increase in nuclear energy even to meet current global demand. Renewables can not provide a large fraction of the energy demand; development of ecologically clean electricity production from coal will take time and significantly increase the price. Hopefully pure fusion power will resolve the problem, but even the fast track requires 30-40 years of development. Nuclear power is now much safer and new designs under research and development will consume long lived wastes and massively reduce the waste disposal problem. However, on current growth plans, the nuclear industry will soon have used up all the industrial fuel stocks and face shortage of natural uranium ore.

Comprehensive analysis of the available nuclear fuel resources made by IAEA during the last ten years [1] shows that for the (already proved to be conservative[1]) medium demand scenario of the Nuclear Power development all the reasonably assured fissile fuel resources will fail to satisfy the demand of the Industry from 2024. These studies addressed adequacy of resources; production capability limitations and potential; sensitivity to variations in supply; effect of lowering enrichment tails assays; speculative and unconventional resources; future exploration requirements; lead time between discovery and production; market price implications. By the time of the fuel commercial production shortage, the high enriched fuel from the surplus defence inventories will also be exhausted and the Nuclear Industry will face a severe shortage of the fuel. As fissile fuel production is strongly regulated by Governments and International agreements, this deficit in specific countries may happen much earlier. For example, the deficit between requirements and mined production of the fuel in the Western countries has started in 1990 and by 1998 production satisfied only about 60% of requirements; the remaining requirements were filled by secondary supply. Current global uranium production meets only 58% of demand, with the shortfall made up largely from rapidly shrinking stockpiles. During the last 15 years, the shortfall between production and requirements was made up by excess commercial inventories, uranium released from military use and other secondary sources. These are now in decline, and the shortfall will increasingly

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[1] In November 2007, there were 439 operational nuclear reactors worldwide. A further 33 reactors were under construction, 94 reactors were planned and 222 reactors were proposed ["World Nuclear Power Reactors 2006-07". Uranium Information Centre (2007)].

need to be made up by primary production. Several countries have reached their peak production of Uranium and are currently on a decline. Estimated additional resources may delay the global World deficit appearance by up to 10 years, assuming that the new mining will progress (which does not seem to happen – the short-term available military surplus resulted in the closure of many commercial mines). The development of new mines typically takes 10-15 years. The Generation IV fast nuclear fuel breeders, which in principle could resolve the fuel shortage problem, need another 30-35 years of development. As a result, these studies predicted significant increase in the fuel price by 2013, however, in practice, this increase is already seen. Although the aggressive exploration of potentially available natural recourses may delay the deficit, certainly this will be accompanied by another increase in the ore price. The overall result of this analysis claims the shortage of the fuel between 0.3 and 2 Mt by 2050.

A fusion-fission combined power plant can compliment and provide a super-fast-track solution for coming world energy crisis. The use of fusion neutrons for breeding has been proposed by Thomson & Blackman in 1947 [see ref. in 3], Andrei Sakharov 1950 [4], H Bethe 1979 [5], many other authors later and recently by P-H Rebut 2005 [6]. Detailed design studies of large systems have been performed by Moir [8], and by Russian and Chinese groups [9-11]. The result of breeding efficiency analysis has shown that an optimal fusion breeder would be a low-power (<50MW) compact fusion reactor [7 – 12]. We find that a suitably designed, fission suppressed, liquid blanket of Depleted Uranium fluoride salts around the 25-30MW fusion reactor can produce enough fuel each year to make up the fissile fuel requirements of a 600MW-thermal fission reactor [12], showing these small devices to be practical contributors. The radioactive waste content of the breeding liquid will always be far below that of spent reactor fuel and the separation of fuel will be correspondingly cheaper and simpler than current reprocessing facilities to make normal mixed oxide reactor fuel. The low concentrations, <0.5%, make the liquids safe to handle with no possibility of coming anywhere close to criticality. Similarly, the tiny amounts of fission product waste will also be separated out and could even be burned up in small blanket packages returned to the fusion blanket. At these power levels, neutron damage to the first wall of the fusion reactor must be kept as low as possible. It will be shown below that in the device proposed here the neutron flux averages 0.25-0.3MW/m^2 and gives a reasonable life to the first wall materials. The actual values will be double this on the main circumference but lower near the poles. A compact low power reactor needs less scarce materials, shorter construction time, less heating power, less demand of industrial and manpower capacity and can be built by one country. Surely, much bigger devices could also be potentially used for breeding: ITER [6], IGNITOR or TFTR [7], however, these require advanced materials and engineering which will only be available after ITER is complete.

3. Engineering and Physics basis for Compact Reactor.

For electricity generation, steady-state operations will be desirable. A high duty cycle with >70% availability could be sufficient for other applications, limited only by power supply and maintenance needs. Heavy maintenance, like removal of damaged or worn-out components (central post, divertors, etc.), should be on a long cycle. The thermal inertia of the blanket, together with material stresses will also constrain the pulse frequency and duration. The blanket must be kept operating steadily to avoid lengthy warming-up and cooling-down and all timescales should be optimized for efficient blanket operation. These timescales will determine the shortest plasma pulse duration for the effective plant operations. The tritium reproduction may be also an important requirement and should be considered; however, a fission-fusion plant may be self-sufficient. Finally, the whole plant cycle will be evaluated to make the tokamak design consistent with all operations (start-up, current and power ramp, steady state and current drive (CD) requirements, wall and divertor loads, impurities/ash removal timescale, pulse shut-down). The blanket design should incorporate all available experience from the previous studies in Fusion including the ITER blanket design, and all knowledge and experience from the fission blanket developments.

The full physics of plasma fields, currents, stability, transport and fusion performance has been evaluated around the selected design parameters by various and separate methods as shown below. In-depth studies of neutral beam and ECRH/EBW heating, plasma profile control, the balance of bootstrap and driven currents, refueling, and plasma exhaust are in progress. Preliminary studies have developed a consistent outline design for a multi-functional compact tokamak reactor which will produce enough neutrons to satisfy the requirements for efficient blanket operation. These studies show that a spherical tokamak with increased toroidal field is the most efficient candidate for all the proposed applications.

Figure-of-merit system analysis [15] of the performance of the multi-functional compact tokamak reactor has been performed based on the solution of the global power balance equation with the convection and conduction losses modeled by empirical scaling laws (ITER scaling law in particular). The analysis considers requirements for the minimum necessary fusion power, positive power gain and constraints on the average wall load. Stability issues related to the toroidal beta limit, safety factor and density limit are taken into account. The plasma model includes geometrical aspects, profiles and impurity effects, neoclassical effects, and stability constraints. Fig.1 shows result of this analysis for R=1.2m, a=0.75m, κ=2.75, δ=0.5, I_p=5MA, B_t=3.5T and H=1.8 (IPB98). The analysis shows weak dependence of the Q = P_{fus}/P_{aux} on density and input power.

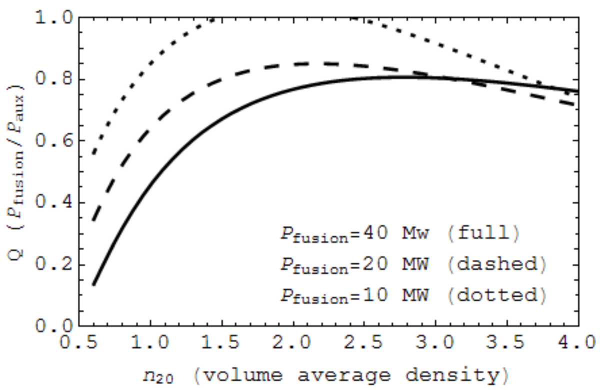


Fig.1. Results of figure-of-merit analysis for R=1.2m, a=0.75m, κ=2.75, δ=0.5, I_p=5MA, B_t=3.5T and H=1.8 (IPB98)

The calculations were carried out with a zero dimensional numerical model, which allows for proper description of the magnetic configuration in terms of major radius R, minor radius a, ellipticity κ and triangularity δ, and full fusion integral [2,16], taking the radial profile functions of density and electron and ion temperatures in form $x(r) = x_0(1-(r/a)^2)^{\alpha}$. We have explored broad profiles with α~0.25, following [2], and somewhat less broad ones at a~0.75 which puts more than half of the fusion power within half of the plasma radius, or a quarter of the volume. The all important function in the model is the energy confinement time, τ_E. This is derived from the internationally agreed IPB-98y,2 scaling.

IPB98y,2: $\tau_E = 0.0562\, Ip^{0.93}\, Bt^{0.15}\, R^{1.97}\, (a/R)^{0.58}\, M^{0.19}\, ne^{0.41}\, k^{0.78}\, P\_in^{-0.69}$

Experiments on STs at Culham (MAST) [14] and Princeton (NSTX) [13] show that the H-factor can be as high as 1.8 for the chosen design point parameters and indeed suggest a revised form of the τ_E scaling:

Kaye-Valovic: $\tau_E = Cp\, I_p^{0.51\text{-}0.59}\, Bt^{1.08\text{-}1.6}\, R^{1.97}\, (a/R)^{0.58}\, M^{0.19}\, n_e^{0.0\text{-}0.44}\, k^{0.78}\, P_{in}^{-(0.27\text{-}0.73)}$

The coefficient Cp has been chosen by benchmarking these scalings with the experimental data. Fig.2 presents results of the system code for P_{fus}=25MW, R=1.2m, a=0.75m, κ=3, δ=0.4 and H=1.5 (IPB98). The chosen design point is shown with a shaded area. It is shown that Q >1 for P_{fus}~ 25MW could be achieved in a compact ST with high field.

Results of both figure-of-merit and the system code analysis have been benchmarked with the ASTRA-ESC transport modelling showing good agreement for the design point. In these simulations, the ion transport has been chosen to be neoclassical (which is in agreement with the experimental observations [17]), electron thermoconductivity profile – broad parabola, scaled to get H (IPB98) = 1 - 1.8, density – prescribed, with profile and $n(0)/n_G$ scans, NBH power deposition – prescribed and checked for some cases with NUBEAM, equilibrium – ESC. Results of these simulations are shown in Fig. 3-4.

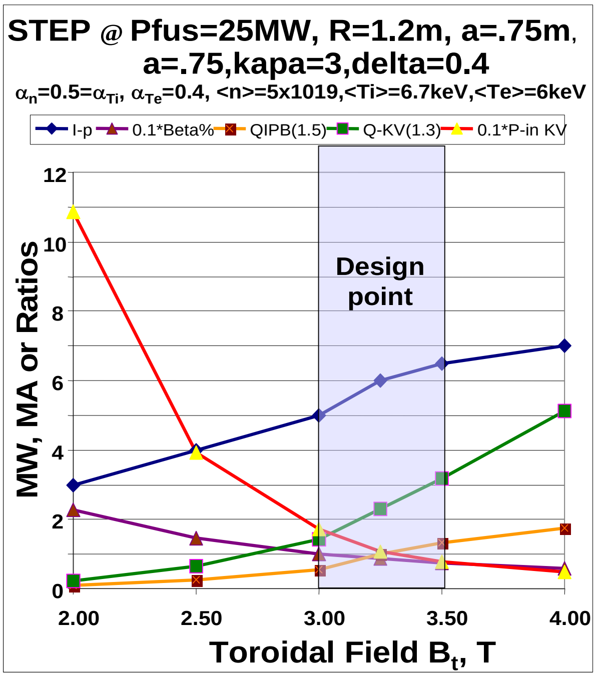


Fig.2. Results of the system code analysis. The design point for P_{fus}=25MW, R=1.2m, a=0.75m, κ=3, δ=0.4 and H=1.5 (IPB98) is shown with a shaded area.

Fig 3 presents dependence of the fusion power and Q on the density for different input NB power. The Greenwald limit n_G is shown by the vertical line. H(IPB98) = 1.6 has been chosen for this case, R=1.2m, a=0.75m, κ=3, δ=0.5, parabolic density profile, I_{pl}=5MA, B_t=3.5T. It is clearly seen that Q ~ 1 and P_{fus} needed for successful operation of the fusion-fission combined power plant can be achieved. Another important conclusion is the low dependence of the Q-value on the input and output power which confirms the possibility of operations at a low power.

The optimal density strongly depends on the heating power for the chosen deposition profile. Full simulations of the power deposition will be needed to confirm this

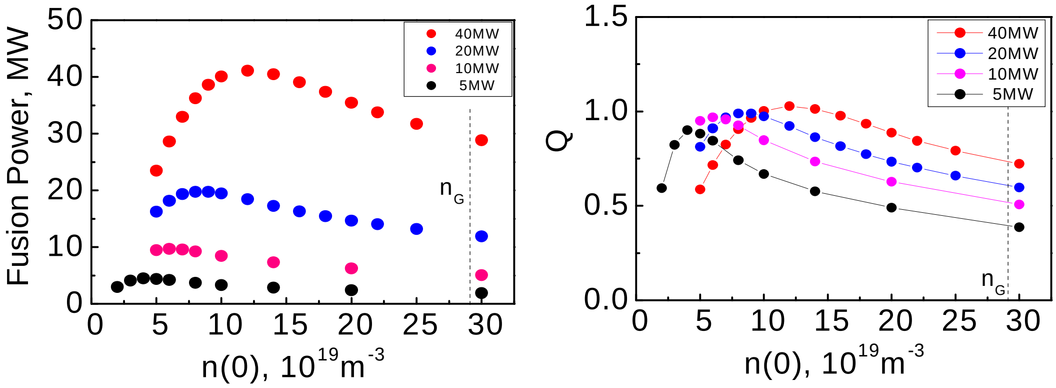
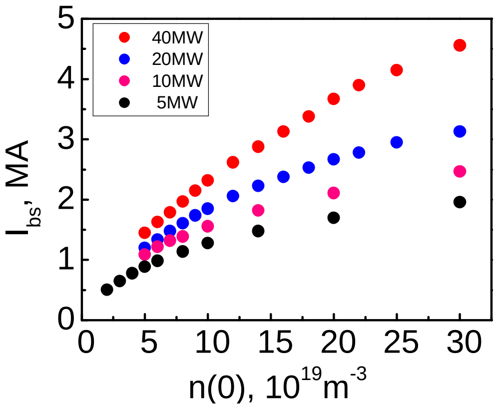


Fig 3. Dependence of the fusion power and Q on the density for different input NB power.



dependence. However, our simulations show strong dependence of the bootstrap current on density, Fig.4.

The aim for a high bootstrap faction is dictated by the need to reduce as much as possible the value of the auxiliary power for the

Fig.4. Bootstrap current I_{bs} dependence on the central density for different NB power. Plasma and device parameters as in Fig.3.

current drive. It is seen that to increase the bootstrap current from 3MA to 4.7MA we need an extra 20MW of heating power. It might be possible to achieve high bootstrap fraction with the neutral beam (NB) current drive at lower power and full current drive simulations should be performed to optimise this.

As a high bootstrap fraction is also needed for the current ramp-up (due to limited or absent central solenoid), simulations with the DINA/ASTRA 1D evolution code have been performed for the current ramp-up phase. Conditions have been optimised for high f_{bs}, k = 3, using flat temperature profiles that have been achieved on MAST. Lower values for the target plasma current (3MA) and toroidal field (2T), P_{NBI}=10MW, $T_{e,i}(0)$ ~ 3.0keV, $n_e(0)$ ~ $1.7\times10^{20}m^{-3}$ have been chosen for these simulations to bring parameters close to these achieved on MAST (see below). Fig. 5 presents evolution of the plasma current (left), I_{bs} (middle) and τ_E (right). The analysis shows that the bootstrap overdrive can be achieved even for these reduced parameters.

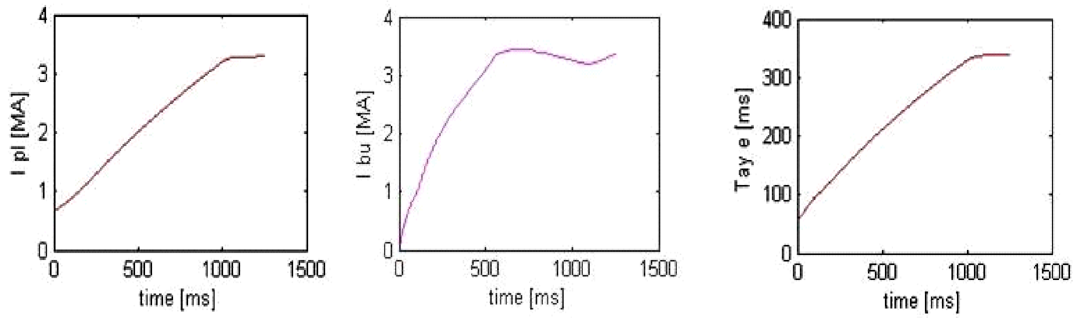


Fig.5. Evolution of the plasma current (left), I_{bs}(middle) and τ_E (right) , DINA/ASTRA

In these simulations, a new formula for the bootstrap fraction has been used:

$$f_{bs}=I_{bs}/I_p=0.234\varepsilon^{1/2}\beta_P c_p^{\,0.80}$$

where $\varepsilon=a/R_0$ inverse aspect ratio, $\beta_P=\int p\, dV/\int B_P^2/2\mu_0\, dV$, $c_p=p_0V_p/\int p\, dV$, ratio between central and volume averaged pressure. This gives a good estimate for f_{bs} in STs, as shown in Fig.6, and was derived for equilibria with non-hollow pressure profiles.

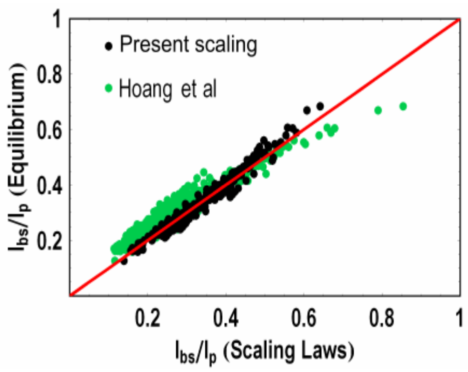


Fig.6. Comparison of the new scaling (black) with the SCENE modelling

It has been shown [13, 18, 19] that the L-mode confinement in ST approaches that in H-mode, so an ST reactor can operate in L-mode (without problems associated with high wall and divertor loads from ELMs), with only slightly reduced output. Fig. 7 presents results of L-mode studies on MAST [19]. With a moderate 2MW NB heating and I_p ~ 1MA, $T_{i,e}$ in the range of 2-3keV have been achieved with broad profiles. These results provide optimism for the prediction of the MF-CTR performance, which requires only 2-3 times higher temperatures at 5 times higher I_p and 6 times higher B_t.

The weak plasma current dependence of confinement in ST allows a reduction in I_p in MF-CTR which will give an increase in the non-inductive current fraction. As a result, most of α-particles will be lost, mainly on the first orbit, Fig. 8. Here the asymptotic expression of first-orbit loss model has been used, with lines presenting adjustment of the pressure profile, dots simulation for A=1.6, R=0.8m, κ=2.5, δ=0.4, B=2.8T [20]. The simulation includes collisionless prompt and TF ripple losses, and relatively close walls. Results of modeling of different ST configurations ($1.6 \le A \le 2.0$, $2.0 \le \kappa \le 3.0$) show low sensitivity to elongation and aspect ratio. These high losses at $I_p < 4$MA help to avoid high peaking factor of losses due to MHD, reduce/avoid ash accumulation and provides the possibility for additional first wall heating from α-particles which may be utilised as a non-negligible contribution to efficiency in a low-power reactor. As there will be no significant α-heating, the reactor will be mainly beam-driven, which may be beneficial for regime control. However, irradiation damage from α-particle and fast ion losses should be taken into account.

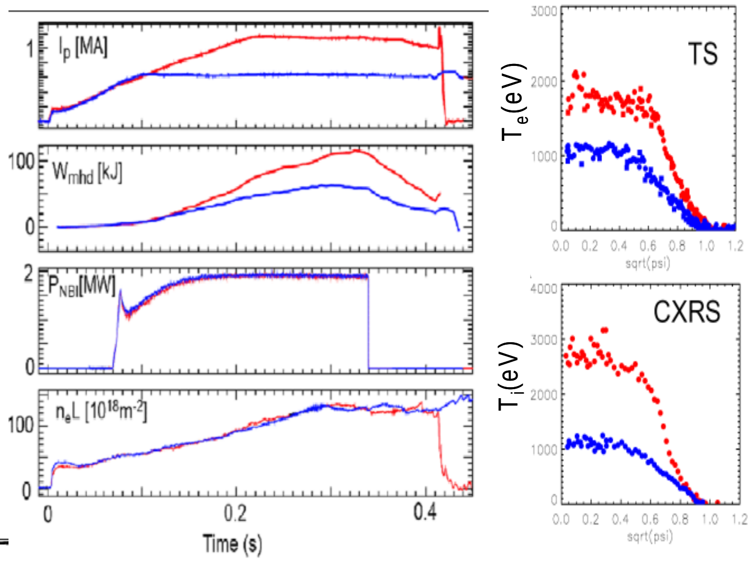


Fig.7. L-mode discharges on MAST show high $T_{i,e}$.

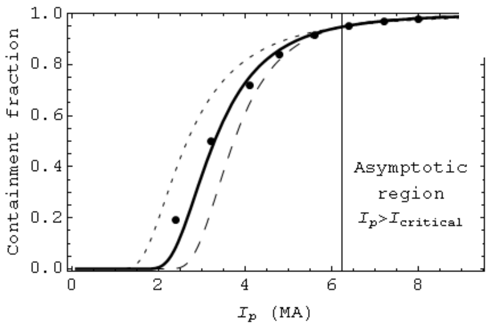


Fig.8. α-particles losses vs I_p in ST

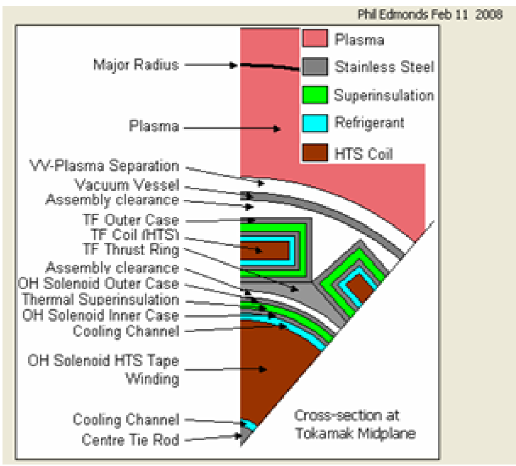


Fig.9. Central post design with HTS TF and central solenoid

One of the main design problems previously associated with the use of the ST concept for a pure-fusion reactor is high power dissipation in the toroidal magnet, typically reaching 100-500MW. This leads to GW level of the required fusion power, very high wall load, large size, extreme heating and CD requirements, and high tritium consumption. Although pulsed operations may soften the power dissipation issue, the only way to resolve it for a long pulse or steady state reactor is to use superconducting coils. The possible use of high temperature superconductors (HTS) in a compact ST has been assessed using a system code, Fig 9, with promising results. The experience gained from the existing commercial and fusion applications of the HTS is used in the HTS tokamak magnet design. In parallel, a test HTS TF coil will be designed, manufactured and tested.

The HTS allows design of a much more compact high-field TF magnet. The compactness of HTS provides a solution for the neutron shielding of the central post. For P_{fus}=1MW/m^2(5x10^{17}n/m^2), the MIT cryoshield concept [21] will reduce the neutron load to 1.6x10^{17}n/m^2 with a 12cm W shield and down to 3.3x10^{16}n/m^2 with a 24cm W shield, providing up to 15 times reduction.

The multi-functional blanket design is based on the He/LiPb dual-cooled triple-layer design [9, 10, 22, 23]. This design features high energy multiplication with emphasis on circulating particle or pebble bed fuel configuration considering the geometry of the ST and the frequency of fuel discharge and reload. Other designs [6,8, 11] can also be considered.

4. Conclusions.

We have shown that a compact spherical tokamak with high field is a plausible candidate for a low-power fusion reactor (P_{fus} < 50MW, wall load < 0.5 MW/m^2). The design of such a reactor is based on known physics and technologies, and uses commercially available first wall materials and (optionally) high temperature superconductors. A fleet of such fusion reactors can supply the nuclear industry with sufficient amount of fissile fuel, and also clean waste from nuclear reactors.

This work is supported by strong international collaboration. It is foreseen that different compact multi-functional reactors for commercial exploitation of fusion power and reactor prototypes aimed at deeper studies of the physics and technology will be designed and constructed in parallel. In addition, several smaller devices for technology and material tests should be considered (e.g. a small tokamak with an HTS magnet).

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