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A component test facility based on the spherical tokamak

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Abstract

Recent experiments (Synakowski *et al* 2004 *Nucl. Fusion* **43** 1648, Lloyd *et al* 2004 *Plasma Phys. Control. Fusion* **46** B477) on the Spherical Tokamak (or Spherical Torus, ST) (Peng 2000 *Phys. Plasmas* **7** 1681) have discovered robust plasma conditions, easing shaping, stability limits, energy confinement, self-driven current and sustainment. This progress has encouraged an update of the plasma conditions and engineering of a Component Test Facility (CTF), (Cheng 1998 *Fusion Eng. Des.* **38** 219) which is a very valuable step in the development of practical fusion energy. The testing conditions in a CTF are characterized by high fusion neutron fluxes $\Gamma_n \approx 8.8 \times 10^{13}\,\mathrm{n\,s^{-1}\,cm^{-2}}$ ('wall loading' $W_L \approx 2\,\mathrm{MW\,m^{-2}}$), over size-scale $>10^5\,\mathrm{cm^2}$ and depth-scale $>50\,\mathrm{cm}$, delivering >3 accumulated displacement per atom per year ('neutron fluence' $> 0.3\,\mathrm{MW\,yr^{-1}\,m^{-2}}$) (Abdou *et al* 1999 *Fusion Technol.* **29** 1). Such conditions are estimated to be achievable in a CTF with $R_0 = 1.2\,\mathrm{m}$, $A = 1.5$, elongation ~ 3, $I_p \sim 12\,\mathrm{MA}$, $B_T \sim 2.5\,\mathrm{T}$, producing a driven fusion burn using 47 MW of combined neutral beam and RF heating power. A design concept that allows straight-line access via remote handling to all activated fusion core components is developed and presented. The ST CTF will test the lifetime of single-turn, copper alloy centre leg for the toroidal field coil without an induction solenoid and neutron shielding and require physics data on solenoid-free plasma current initiation, ramp-up to and sustainment at multiple megaampere level. A systems code that combines the key required plasma and engineering science conditions of CTF has been prepared and utilized as part of this study. The results show high potential for a family of relatively low cost CTF devices to suit a range of fusion engineering and technology test missions.

1. Introduction

Successful development of practical fusion energy will require research and development that combine fundamental and applied science. A fusion energy Component Test Facility (CTF), aimed at advancing the required fusion engineering and technology, will necessarily entail similarly combined efforts in fusion plasma science. This view is consistent with the extensive experience of fission power development during 1950–60s, when 45 small fission test facilities were built and operated at the site of the Idaho National Laboratory [5], which contributed to the material science, engineering, safety and environmental technology basis for commercial fission power.

A recent plan issued by the USDOE Office of Science [6] identified a broad strategic goal to '*develop the new materials, components and technologies necessary to make fusion energy a reality*' for the US Fusion Energy Sciences Program. In this plan, a CTF would be built to succeed the International Tokamak Experimental Reactor (ITER) [7] construction to address this goal before Demo construction. The fusion engineering and technology conditions to be produced by the CTF to achieve its mission are summarized in section 2. Data from CTF will determine how the full and steady state fusion conditions affect plasma chamber materials and components, and limit their operating life. This will in turn enable improvements in the engineering and technology knowledge base to support a decision to build a demonstration power plant (Demo) that aims to produce net electrical output. A CTF will therefore provide, beyond the levels planned for ITER, the testing conditions in high material dpa and operational duty factor very valuable in establishing the engineering basis for Demo.

To create a cost-effective CTF, one that is much smaller in size and power than a Demo or ITER, full advantage must be taken of the progress made in determining '*the most promising approaches and configurations to confining hot plasmas for practical fusion energy systems,*' which is also a strategic goal of the Fusion Energy Sciences Program [6]. Aimed at this goal are the Innovative Confinement Concept experiments in a number of confinement configurations. Among these, the physics of the spherical tokamak (ST) [8] plasma made strong progress due to the rapid deployment and experimentation in recent years of major ST facilities such as NSTX [1] and MAST [9]. It has thus become timely to update an earlier concept [10] of the volume neutron source plasma for the CTF. To ensure high duty factor operation, the CTF plasmas must operate in a physics regime with substantial margins to the anticipated limits in stability, confinement, sustainment and boundary interactions. The most recent results from ST research strengthened the basis for the CTF concept and are summarized in section 3.

Based on the ST physics progress, a relatively modest size CTF, as previously surmised, [10] can be retained. The low aspect ratio $A(=$ major radius/minor radius $= R_0/a)$ of the ST with demountable toroidal field (TF) coils further permits modularization of the chamber and the TF coil systems, allowing direct access for remote handling to achieve the required neutron fluence and duty factor. The engineering design features to achieve this with an ST CTF are presented in section 4.

The appropriate plasma and engineering design conditions of the CTF are modelled in approximation in a systems optimization algorithm to survey the range of acceptable designs. A design with $R_0 = 1.2\,\mathrm{m}$, delivering the baseline performance of fusion neutron wall flux $\Gamma_n = 4.4$–$8.8 \times 10^{13}\,\mathrm{n\,s^{-1}\,cm^{-2}}$ ($W_L = 1$–$2\,\mathrm{MW\,m^{-2}}$), is set forth as a good trade-off between size, performance, cost and risk. If the performance is pushed towards the physics limits of the advanced regimes envisioned for a power plant [11, 12], this CTF is estimated to deliver $\Gamma_n = 17.6\times10^{13}\,\mathrm{n\,s^{-1}\,cm^{-2}}$ ($W_L = 4\,\mathrm{MW\,m^{-2}}$), which is envisioned for the Demo. However, this would also require that all CTF chamber systems and facilities are developed to deliver and handle this level of performance. The plasma and engineering parameter space of the compact ST CTF will be presented in section 5.

An updated understanding of the CTF presents a new opportunity to identify, by comparing the desirable plasma conditions of the CTF with the current research using the ST, the major scientific issues of the CTF plasma. The CTF scientific bases are identified in section 6 in reference to the latest progress in ST research [14, 15]. Of note is the critical importance assigned to the scientific basis for generating poloidal magnetic flux in the plasma without induction from a central solenoid magnet and handling high average plasma heat flux from a compact device of high power density.

It is appropriate to assume that ITER [7] will demonstrate in the 2020s the physics of self-heated burning plasmas, beyond the level required by the driven burning plasma in CTF. It is further appropriate to utilize the ITER chamber components and engineering systems as starting approaches to heat, fuel, pump and confine the driven steady state burning plasmas in CTF, where the steady state baseline flux of $\Gamma_n = 4.4$–$8.8 \times 10^{13}\,\mathrm{n\,s^{-1}\,cm^{-2}}$ would be 1–2 times the ITER level. The requirements in fusion engineering and technology for the baseline CTF operation and control, including the single-turn normal conducting TF coil centre leg, will be covered in section 6.

The paper closes with a conclusion in section 7 of the key results of the study and a discussion of the broader scientific and engineering implications of CTF.

2. Component testing mission and conditions

The CTF is a facility for establishing the integrated fusion engineering and technology base for the chamber systems to produce practical fusion power. The chamber systems for magnetic fusion have been characterized in a number of fusion reactor concept studies [16]. A comprehensive assessment of the required knowledge base of the fusion chamber systems was reported by Abdou *et al* [4].

Many complex phenomena will occur in fusion chamber systems, within and at the interfaces among coolants, tritium breeders, neutron multipliers, structural materials, conducting shells, insulators and tritium permeation barriers. These phenomena include MHD reorganization and damping of turbulent flow structures affecting the transport phenomena in conducting coolants, neutron-induced ballistic mixing of nano-scale features in structural materials, deformation and fracture dynamics in materials and tritium desorption and recombination phenomena on the surface of breeding ceramics. Progress in understanding of these phenomena requires efforts involving many disciplines including large-scale computing modelling, in concert with the progress in developing a fusion energy knowledge base derivable from the safe and successful operation of ITER. The phenomena that affect tritium self-sufficiency, in particular, involve all critical aspects of the fusion system. Establishing the knowledge base of the D–T cycle therefore requires parallel and highly interactive research in plasma physics, plasma control technologies, plasma chamber systems, materials science, safety and systems analysis. The CTF will provide the 'full conditions' to test and develop such a knowledge base for Demo.

The key ingredients of the full conditions have been identified for CTF [4], and can be summarized in table 1, in comparison with the ITER design and those anticipated for a full remotely maintainable Demo [11, 12] that assumes a 2-year maintenance cycle (see, section 4).

Table 1. Key fusion engineering science conditions to be provided by CTF, relative to ITER design and a Demo concept assuming a 2-year maintenance schedule.

Condition	ITER	CTF	Demo
Fusion neutron flux through wall surface, Γ_n (10^{13} n s^{-1} cm^{-2})	~3.4	4.4–8.8	~18
Fusion neutron heat flux through wall surface, W_L (MW m^{-2})	~0.8	1–2	~4
Depth of energetic neutron-material interactions (m)	~0.5	~0.5	~0.5
Transverse spatial scale of energetic neutron-material interactions (m)	~10	~5	~10
Total chamber systems displacement per atom, dpa	~3	>60	60–200
Dpa per full-flux-year, D	~6	>10	~40
Duration of sustained neutron interactions (s)	$\sim 10^3$	$> 10^{6-7}$	$\sim 10^{7-8}$
Tritium self-sufficiency goal (%)	~TBD	80–100	>100
Duty factor, F_D(%)	2.5	30	75

It is seen that the CTF is required by its mission to approach the Demo chamber conditions in all aspects except in fusion neutron and heat fluxes. There is therefore a premium value to enhance the CTF conditions towards those of the Demo by increasing these fluxes. ITER provides adequate conditions in spatial scales of materials depth and width of interest, falls short of the Demo neutron and heat fluxes; but falls far short in dpa, duration and (except in the test blanket modules) tritium self-sufficiency. A successful ITER construction will therefore provide an incentive to deploy CTF on the path towards the Demo.

To support a timely establishment of the fusion engineering and technology base for the Demo, the CTF would do well to complete its mission in a time scale T of 10 years, during the Demo design and construction to accrue the most benefits. To reach the lifetime dpa, the required duty factor, F_D, would be

$$F_D = \frac{\text{dpa}}{D \times T}.$$

This indicates an $F_D > 30\%$ for a CTF operated at a neutron flux of 2 MW m^{-2}. For ITER, $F_D = 2.5\%$ to achieve 3 dpa in 20 years. This would, however, be more than an order of magnitude progress beyond the accumulated duty factor of major magnetic fusion experiments to date and therefore a reasonable step towards the CTF conditions.

In this assessment we consider tests of tritium breeding and recovery to be an integral part of the CTF mission, as >100% tritium reproduction will be required for the Demo operation. A tritium self-sufficiency goal of 80–100% during the lifetime of CTF is further considered appropriate to fit the limited available tritium supply anticipated in the next 2–3 decades. Specifically, it is estimated [13] that the world tritium supply is likely to peak at 27 kg, while ITER is anticipated to utilize about 11 kg through its lifetime. An amount of 10–15 kg would remain available for use in a CTF or a Demo, the latter of which would burn tritium at a rate of 2.7 kg per week to produce 2.5 GW fusion power. It is therefore very valuable to test and develop tritium breeding and recovery technology in a device with low fusion power of the order of 100 MW in fusion power in such as the CTF, before deployment in the Demo.

3. Recent progress in ST plasma physics base

To achieve high duty factor operation in CTF, the necessary plasma conditions must also be reliably produced in steady state. The assumed plasma conditions must therefore be sufficiently removed from known limits of plasma stability and confinement. For this discussion, we choose a 'baseline' CTF that has $R_0/a = 1.2$ m 0.8 m, $\kappa = 3.2$, $I_p = 8.4$–12.2 MA, $I_{TF} = 15.3$ MA, $B_{T0} = 2.5$ T, $n_e = 0.69$–1.0×10^{20} m^{-3}, $\beta_T = 14$–30%, $\beta_N = 3.0$–4.9, $P_{AUX} = 38$–47 MW,

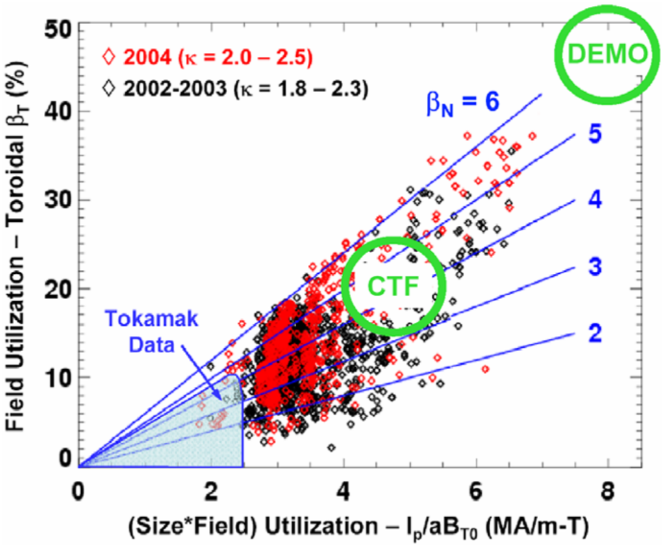

Figure 1. Toroidal betas (β_{T}) as a function of the normalized plasma current ($I_{\mathrm{N}} = I_{\mathrm{p}}/aB_{\mathrm{T0}}$) obtained so far on the NSTX, relative to the regimes of interest to CTF, Demo and the normal aspect ratio tokamak.

$E_{\mathrm{NB}} = 110$–$160\,\mathrm{kV}$, $P_{\mathrm{DT}} = 72$–$144\,\mathrm{MW}$ to produce $\Gamma_n = 4.4$–$8.8 \times 10^{13}\,\mathrm{n\,s^{-1}\,cm^{-2}}$ at the outboard mid-plane wall. More detail of how these parameters are determined will be provided in section 5.

3.1. Pressure and current limits

Recent studies of the global plasma stability beta limits in the ST [17,18] and comparisons with the recent experimental results [1,14,15] have shed additional light on how a substantial range of plasma parameters of interest to the CTF can be maintained while staying substantially below these limits. Figure 1 presents a summary of the toroidal beta values ($\beta_{\mathrm{T}} \propto p/B_{\mathrm{T0}}^2$, where $p =$ average plasmas pressure and $B_{\mathrm{T0}} =$ applied TF at the plasma major radius R_0) achieved so far on the NSTX without active feedback control. Also indicated are the parameter regimes of interest to the CTF under consideration (section 5) and the ST Demo [12]. Table 2 provides additional parameters achieved in relatively long ($\tau > \tau_{\mathrm{skin}}$) and short ($\tau < \tau_{\mathrm{skin}}$) durations. It is seen that a sizeable parameter space in I_{N}, q_{cyl}, β_{T} and β_{N} is identified for the longer pulse plasmas already produced on the NSTX. This parameter space resides significantly within that, which has been reached for shorter durations.

The magnitude of the normalized current I_{N} in ST is increased substantially due to strong plasma shaping including elongation κ and triangularity δ of the plasma cross section, as well as the strong magnetic field curvature associated with the very low aspect ratio [8]. These combine to increase the plasma safety factor q_{cyl} and enhance stability against current driven instabilities at high plasma current. Approximately, in MKS units, the product of $I_{\mathrm{N}}q_{\mathrm{cyl}}$ increases strongly with the inverse aspect ratio ($\varepsilon = a/R_0$) and κ:

$$I_{\mathrm{N}}q_{\mathrm{cyl}} = \frac{\pi\varepsilon(1+\kappa^2)}{10^6\mu_0}.$$

It should be noted that for the large values of I_{N} in the NSTX, data collected in figure 1 are characterized by relatively high q_{cyl} (>2) and relatively low plasma internal inductance [17].

Table 2. Estimates of simultaneously achieved conditions of NBI driven H-mode plasmas for relatively long ($\tau > \tau_{\rm skin}$) and short ($\tau < \tau_{\rm skin}$) pulses on the NSTX.

Plasma parameters	$\tau > \tau_{\rm skin}$	$\tau < \tau_{\rm skin}$
Major radius, R_0(m)	0.85	
Plasma aspect ratio, A	1.4	
Plasma elongation, κ	2.0–2.3	
Applied TF, $B_{\rm T0}$ (T)	0.45	
Plasma current, $I_{\rm N} = I_{\rm p}/aB_{\rm T0}$ (MA m T)$^{-1}$	$\leqslant 4.4$	$\leqslant 6.0$
Safety factor, $q_{\rm cyl}$	$\geqslant 2.5$	$\geqslant 2$
Average toroidal beta, $\beta_{\rm T}$	$\leqslant 23$	$\leqslant 40$
Normalized beta, $\beta_{\rm N}$	$\leqslant 5.6$	$\leqslant 7.0$
Normalized density, $n_{\rm e}/n_{\rm GW}$	$\leqslant 0.7$	$\leqslant 0.9$
Global H-factor, H_{98}	$\leqslant 1.3$	$\leqslant 1.5$
Electron energy confinement H-factor, $H_{98\rm e}$	$\leqslant 0.7$	$\leqslant 1.0$
Ion energy confinement H-factor, $H_{98\rm i}$	$\leqslant 4.0$	$\leqslant 2.0$
Ion neoclassical energy confinement factor, $H_{\rm NCi}$	$\leqslant 0.7$	$\leqslant 0.3$
Non-inductive current fraction, $(I_{\nabla p} + I_{\rm CD})/I_{\rm p}$	$\leqslant 0.6$	N/A

The normalized beta ($\beta_{\rm N} = \beta_{\rm T} a B_{\rm T0}/I_{\rm p}$) measures the plasma stability against pressure driven instabilities, which was first noted by Sykes [19] and Troyon [20] based on extensive stability computations. Relative to the normal aspect ratio tokamak data, $\beta_{\rm N}$ in ST shows a substantial increase in part due to contributions from a strong poloidal magnetic field, which is comparable to the toroidal magnetic field [8,17]. This together with the large $I_{\rm N}$ enables the high $\beta_{\rm T}$.

In practical terms, the data in figure 1 and table 2 indicate high utilization of the applied magnetic field and plasma size, which translates to cost and size-effective ST.

3.2. *Energy confinement*

With neutral beam injection (NBI) alone, relatively long-pulse plasmas have been routinely obtained that have properties of interest to the CTF. The parameters and the temperature, density and rotation profiles of an example of such plasmas are shown in figure 2 and also indicated in table 2.

This type of NSTX plasma is characterized by $T_{\rm i} > T_{\rm e}$ in the plasma core, relatively flat density profiles and hollow impurity $Z_{\rm eff}$ profiles (largely from C-VI in this case). Analysis using the TRANSP code [23] indicated [24] that, while the electron thermal diffusivity is large ($\chi_{\rm e} \approx 10\,{\rm m^2\,s^{-1}}$), the ion thermal diffusivity can be at the neoclassical level ($\chi_{\rm i} \sim \chi_{\rm NC} = 1$–$2\,{\rm m^2\,s^{-1}}$) in a substantial region ($\Delta R \geqslant 10\,{\rm cm}$) extending to $R \sim 140\,{\rm cm}$, coincident with steep gradients in $T_{\rm i}$ and V_ϕ. These are similar to the description of an ion internal transport barrier (iITB) [25], the verification of which is in progress and is expected to have important implications for the plasma conditions of future ST devices including the CTF. The values of β_T for such plasmas in the range of 16–23% have been obtained, in 2002–2005, for stationary plasma durations substantially larger than $\tau_{\rm skin}$.

The resulting plasma thermal energy confinement times τ_E, which are dominated by the electron energy loss, are still favourable compared with the standard ITER H-mode scaling [26] given below:

$$\tau_E^{98[y,2]} = [0.0562 M^{0.19} I_{\rm p}^{0.93} R_0^{1.97} B_{\rm T}^{0.15} \varepsilon^{0.58} \kappa a^{0.78} \bar{n}_{\rm e19}^{0.41} / P_{\rm Tot}^{0.69}].$$

Here M is the average plasma ion mass and $P_{\rm Tot}$ the total thermal plasma loss power. Results of analysis of a number of such H-mode plasmas indicate that H-factors up to 1.3 can be obtained [28].

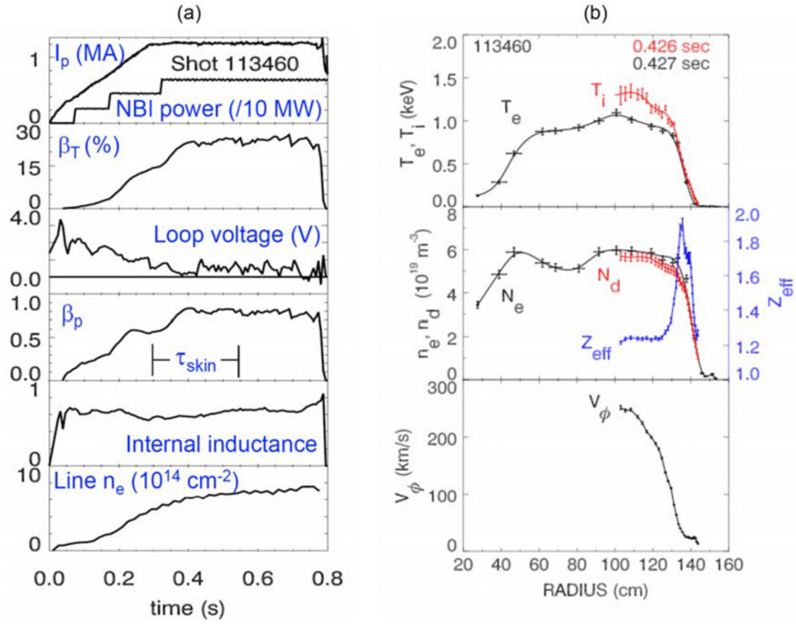

Figure 2. (*a*) Parameters of a relatively long-pulse H-mode plasma driven by deuterium NBI at 6 MW and 90 kV and (*b*) electron and ion temperatures and densities (T_e, T_i, n_e, n_D), plasma effective charge Z_{eff} and C-VI ion toroidal rotation (V_ϕ) profiles measured by charge–exchange recombination spectroscopy [22] and laser Thomson scattering [23].

Recent research on MAST [27] suggested that sawtooth-free L-mode plasmas with a weak central magnetic shear can also exhibit improved core confinement to be of interest to CTF operation. Such plasmas are characterized by medium average densities ($n_e/n_{GW} \sim 0.5$), $T_i/T_e > 1$, hollow J profile for $\rho < 0.4$, $\chi_i \approx 4\chi_{NC}$ for $\rho = 0.4$–0.6 where $E \times B$ shearing rate $\omega_{SE} > \gamma_m$ the ITG microinstability growth rate and $\chi_e = 2$–$4\ \chi_i$. An expected advantage of the L-mode over the H-mode plasma would be the relatively increased plasma edge turbulence and dispersed heat flux on the divertor plate, which is expected to be high in CTF.

Since χ_i can be substantially different from χ_e, it is necessary to separate the energy confinement times of the electrons and ions. By using the measured profiles (figure 2), accounting for energy transfer between electrons and ions, and subtracting the stored energy of the NBI ions, we arrive at an approximate partition of the energy loss channels in the plasma core, which is also included in table 2. The global energy confinement time τ_E and the separate energy confinement times, τ_{Ee} and τ_{Ei}, are related approximately by

$$\frac{W_i + W_e}{\tau_E} = \frac{W_i}{\tau_{Ei}} + \frac{W_e}{\tau_{Ee}},$$

where W_i and W_e represent the ion and electron stored thermal energies, respectively. The separate H-factors relative to $\tau_E^{98[y,2]}$ and τ_{NCi} therefore provide an approximate basis for making projections to CTF.

3.3. Plasma pressure gradient and externally driven currents

An important component of sustained current arises from the plasma pressure gradient. The 'bootstrap' current I_{BS} [29] has been estimated to be substantial on the NSTX owing to the relatively high β_N and q_{cyl}. Figure 3 shows the estimated bootstrap current fraction

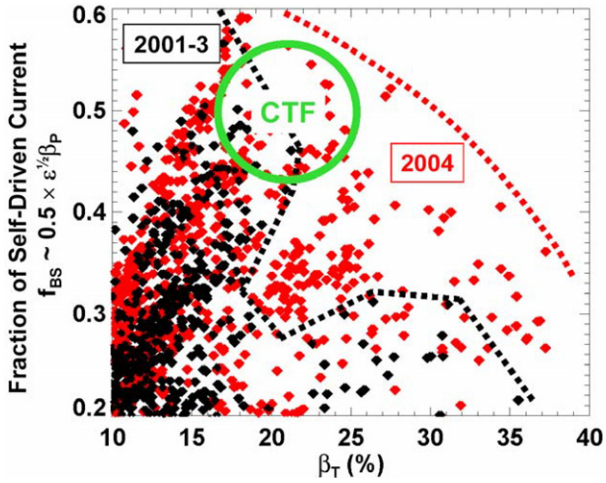

Figure 3. Progress of bootstrap current fraction versus β_T on the NSTX for 2001–2003 and 2004.

$f_{BS} = I_{BS}/I_p$ as a function of β_T, measured so far in the NSTX. Because of the high β_N and outboard B_p/B_T ($\sim$1) in an ST, the contributions from the toroidal component of the equilibrium diamagnetic current (I_{GP}) and the Pfirsch–Schlüter current (I_{PS}) should be also included. The regime of interest to the CTF is located around $f_{\nabla p} \sim 0.5$ and $\beta_T \leqslant 24\%$, which is nearly the range of long-pulse ($\tau > \tau_{skin}$) parameters already produced in the NSTX. In contrast, the regime of interest to the ST Demo is near $f_{BS} \sim 0.9$ and $\beta_T \sim 50\%$ and indicates an important direction of longer term ST research in fusion plasma physics.

The H-mode plasmas on the NSTX are aided by substantial $f_{\nabla p}$ and current driven by NBI (I_{NB}) injected tangentially in the direction of the plasma current. An estimate of I_{NB} can be provided by

$$I_{NB} = \frac{\gamma_{NB} P_{NB}}{n_{20} R_0},$$

where the current drive efficiency (γ_{NB}) in 10^{20} A W^{-1} m^{-2} is approximately given by [30]

$$\gamma_{NB} = E_{NB}^{0.533}(-8.47 \times 10^{-4} + 1.85 \times 10^{-3} T_{e\text{-avg}} - 5.31 \times 10^{-5} T_{e\text{-avg}}^2).$$

It is seen from this that I_{NB} can be in the range of 0.1–0.3 MA ($I_{CD}/I_p \leqslant 0.4$) on the NSTX for the given values of P_{NB} up to 7 MW, $n_{20} = 0.25$–0.75, E_{NB} up to 100 kV and $T_{e\text{-avg}}$ of the order of 1 keV. The combination of these two currents has led to the relatively long pulses in the H-mode plasma with substantially reduced induction loop voltage from the central solenoid magnet (figure 2). Added to table 2 are the estimated maximum level of non-inductive driven current fraction, $(I_{\nabla p} + I_{CD})/I_p$. Research on the NSTX is continuing to understand the remaining inductive current drive fraction and test operating scenarios to reduce this fraction to zero [31].

To sustain a driven burn ($Q \sim 2$–4) in the CTF, it is necessary to maintain the fusion product of $T_i n_i \tau_E$ up to the level of 5×10^{19} keV s^{-1} m^{-3}. The normalized fusion product $\beta_N H_{89P}$ represents an equivalent plasma condition that can be tested on the NSTX. Here H_{89P} is the confinement time factor relative to the so-called ‘L-mode’ plasma [26]. Recent progress [32] in this normalized fusion product, for durations long compared with the plasma energy confinement time, is shown in figure 4 in contrast with the CTF requirements, for a range of normalized fusion products. It is seen that the results on the NSTX, where $\beta_N H_{89P} \sim 10$, are encouraging for the CTF baseline conditions of producing a fusion neutron flux W_L up to

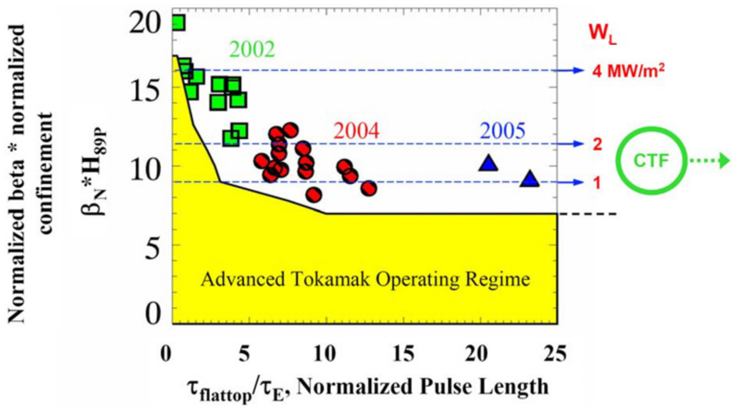

Figure 4. Progress on the NSTX in the normalized fusion product $\beta_N H_{89P}$ versus the plasma flattop time normalized to τ_E, in contrast with the equivalent conditions obtained in tokamaks so far. The flattop times have also reached beyond the plasma current redistribution times.

$2\,\mathrm{MW\,m^{-2}}$. To double W_L in CTF towards the level of the Demo would require a substantially higher $\beta_N H_{89P}$ ($\sim$16).

3.4. Estimates of steady state conditions in CTF

To maintain steady state conditions, it is necessary to calculate the plasma current profile evolution driven by a combination of NBI, bootstrap effect and, if necessary, a moderate amount of RF for profile tailoring. Assuming stabilization of global MHD modes via strong plasma rotation without active feedback control, it is further necessary to determine if the plasma profiles so determined would be stable. The TSC [33] and PEST-II [34] codes are used in these calculations, for the baseline case producing $W_L = 1\,\mathrm{MW\,m^{-2}}$, at a density $n_e = 0.69 \times 10^{20}\,\mathrm{m^{-3}}$ and $E_{NB} = 110\,\mathrm{kV}\,\mathrm{D}^0$. TFTR-type positive ion beam system, upgraded to steady state operation [35], is assumed.

The present TFTR neutral beam source has a vertical height of $\sim$40 cm. Application of beam power within this height on the CTF plasma mid-plane would lead to a highly peaked deposition profile. It is therefore desirable to spread the neutral beam cross section vertically to span the height anticipated for the mid-plane access to help produce a relatively broad NBI driven current profiles in the CTF. Figure 5 shows the J_{NB} profiles produced when $P_{NB} = 30\,\mathrm{MW}$ is applied to produce a total $I_{NB} \sim 5\,\mathrm{MA}$ for both values of H_{NB} of the NBI cross section. It is seen that a broad J_{NB} profile can be obtained by increasing H_{NB} to 160 cm, within the total vertical height ($\sim$175 cm) of the mid-plane radial access (see, figure 8). This is expected to help maintain a relatively broad plasma current profile required for low internal plasma inductance $\ell_i(1)$, high central safety factor q_0, high bootstrap current and macroscopic plasma stability.

Free-boundary equilibrium calculations are carried out and indicate (figure 6) that plasma elongations up to 3.2 can be produced with the distant PF coils for $\ell_i(1) < 0.5$, for $3.0 \leqslant \beta_N \leqslant 4.5$. In the case of inboard limited plasmas, this is accomplished by controlling the location of the X-point inside the VV without allowing the plasma to connect to it. However, the triangularity δ reaches 0.45 only at the lower $\ell_i(1)$ values of about 0.3, progressively decreasing to $\delta = 0.2$ as $\ell_i(1)$ rises to 0.5. Ideal MHD stability of the $n = 1$ kink mode, without a wall, shows that the reference shape $\kappa = 3.2$ and $\delta = 0.4$ are stable in the range of $3.0 \leqslant \beta_N \leqslant 4.5$, required for CTF, with $\ell_i(1) < 0.5$. Ideal MHD stability of lower elongation

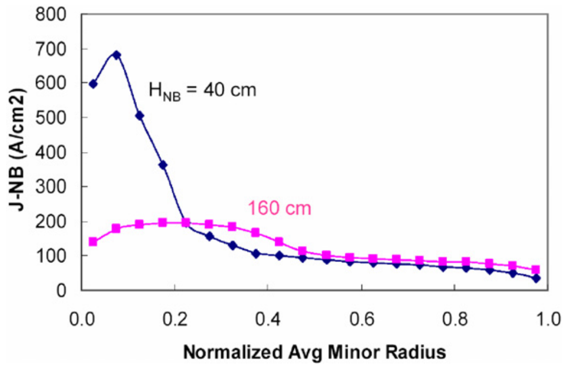

Figure 5. NBI driven current profile J_{NB} for the baseline CTF operation, using 40 and 160 cm heights (H_{NB}) for the beam cross section.

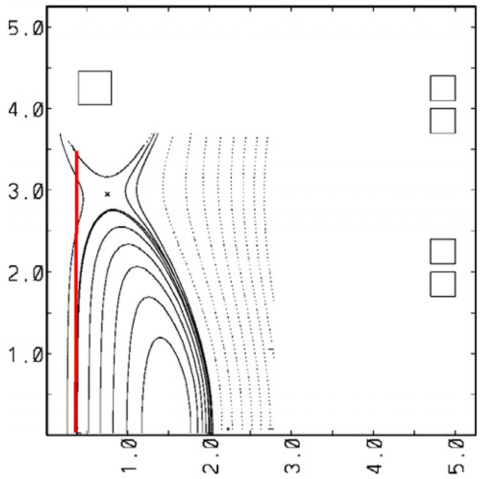

Figure 6. Inboard limited CTF plasma with $\ell_i(1) = 0.25$, $\kappa = 3.2$, $\delta = 0.4$, $\beta_N = 4.0$ and $\beta_T = 20\%$. The boxes indicate the locations and cross sections of the poloidal field coils.

and triangularity plasmas is also examined and can be made stable by further adjustments to the current profile.

The broad NB deposition and driven current profiles (figure 5) are combined with bootstrap current and an assumed off-axis current produced by electron Bernstein wave (EBW) to enable a range of $0.25 \leqslant \ell_i(1) \leqslant 0.5$. The consistency of the current profile, pressure profile, plasma shape, PF coil capability and ideal MHD stability without active feedback are studied and expected to be possible. Recent modelling calculations of EBW current drive in the NSTX high beta plasmas [36] indicated that it would be appropriate to use frequencies just above the first or the second harmonic (>70 or 140 GHz for $B_{T0} = 2.5$ T) to localize the EBW-driven current profile at relatively large minor radius. A relatively high efficiency ζ_{EBWCD} $(= 0.33 I_{CD} R n_e / P_{CD} T_e) \sim 0.4$ is also anticipated, providing $\sim$0.1 MA MW^{-1} for nominal CTF plasma conditions of $T_e \sim 10$ keV and $n_e \sim 10^{14}$ cm^{-3}. This suggests that a nominal EBW power of $\sim$10 MW would be required to drive $\sim$1 MA EBW current in CTF.

The free-boundary evolution code TSC is used to examine the flattop plasma profiles using the extrapolated NSTX thermal diffusivities (table 2) according to the ITER H-mode

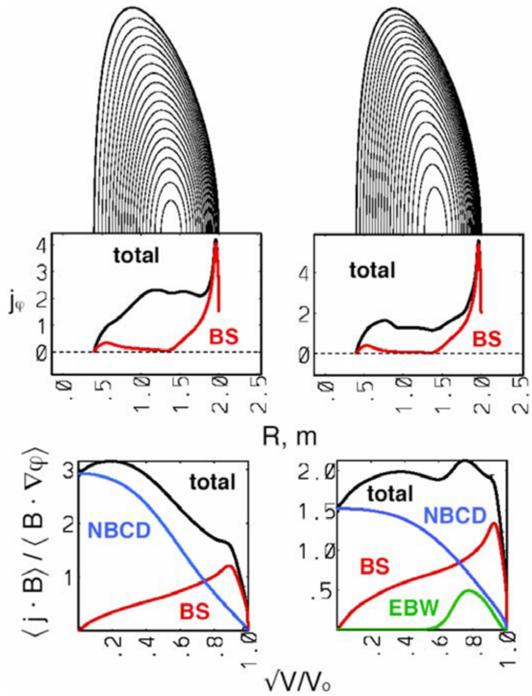

Figure 7. CTF plasma current profiles calculated by the JSOLVE code for the steady state TSC simulation. A profile with $\ell_i(1) = 0.5$ & $q_0 \sim 2$ can be maintained by I_{NB} and I_{BS} (left-hand side) using $P_{NB} = 30\,\mathrm{MW}$, while adding $I_{EBW} = 1\,\mathrm{MA}$ would allow $\ell_i(1) = 0.25$ & $q_0 \sim 4$.

scaling of the confinement time with $H_{98} = 1.5$ and to examine the solenoid-free ramp-up requirements. Figure 7 shows the CTF plasma profiles for the $W_L = 1\,\mathrm{MW\,m^{-2}}$ operation conditions indicated in table 3.

Startup of the plasma current to several hundred kiloamperes level without using a central solenoid was recently demonstrated in JT-60U [37]. There an ECW power of $\geqslant 1\,\mathrm{MW}$ was applied initially at a resonance location (for 3.4 T) just inside of the major radius (~3.4 m), where a poloidal magnetic field null was temporarily maintained. Simultaneously, the vertical field was swung from −0.08 T to +0.08 T in a time scale of about 20 ms to provide an initial flux swing of the order of up to 2 Wb. A relatively low initial gas pre-fill of $\sim 0.2\,\mathrm{Pa\,m^{-3}}$ is necessary to ensure effective generation of ~100 kA, which was subsequently sustained for up to 200 ms. Tests of similar scenarios in much smaller ST devices TST-2 [38] and LATE [39] have been successful in producing ~10 kA currents, which were nominally ~10% of the full plasma current. These encouraging results suggest that a similar scenario can be applied in CTF using ~1 MW ECW at 70 GHz (fundamental resonance at $R_0 = 1.2\,\mathrm{m}$) frequency to initiate a low density target plasma at ~100 kA in current. Subsequent application of heating and current drive (via ECW–EBW, NBI) and increased vertical field to raise the plasma current further should also become possible as suggested by the above-mentioned point model simulation.

A point model simulation of the plasma ramp-up to steady state operation is also carried out [40] to assess the global plasma behaviour and requirements of the CTF plasma. The results show that, assuming a modest initial I_p of 100 kA, an appropriate combination of β rise, poloidal field coil induction, fuelling, heating and the external and internal driven currents could bring I_p to the full level (~10 MA), producing a sustained fusion power P_{DT} of ~150 MW and $W_L \sim 2\,\mathrm{MW\,m^{-2}}$ in neutron flux.

Table 3. CTF parameters with $R_0 = 1.2$ m, $a = 0.8$ m, $\kappa = 3.2$, $B_{T0} = 2.5$ T, $I_{TF} = 15$ MA, $n_D = n_T$, for $W_L = 1$, 2 and 4 MW m^{-2}, assuming $H_{98e} = 0.7$ and $H_{NCi} = 0.7$.

Operation phase	I	II	III
W_L (MW m^{-2})	1	2	4
I_p (MA)	9.6	12.3	15.0
q_{cyl}	4.0	3.1	2.6
β_N (% m T MA^{-1})	4.1	4.9	6.1
β_T (%)	19	30	45
n_e (10^{20} m^{-3})	0.66	1.0	1.5
n_e/n_{GW} (%)	15	17	21
T_i (keV)	34	31	28
T_e (keV)	11	12	14
Equivalent H_{98}	2.6	2.1	1.9
f_{BS} (%)	67	60	60
P_{NB+RF} (MW)	21	29	41
E_{NB} (kV) for D^0	105	158	240
P_{DT} (MW)	72	144	288
$P_{Beam-Plasma}/P_{DT}$ (%)	23	20	15
f_{Rad} (%) for $\Gamma_{Div} \leqslant 15$ MW m^{-2}	49	71	85
Achievable f_{BR} (%)	100	100	95

In CTF design concepts with vertical field coils placed close to the plasma chamber relative to the conducting blanket and shield, an additional technique of double-null merging technique could be used to initiate large plasma current without the use of central solenoid. This technique has been successfully demonstrated in MAST [41] recently, albeit using in-vessel vertical field coils, producing high temperature and density plasmas with up to 340 kA in current, which is a substantial fraction (∼25%) of the full operating plasma current. These considerations strongly indicate that solenoid free initiation and ramp-up of plasma current in CTF is achievable, given continued research.

4. An attractive CTF design configuration

The ST plasma conditions indicated in section 3 are expected to enable attractive CTF design concepts. To produce the CTF fusion engineering and technology testing conditions, including an operational duty factor that is one order of magnitude larger than the operational target of ITER, all chamber systems must allow relatively straightforward replacement via remote handling, to minimize the mean-time-to-repair (MTTR) [4]. Figures 8 and 9 depict the arrangements of the chamber systems in such a CTF. In this design configuration, the TF coil, the vacuum boundary and the personnel shielding systems are combined into an integrated component. A demountable exoskeleton steel structure (figure 8) will be required to carry the TF coil in-plane vertical and out-of-plane twisting loads, which are estimated to be ∼60 metric tons and ∼60 metric ton-m, respectively, and are considered to be moderate for the size of the CTF.

The properties of the ST fit well with the CTF mission in small device size and fusion power. As shown in figure 8, the tendency for the ST plasma cross section to elongate vertically [8] at low internal inductance $\ell_i(1) < 0.5$ permits the use of relatively far away vertical and poloidal divertor field coils relative to the plasma minor radius. For a relatively fixed thickness required for a blanket and shield, this feature allows relatively small plasma cross section, as in this case. The large normalized plasma current I_N $(= I_p/aB_{T0}) \leqslant 8$ MA m^{-1}T^{-1} at high safety

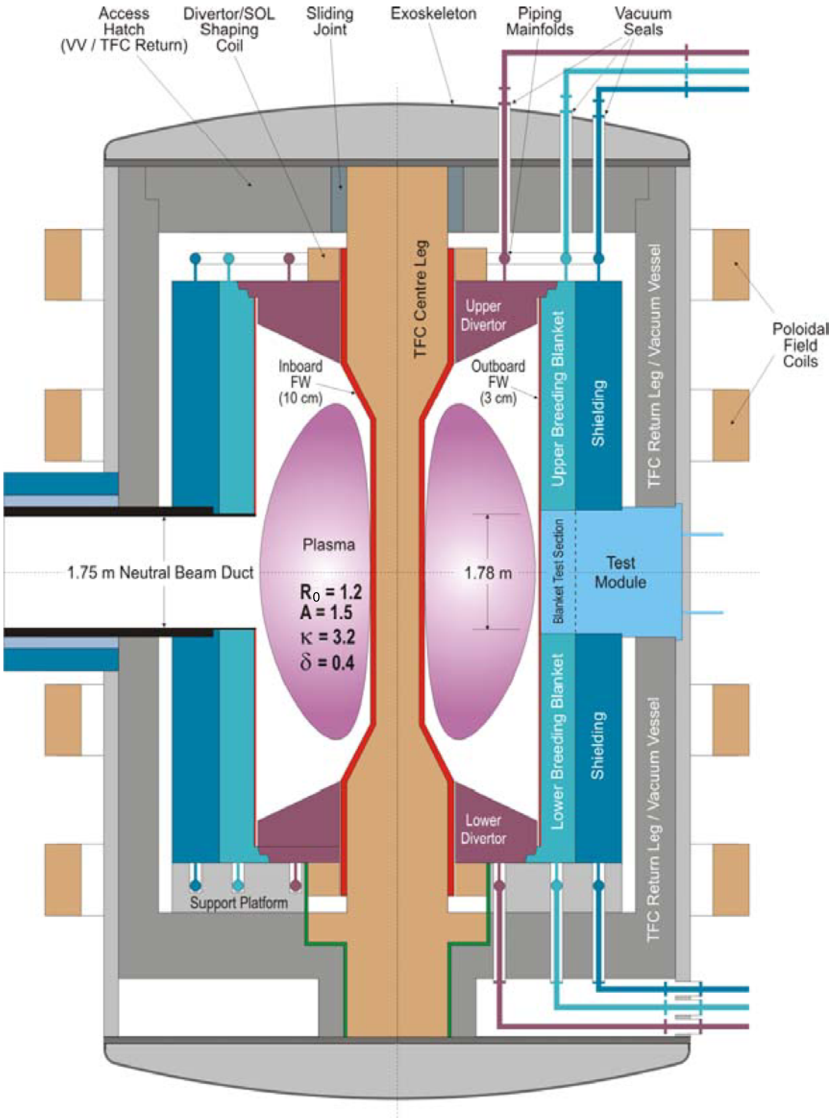

Figure 8. Vertical cross section view of a CTF configured for full remote handling of all chamber systems.

factors $q_{\rm cyl}$ improves the overall expected plasma confinement despite a moderate magnetic field. This together with a moderate current in the TF $I_{\rm TF} \sim I_{\rm p}$ enables the consideration of a slender, demountable, single-turn centre leg [42]. The ability to initiate, ramp-up and maintain the plasma current without the central solenoid and the associated electrical insulator allows the elimination of the inboard nuclear shielding. This makes it feasible further to eliminate inboard blanket modules, leading to a compact ST device with small radius and aspect ratio. In addition, the slender TF centre leg would intercept only $\sim$7% of fusion neutrons [42], which in turn contributes to increasing the tritium breeding ratio in a ST fusion device, including the CTF.

The chamber systems that require frequent replacement, such as the test blanket modules to develop the engineering and technology base for strong fusion power conversion and

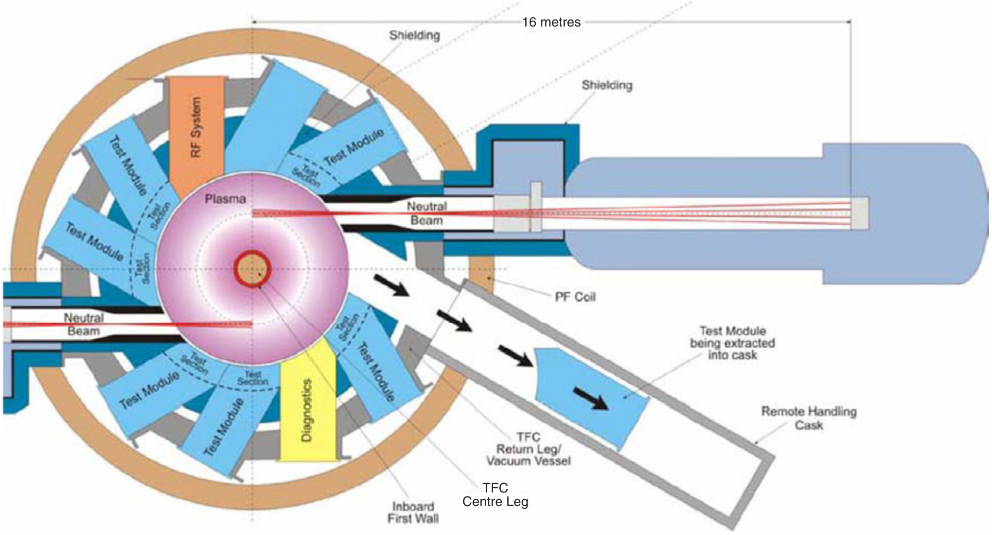

Figure 9. Mid-plane view of a CTF configured for full remote handling of all chamber systems.

tritium breeding, are placed on the mid-plane for direct horizontal replacement. The transfer cask concept for handling the activated components and test blankets in ITER [43] can be used in CTF. Other systems that are likely to require similar un-scheduled access, including radiofrequency launchers, diagnostic access module and neutral beam injectors, should therefore also be placed on the mid-plane. Assuming tangential NBI, the mid-plane chamber systems could be arranged in 'daisy-chains' with nearly identical modules and plasma facing wall area (about 1.5 m × 1.8 m for the case with $R_0 = 1.2$ m) and hence nearly identical exposure to the fusion plasma and neutron fluxes and remote handling casks. As shown in figure 10, a maintenance cask for the neutral beam system, similar to that envisioned for ITER, can be assumed.

Other chamber systems would acquire somewhat more time-consuming, vertical access for remote handling. Figure 10 depicts an arrangement that would make this possible. A sizeable vertical maintenance cask can be envisioned to handle the demountable TF coil centre leg, which is estimated to have a total height of about 15 m and a total weight of about 150 metric tons. All other relatively moderate-sized chamber systems would also be handled within this cask. With this design approach, all chamber systems can be accessed vertically following hands-on evacuation and disconnection of services from outside of the shielding and personnel access boundary of the CTF, which is at present assumed to be the outer boundary of the combined TF coil and vacuum boundary. This basic assembly and disassembly concepts have been adopted in other ST-based fusion power plant concepts [11, 12, 42].

5. Choices of CTF parameters

'Systems Codes' have been developed and used [44] to estimate the major parameters and their tradeoffs of toroidal device designs. For ST devices, a new code [45] has been developed to capture in approximation the unique features of the ST plasma and device configuration and design for this purpose. The parametric study reported here determines the parameters that minimize the total auxiliary power while producing a prescribed W_L at the mid-plane test module for a given CTF device design configuration (section 4).

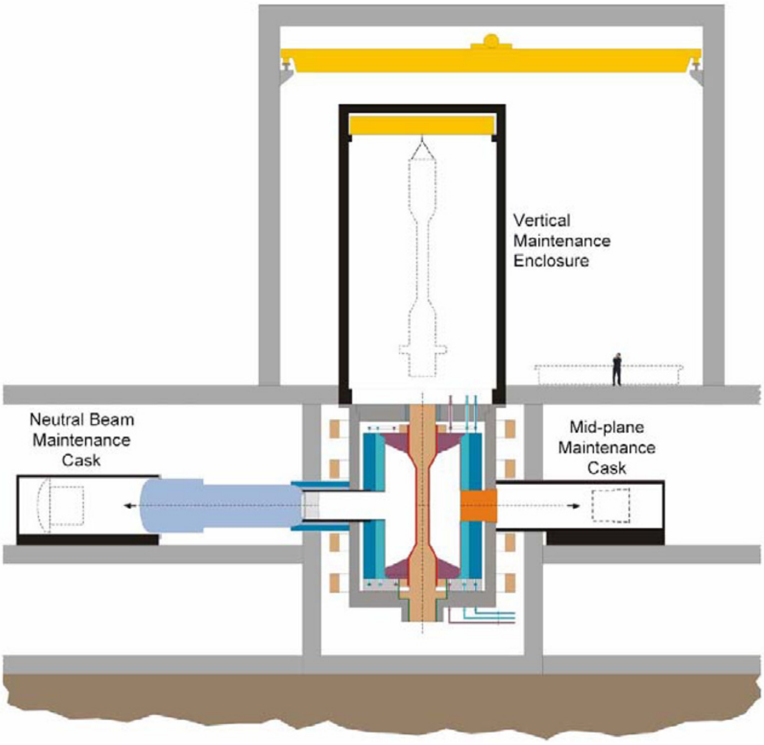

Figure 10. Maintenance cask systems are envisioned to allow horizontal remote replacement of mid-plane modules and the neutral beam systems and vertical remote replacement of other chamber systems in CTF.

In this systems code, relatively standard models for the plasma properties in vertically elongated cross section of toroidal geometry are included, as guided by the latest physics results summarized in section 3. Engineering features described in section 4 are also modelled in approximation and included in the code.

The Systems Code is implemented using EXCEL with the nonlinear optimizer SOLVER to find solutions of design parameters. A typical set of independent variables adjusted by SOLVER include impurity radiation level in divertor-scrape-off layer region, n/n_{GW}, β_{Ni}, β_{Ne}, $T_{0\mathrm{i}}$, $T_{0\mathrm{e}}$, q_{cyl}, P_{DT}, η_{CD} up to the physics limits and J_{TF} up to the engineering limits. Solutions are constrained by power balance and various physics and engineering limits. Table 3 summarizes the key parameters of the CTF assuming $H_{98\mathrm{e}} = 0.7$ and $H_{\mathrm{NCi}} = 0.7$ (table 2) for three levels of fusion neutron flux ($W_{\mathrm{L}} = 1, 2, 4\,\mathrm{MW\,m^{-2}}$), designated as Phases I, II and III.

It is seen that the Phases I & II operation of CTF requires plasma conditions that are near or within the achieved limits (table 2) in q_{cyl} ($\geqslant$2.5 for current driven mode stability), β_{N} ($\leqslant$5.6 for pressure-driven mode stability in the presence of a conducting wall and large plasma rotation) and n/n_{GW} ($\leqslant$1 for edge thermal stability). The modest plasma density in this case also allows $T_{\mathrm{i}}/T_{\mathrm{e}} \geqslant 2$, leading to an apparent enhancement of H_{98} to $\approx$3 without changing $H_{98\mathrm{e}}$ and H_{NCi} from the values given in table 2. The required E_{NB} of $\leqslant$160 kV will permit the use of the TFTR-type positive-ion NBI system [35]. The modest density further leads to a substantial level of beam–plasma fusion fraction in the range of up to 30%.

As W_{L} is doubled to the Demo level for the Phase-III operation, β_{N} is increased up to $\approx$6, which will likely be above the conditions that require active feedback control of the resistive

Table 4. Initial plasma and engineering conditions for the CTF, assuming $H_{NC} = 0.7$ and operating at reduced heating power and $B_{T0} = 1.5$ T to test divertor solutions.

Divertor testing phase	I	II
W_L (MW m^{-2})	0.25	0.5
I_p (MA)	5.9	7.9
q_{cyl}	3.8	2.9
β_N (% m T MA^{-1})	4.4	5.7
β_T (%)	22	37
$\langle n_e\rangle$ (10^{20} m^{-3})	0.34	0.45
$\langle n_e\rangle/n_{GW}$ (%)	12	12
$\langle T_i\rangle$ (keV)	27	35
$\langle T_e\rangle$ (keV)	6.8	8.7
Equivalent H_{98}	3.0	3.0
f_{BS} (%)	72	68
P_{NB+RF} (MW)	11	16
E_{NB} (kV) for D^0	55	73
P_{DT} (MW)	18	36
$P_{Beam-Plasma}/P_{DT}$ (%)	18	22
Γ_{Div} (MW m^{-2}) @ $f_{Rad} = 50\%$	7.5	12
Achievable f_{BR} (%)	100	100

wall modes (RWMs) [47], while n_e/n_{GW} still remains modest. The density is increased so that $T_i/T_e \approx 2$ and $E_{NB} \approx 300$ kV, which will require JT-60U [48] and LHD-type [49] of negative-ion NBI system. In all three cases, f_{BS} remains in the range of 60%. As additional mid-plane ports are utilized by increased auxiliary heating power, the achievable f_{BR} by tritium breeding blankets also decreases to 95%, resulting in substantially increased rate of tritium consumption. Detailed neutron scattering and absorption analysis, accounting for the various materials in the chamber systems [46], will be required to determine the achievable f_{BR} more reliably.

It is seen that the fusion amplification Q could be in the range of 3.5–7.0. In principle a net tritium breeding fraction f_{TB} of $\sim$100% can become possible if the fusion blankets are capable of a local tritium multiplication near 140%, which should be achievable together with high power conversion efficiency [4]. These lead to a closer approach to the requirements of the Demo [11, 12].

These results show that the CTF driven by NBI has high potential for reliable plasma operations for W_L in the range of 1–2 MW m^{-2} without requiring active feedback control of MHD modes. It further has the potential to achieve W_L up to 4 MW m^{-2}, which is the Demo level, if active feedback control of field errors and RWMs could reliably allow access to plasma conditions of higher β_N ($\approx$6) and f_{BS} ($\approx$60%).

One of the severe challenges for the CTF is expected to be its high divertor heat flux. For operation at $W_L = 1$ MW m^{-2}, the heat flux parameter P/R is about 30 MW m^{-1} (table 3), which is comparable to that anticipated in ITER. To test and develop divertor solutions for the CTF, it may be necessary to operate initially at a reduced performance, examples of which are provided in table 4 for $W_L = 0.25$ and 0.5 MW m^{-2} with $B_{T0} = 1.5$ T.

It is shown that an initial divertor testing phase could be operated with P/R as low as 13 MW m^{-1}, while producing a moderate W_L, which also could be appropriate for initial shake-down operation of all the chamber systems. The plasma current required could be reduced to $\approx$6 MA initially, while the normalized parameters remain in the reliable operating envelope as described before. To minimize the heating power, the systems code determines a very low density regime ($n_e/n_{GW} = 12\%$) for operation, which in turn points to a moderate NBI energy ~60 kV. This suggests the further possibility of operation with increased density and NBI energy (by about a factor of 3) leading to somewhat increased heating and current drive power.

It is noted that the total heating power of 11 MW with an estimated $\Gamma_{Div} \sim 8\,\mathrm{MW\,m^{-2}}$ would be within long-pulse operation conditions of today's tokamaks.

6. Fusion plasma and engineering parameter regimes for CTF

With worldwide preparation of the physics basis for ITER [50] and the anticipated ITER construction beginning in 2006–2007, the burning plasma (Q~10) physics base, for tokamak with a large plasma size scale ($\rho_i^{*-1} = a/\rho_i \sim 10^3$) at $T_i \sim 20$ keV, is expected to be completed during 2020–2030. Here ρ_i is the average plasma ion gyro-radius. This, coupled to progress in the USDOE strategic goal for fusion [6] to '*Develop a fundamental understanding of plasma behavior sufficient to provide a reliable predictive capability for fusion energy systems*,' would also establish the driven burning plasma ($Q \sim 3$–7) knowledge base for CTF, which is characterized by a moderate plasma size scale ($\rho_i^{*-1} \sim 10^2$) at $T_i \sim 20$ keV. However, owing to the large extensions in the ST of the fusion plasma parameter regimes [2], it is necessary to establish the extended knowledge base prior to the CTF operation, in the parameter ranges suggested in table 4. Further, solenoid-less initiation, ramp-up and sustainment of I_p are needed and uniquely important to CTF.

6.1. ST Plasma physics regime

Section 3 presented several important advances in the ST plasma physics that have guided the selection of the basic CTF parameters. Though the presented parameters and design concept of the CTF indicate relatively attractive cost-effectiveness, the results are subject to the rather unique plasma regimes being investigated in today's ST plasmas [1, 14, 15]. The conditions, which define the anticipated characteristics of the ST plasma in the NSTX and in CTF, are provided in table 5, in contrast with ITER.

It is shown that the ST extends the plasma parameter regimes beyond those of ITER and as a result, provides complementary data to normal aspect ratio experiments in addressing key issues of interest to ITER physics optimization [50]. It will therefore be important to answer the key questions of fusion plasma physics that stem from the extended ST plasma regimes.

From table 5 can be identified the following plasma physics questions of importance to improved estimates of the anticipated CTF plasma properties. These physics questions, also of interest to the Tokamak physics, include the effects of

- large ε, κ, B_p/B_T and mirror ratio at the plasma edge on the edge localized modes (ELMs);
- large flow on the plasma equilibrium and the global pressure-driven MHD modes;
- strong shaping and the large flow on the MHD mode locking as a function of the error field magnitude;
- super-Alfvénic fast ions on the various Alfvén modes or possibly introducing new modes in the plasma, particularly at β values of interest to the CTF;
- very high β and low collisionality ν^* on the electron and ion energy confinement times;
- strong plasma flow and negative magnetic shear on the electron energy confinement in low density L-mode plasmas;

Table 5. Fusion plasma physics regimes revealed in the NSTX and projected for CTF, compared with those of ITER.

Plasma conditions	NSTX	CTF	ITER
Toroidicity, $\varepsilon = a/R_0$	$\leqslant$0.71	$\leqslant$ 0.67	$\leqslant$0.3
Elongation, κ	$\leqslant$2.5	$\leqslant$3.2	$\leqslant$2
B_p/B_T in large-R region	$\sim$1	$\sim$1.5	$\sim$ 0.2
β_T/β_0 (central local β)	$\leqslant$0.4/$\sim$1	$\leqslant$0.45/$\sim$1	$\sim$0.02/0.06
Edge mirror ratio, M_B	$\leqslant$4	$\leqslant$4	$\leqslant$2
Plasma Alfvén Mach number, M_A	$\sim$0.4	$\sim$0.4	$\sim$0.01
Flow shearing rate (s^{-1})	$\sim 10^6$	$\sim 10^6$	Small
$V_{fast}/V_{Alfvén}$	1–5	3–6	1–2
$\beta_{fast}(0)/\beta_0$	$\leqslant$0.8	$\leqslant$0.6	$\leqslant$0.2
Normalized plasma size, $1/\rho_i^*$	$\sim$40	$\sim$80	$\sim$800
Normalized fast ions plasma size, $1/\rho_{fast}^*$	$\sim$ 5	$\sim$8	$\sim$80
Dielectric constant, ε_e ($= \omega_{pe}^2/\omega_{ce}^2$)	$\sim 10^2$	$\sim$10	$\sim$1
Internal poloidal flux, $\sim \ell_i R_0 I_p$ (MA m)	$\sim$0.3	$\sim$4	$\sim$60

- large dielectric constant ε_e and large particle trapping fraction (low aspect ratio) on the edge conversion and core propagation and absorption of the Electron Bernstein Wave (EBW);
- low aspect ratio, strong shaping and low internal inductance on the solenoid-less current initiation using a combination of RF electron heating and vertical field swing.

Whereas present-day experiments [14, 15] will shed much light on these physics topics, the ST fusion plasma science knowledge base at several megaampere level, including the goals of the next-step spherical torus (NSST) [51], and in truly steady state conditions extending beyond the achievements of Tore-Supra [52] and TRIAM-1M [53] towards $\beta_T \sim 20\%$, will be needed. New experimental results from present-day ST experiments, when adequately studied and understood, will help determine realistic conditions and requirements for reliable plasma operations in CTF.

6.2. Engineering science base

Successful ITER plasma operations through 2020–2030 are expected further to establish the plasma control engineering and technology base for long pulse ($\sim 10^3$ s) burning plasmas producing a fusion neutron wall flux $W_L \sim 0.8\,\mathrm{MW\,m^{-2}}$. The systems used to heat, fuel, pump, confine and control the ITER plasma would establish the basis for the initial operation of CTF at $W_L \sim 1\,\mathrm{MW\,m^{-2}}$. The relatively moderate E_{NB} ($\leqslant$160 kV, 250 kV) determined in section 5 for $W_L \leqslant 2\,\mathrm{MW\,m^{-2}}$ and $4\,\mathrm{MW\,m^{-2}}$ operation, respectively, suggests that present-day positive-ion [35] and negative-ion [48, 49] NBI techniques can be applied but need to be extended to steady state operations. The ECW technology ($\sim$20 MW at 140–170 GHz) applied to ITER would establish the needed basis for the anticipated EBW application ($\sim$10 MW at 70–140 GHz) in CTF. An experience base of high heat flux divertor material and power handling from ITER are expected to be adequate for $W_L = 1\,\mathrm{MW\,m^{-2}}$ operation in CTF. However, CTF operation at WL $= 2$–$4\,\mathrm{MW\,m^{-2}}$ will likely require new advances. Further, the engineering science knowledge base for the water-cooled, single-turn, normal conducting centre leg of the TF coil is needed and uniquely important to CTF.

The fusion plasma control engineering and technology capabilities so provided in CTF, together with the full remote handling capabilities indicated in section 4, would introduce the reactor-like conditions in which all chamber systems can be tested effectively to dpa $\sim$60. It is important to note that the test modules would be provided by the fusion engineering, material and technology R&D community, who would be users of the CTF to carry out the testing program. It is anticipated that extensions of the ITER test blankets, divertor modules and other plasma-facing components would provide the basis for the initial systems to be tested on CTF.

7. Discussion

In view of the results of this work, the following discussions have become appropriate.

This study indicates that the CTF has high potential to produce reliably, in moderate size and fusion power, W_L levels that are 2–3 times those anticipated in ITER with two orders of magnitude longer duration in sustained operation. The level of fusion plasma and technology base to be established by ITER available in the 2010s will likely define the Phase-I ($W_L = 1\,\mathrm{MW\,m^{-2}}$) plasma operation conditions of the CTF (table 3). Continued progress in plasma physics and that in engineering and technology using the CTF towards practical fusion energy will therefore need to be advanced in tandem; advances in one will motivate and require those in the other, eventually reaching the level of the Demo near and up to $W_L = 4\,\mathrm{MW\,m^{-2}}$. A design with full remote handling assembly and disassembly will therefore likely to be indispensable for this progress to be achieved in a timely and cost-effective manner.

The availability of effective remote handling of all chamber systems in a fusion energy producing device, to be tested and demonstrated in CTF, may have an important implication in the material dpa testing level required to develop practical fusion power. With a 2-year maintenance cycle, a Demo delivering $4\,\mathrm{MW\,m^{-2}}$ flux and 75% duty factor would accumulate 60 dpa between scheduled replacement of chamber systems. As a result, CTF with Phase II capability (table 3) and 30% duty factor would deliver in 10 years the engineering science knowledge base for the initial Demo operation. This implies that the goals for fusion material science testing may be reduced to 60 dpa for the next three decades in support of the effort to deliver net fusion electricity.

The results of the CTF systems code analysis suggest that a wider range of parameters and performance of CTF would be possible and be of interest to an effective development of fusion energy. The lower end could be a small fusion unit with $R_0 \leqslant 1\,\mathrm{m}$ producing reduced P_{DT} ($\sim$10 MW) and W_L ($\sim 0.25\,\mathrm{MW\,m^{-2}}$) for extended plasma, engineering and nuclear studies. The higher end could be a Pilot Plant [54] with $R_0 \sim 1.5\,\mathrm{m}$ capable of testing the integrated operation of fusion electricity production at substantial P_{DT} ($\sim$300 MW), assuming reliable ST plasma conditions similar to those assumed for the Phases I & II of the CTF.

The ST provides an extended fusion plasma parameter regime in simplified device configurations and reduced size. The potential benefits of this special combination are only beginning to be examined in the example of the CTF. More investigation on this subject is therefore likely to bring forward additional insights of its potential benefits and challenges to the development of plasma science and fusion energy.

Finally, the cost for the CTF capable of Phase-I ($W_L = 1\,\mathrm{MW\,m^{-2}}$) operation is estimated, scaled from those of the major systems designed for the Phase-I ITER operation. The results suggest a total cost of the order of \$1.05B in 2002 dollars, not including contingency, consisting of \$0.19B for toroidal device; \$0.19B for device ancillary systems including remote handling equipment; \$0.09B for device gas & coolant systems; \$0.12B for power supply & control; \$0.21B for heating, current drive, & initial diagnostics and \$0.25B for site, facilities and equipment.

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