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FUSION ENGINEERING AND PLASMA SCIENCE CONDITIONS OF SPHERICAL TORUS COMPONENT TEST FACILITY

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A broadly based study of the fusion engineering and plasma science conditions of a Component Test Facility (CTF),[1] using the Spherical Torus or Spherical Tokamak (ST) configuration,[2] have been carried out. The chamber systems testing conditions in a CTF are characterized by high fusion neutron fluxes $\Gamma_n > 4.4\times10^{13}$ n/s/cm^2, over size scales > 10^5 cm^2 and depth scales > 50 cm, delivering > 3 accumulated displacement per atom (dpa) per year.[3] The desired chamber conditions can be provided by a CTF with R_0 = 1.2 m, A = 1.5, elongation ~ 3, I_p ~ 9 MA, B_T ~ 2.5 T, producing a driven fusion burn using 36 MW of combined neutral beam and RF power. Relatively robust ST plasma conditions are adequate, which have been shown achievable[4] without active feedback manipulation of the MHD modes. The ST CTF will test the single-turn, copper alloy center 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, and sustainment to multiple MA level. A new 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 lower-cost CTF devices to suit a variety of fusion engineering science test missions.

I. INTRODUCTION

Successful development of practical fusion energy will require research and development that combine fundamental and applied science. Fusion energy Component Test Facilities (CTF), aimed at advancing the fusion engineering sciences required, will necessarily entail similarly combined efforts. A recent plan issued by the USDOE Office of Science[5] identified a broad strategic goal to "*develop the new materials, components, and technologies necessary to make fusion energy a reality*" for the U.S. Fusion Energy Sciences Program. In this plan, a CTF would be created to succeed the International Thermonuclear Experimental Reactor (ITER)[6] construction to address this goal. The fusion engineering science conditions to be produced by the CTF to achieve its mission are summarized in Section II. Data from CTF will determine how the full and steady state fusion conditions affects plasma chamber materials and components, and limits their operating life. This will in turn enable improvements in the engineering science knowledge base needed to support a decision to build a demonstration power plant (DEMO) that aims to produce net electrical output. A CTF will therefore provide, substantially beyond the levels planned for ITER, the testing conditions in high material displacement per atom (dpa) and operational duty factor needed to establish the engineering science basis for DEMO.

The low aspect ratio A (= major radius / minor radius = R_0/a) of the ST readily permits modularization of the chamber and the toroidal field (TF) coil systems, allowing direct access for remote handling, thereby to achieve the required neutron fluence and duty factor. The engineering design features to achieve this with an ST CTF is presented in Section III.

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. Aimed at this goal are the Innovative Confinement Concept experiments in a number of confinement configurations. Among these, the science of the ST plasma made strong progress due to the rapid deployment and experimentation in recent years of major ST facilities such as NSTX[4] and MAST.[7] The progress is further enhanced by well-defined scientific relationships of the ST[8] to the tokamak via high safety factor (q_{cyl}), and to the Reversed Field Pinch (RFP),[9] the spheromak,[10] and the Field Reversed Configuration (FRC)[11] via high plasma β (= average plasma pressure /

magnetic field pressure) and strong magnetic curvature. It has thus become timely to update earlier estimates[12] of the properties of the volume neutron source plasma, which provides the fusion neutrons of the CTF. To ensure high duty factor operation, the CTF plasmas must operate in a plasma 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 IV.

The appropriate plasma and engineering science conditions of the CTF are modeled in approximation in an systems optimization algorithm to survey the range of acceptable designs. A design with $R_0 = 1.2$ m, delivering the baseline performance of fusion neutron wall flux Γ_n of $= 4.4\text{-}8.8\times10^{13}$ n/s/cm^2 is set forth as a good trade-off between size, performance, cost and risk. If the performance is pushed toward the physics limits of the advanced regimes anticipated for a power plant,[13,14] this CTF is estimated to deliver $\Gamma_n = 17.6\times10^{13}$ n/s/cm^2, which is an anticipated level for DEMO. However, this would also require that all CTF chamber systems and facilities are developed to deliver and handle this level of performance. The fusion plasma and engineering science landscape of the compact ST CTF will be presented in Section V.

An updated understanding of the CTF presents a renewed opportunity to identify, by comparing the desirable plasma conditions of the CTF with the current research using the ST to address the major scientific issues of fusion plasmas. For CTF to address efficiently its mission in fusion engineering science, a strong fusion plasma science basis must be available prior to CTF operation. The CTF scientific bases are identified in Section VI in reference to the latest progress in ST research.[15,16] 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.

In addition, it is appropriate to assume that ITER[6] will demonstrate before 2020 the science 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 Γ_n of $= 4.4\text{-}8.8\times10^{13}$ n/s/cm^2 would be 2-3 times the ITER level. The requirements in fusion engineering science for the baseline CTF operation and control, including the single-turn normal conducting TF coil center leg, will also be covered in Section VI.

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

II. CTF FUSION ENGINEERING SCIENCE MISSION AND REQUIRED CONDITIONS

The CTF is a facility for establishing the integrated fusion engineering science knowledge base for the chamber systems needed to produce practical fusion power. The chamber systems for magnetic fusion have been characterized in a number of fusion reactor concept studies.[17] A comprehensive assessment of the required knowledge base of the fusion chamber systems were reported by Abdou et al.[3]

Many complex scientific phenomena 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 ultra-scale computing modeling, 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” with which to test and develop such a knowledge base required for DEMO.

The key ingredients of the full conditions have been identified for CTF,[3] and can be restated in Table I in terms of engineering and material science, in comparison with the ITER design and those anticipated for a full-remotely maintainable DEMO[13,14] that assumes a 2-year maintenance cycle (see, Section III).

It is seen that the mission of CTF requires it to approach the DEMO chamber conditions in all aspects except in fusion neutron and neutron heat fluxes. There is therefore a premium value to enhance the CTF conditions toward those of DEMO by increasing these fluxes. ITER provides adequate conditions in the scale of materials depth and transverse spatial scales of interest; falls short of the DEMO neutron and neutron heat fluxes as in the case of the CTF baseline; but falls far short in dpa, duration, and tritium self-sufficiency. A successful ITER program will therefore provide incentive to deploy CTF on the path toward DEMO.

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

Condition	ITER	CTF	DEMO
14-MeV neutron flux through chamber surface, Γ_n (10^{13}n/s/cm^2)	~2.6	>4.4	~18
14-MeV neutron heat flux through chamber surface (W/cm^2)	~60	>100	~400
Depth of energetic (>1 keV) neutron-material interactions (cm)	~50	>50	~50
Transverse spatial scale of interest to energetic (>1 keV) neutron-material interactions (cm)	~1000	~500	~1000
Total chamber systems displacement per atom, *dpa*	~3	≤60	~60
Dpa per full-flux-year, *D*	~6	>10	~40
Duration of sustained neutron interactions (s)	~10^3	>10^6	~10^7
Tritium self-sufficiency goal (%)	~?	~90	>100
Duty factor, F_D (%)	2.5	30	75

To support a timely establishment of the fusion engineering science knowledge base for DEMO, the CTF would do well to complete its mission in a time scale *T* of 10-20 years. To reach the life-time *dpa*, the required duty factor, F_D, would be:

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

This indicates an F_D > 30% for a CTF operated at the minimum fusion neutron flux for 20 years. 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 toward the CTF conditions.

III. A ST DESIGN TO ACHIEVE CTF MISSION

To achieve the CTF fusion engineering science 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 rapid replacement through remote handling, to minimize the Mean-Time-To-Replace (MTTR).[3] The small aspect ratio of the ST introduces the possibility of a fully demountable TF coil system, if a single-turn, normal conducting center leg is used in the absence of a central solenoid magnet or substantial nuclear shielding.[18] Remote handling of all chamber systems in radial or vertical directions would then be made possible. Figures 1 and 2 depict the arrangements of all chamber systems in such a CTF.

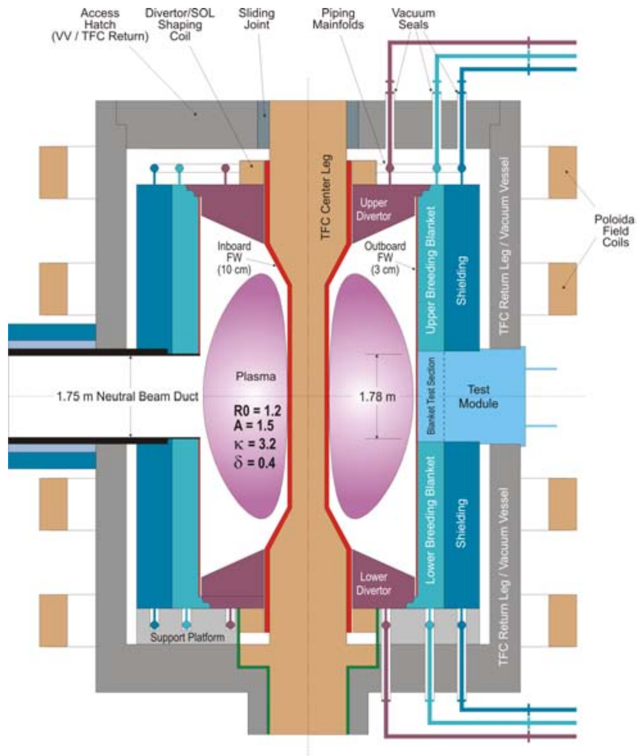

Fig. 1. Vertical cross section view of a CTF configured for full remote handling of all chamber systems.

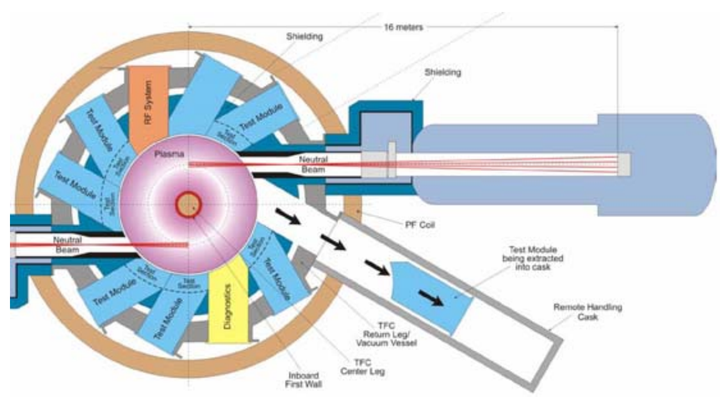

Fig.2. Mid-plane view of a CTF configured for full remote handling of all chamber systems.

The chamber systems that require frequent un-scheduled replacement, such as the modules to test and develop the engineering knowledge base for strong fusion neutron heating and tritium fuel reproduction, are placed on the mid-plane for rapid horizontal replacement. The transfer cask concept for handling the nuclear test blankets in ITER[19] can be used in CTF. Other systems that likely require similar access, including radiofrequency launchers, diagnostic systems, and neutral beam injection, could also be place on the mid-plane. Assuming tangential neutral beam injection, the mid-plane chamber systems could be arranged in "daisy-chains" with nearly identical modules with identical plasma facing wall area (about 1.5m × 1.8m for the case with R_0 = 1.2m), and hence nearly identical exposure to the fusion plasma and neutron fluxes. As shown in Figure 3, a maintenance cask for the neutral beam system, similar to that envisioned for ITER, is used.

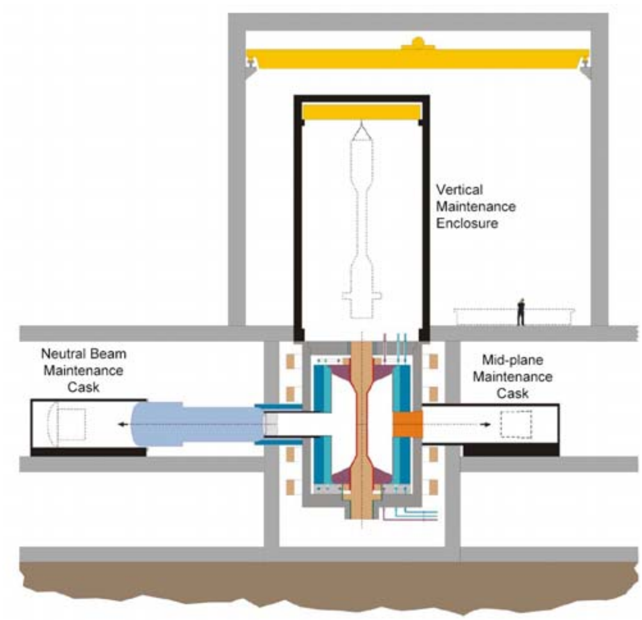

Fig.3. Shielded maintenance cask systems are envisioned to allow horizontal remote replacement of mid-plane modules and the neutral beam systems, and vertical remote replacement of other chamber systems in CTF.

Other chamber systems would acquire vertical access for remote handling. Figure 3 depicts the arrangement that makes this possible. A sizable shielded maintenance enclosure can be envisioned to handle the relatively moderate size of the chamber systems, including the TF coil center leg, which would have a total height of about 15 m and a total weight of about 150 metric tons. As depicted in Figure 4, the chamber systems can be accessed vertically following hands-on evacuation and disconnection of all services from outside of the shield enclosure of the CTF. A complete remote disassembly of the chamber systems would proceed with removal of the single-turn TF coil electrical joint, followed by the top shielding disc, upper poloidal field (PF) coil, the upper

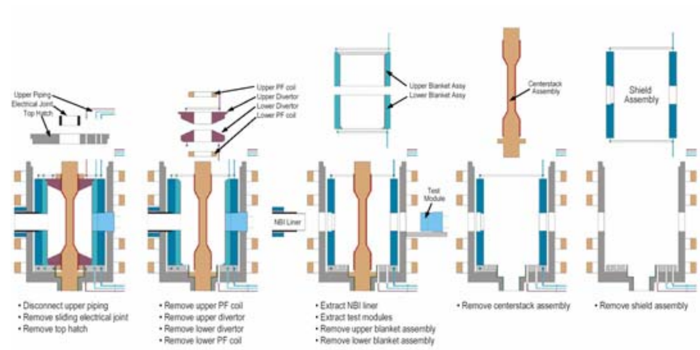

Fig. 4. Vertical remote disassembly procedure envisioned for CTF chamber systems.

and lower divertor assemblies, the lower PF coil, the upper and lower cylindrical blanket assemblies, the TF coil center leg assembly, and the cylindrical shield assembly. The cylindrical TF coil return legs, the rest of the PF coils, and the lower shield systems could be left in place as long as they do not interfere with the remote maintenance activities. The mid-plane modules, including the neutral beam liner, diagnostic systems and radiofrequency launchers would be removed horizontally to facility the disassembly.

The entire procedure of disassembly (and assembly in reversed order) is estimated to require about 60 days,[20] given adequate capabilities in transportation of the transfer casks to and from the hot cell facilities. This approach, which is suitable for the ST configuration, is driven by the high operational duty factor (~30%), which is required to achieve the engineering science testing mission of the CTF. This assembly and disassembly concept has been adopted in other ST-based fusion power plant concepts.[13,14,18]

IV. RECENT PROGRESS IN ST PLASMA SCIENCE KNOWLEDGE BASE FOR CTF

To achieve high duty factor operation in CTF, the fusion plasma science 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 = 1.2m/0.8m, κ = 2.8, I_p = 8.4-12.2 MA, I_{TF} = 15.3 MA, B_{T0} = 2.5 T, $\langle n_e \rangle$ = 0.69-1.0×10^{20} m^{-3}, β_T = 14-25%, β_N = 3.3-4.2, P_{AUX} = 38-49 MW, E_{NB} = 110-160 kV, P_{DT} = 72-144 MW to produce Γ_n = 4.4-8.8×10^{13} n/s/cm^2 at the outboard mid-plane wall. More detail of how these parameters are determined will be provided in Section V.

IV.A. Pressure and Current Limits

Recent studies of the global plasma stability beta limits in ST[21,22] and comparisons with the recent experimental results[4,15,16] 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 5 presents a summary of the toroidal beta values ($\beta_T \propto \langle p\rangle/B_{T0}^2$, where $\langle p\rangle$ = average plasmas pressure and B_{T0} = applied toroidal field at the plasma major radius R_0) achieved so far on NSTX without active feedback control. Also indicated are the parameter regimes of interest to the CTF under consideration (Section V) and the ST DEMO.[13,14]

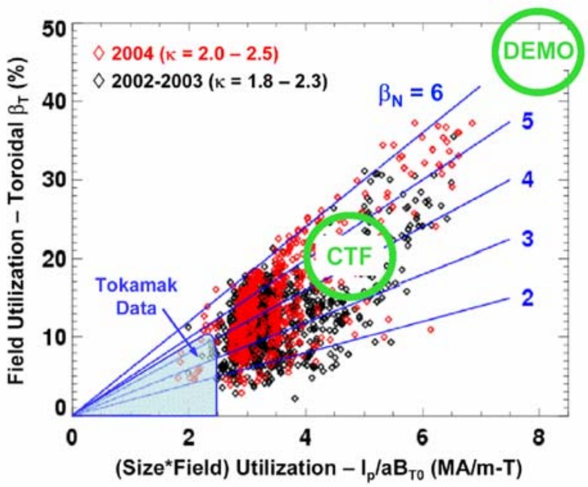

Fig. 5. Toroidal betas (β_T) as a function of the normalized plasma current ($I_N = I_p/aB_{T0}$) obtained so far on NSTX, relative to the regimes of interest to CTF, DEMO, and the normal aspect ratio tokamak.

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 $I_N q_{cyl}$ increases strongly with the inverse aspect ratio ($\varepsilon = a/R_0$) and κ:

$$I_N q_{cyl} = \frac{\pi\varepsilon\left(1+\kappa^2\right)}{10^6\,\mu_0}$$

It should be noted that for the large values of I_N in NSTX, data collected in Figure 5 are characterized by relatively high q_{cyl} (> 2) and relatively low plasma internal inductance.[21]

The normalized beta ($\beta_N = \beta_T a B_{T0}/I_p$) measures the plasma stability against pressure driven instabilities, which was first noted by Sykes[23] and Troyon[24] based on extensive stability computations. Relative to the normal aspect ratio tokamak data, β_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.[8,21] This together with the large I_N enables the high β_T.

In practical terms, the data in Figure 5 indicate high utilization of the applied magnetic field and plasma size, which translates to cost and size-effective ST today and in the future.

IV.B. Energy Confinement

Under neutral beam injection (NBI) alone, relatively long-pulse plasmas have been routinely obtained that have properties of interest to the CTF. The temperature, density, and rotation profiles of such plasmas are shown in Figure 6.

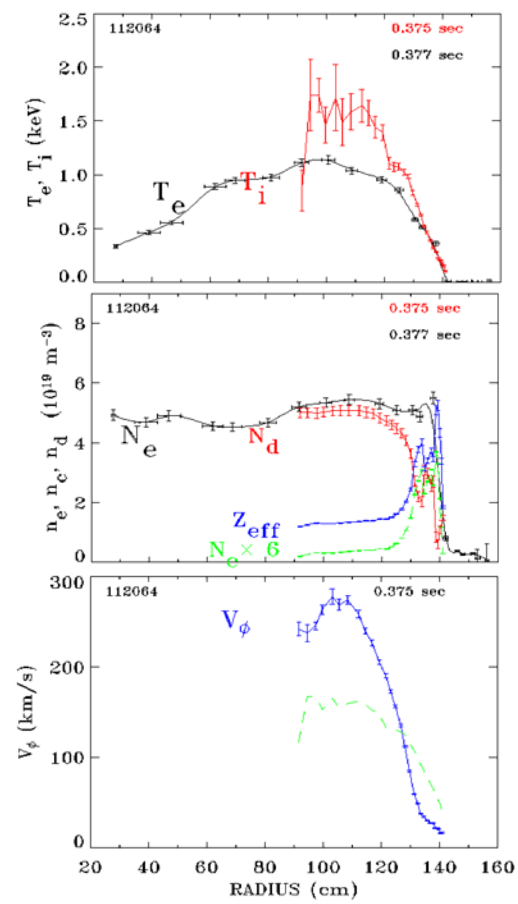

Fig. 6. Electron and ion temperature and density (T_e, T_i, n_e, n_D) and C-VI ion toroidal rotation (V_ϕ) profiles measured by Charge-Exchange Recombination Spectroscopy (CHERS)[25] and laser Thomson scattering[26] of a relatively long-pulse H-mode plasma driven by deuterium NBI at 6 MW and 90 kV.

This type of NSTX plasma is characterized by $T_i > T_e$ in the plasma core, relatively flat density profiles, and very hollow impurity (C-VI) profiles. Transport analysis using the TRANSP code[27] indicated[28] that, while the electron thermal diffusivity is large ($\chi_e \sim 10$ m^2/s), the ion thermal diffusivity can be at the neoclassical level ($\chi_i \sim \chi_{NC} \sim$ 1-2 m^2/s) in a substantial region ($\Delta R \geq 10$ cm) extending to R ~ 140 cm, where T_i and V_ϕ show steep gradients. These are similar to the description of an ion Internal Transport Barrier (iITB),[29] the verification of which is in progress and is expected to have important implications to the plasma science conditions of future ST devices including the CTF. The values of β_T for such plasmas in the range of 16 – 25% have been obtained for durations in which the plasma current can redistribute, during 2002 – 2004.

The resulting plasma thermal energy confinement times τ_E are still favorable compared with the standard ITER H-mode scaling[30] given below:

$$\tau_E^{98[y,2]} = \left[0.0562 M^{0.19} I_p^{0.93} R_0^{1.97} B_T^{0.15} \varepsilon^{0.58} \kappa_a^{0.78} \overline{n_{e19}}^{0.41} / P_{Tot}^{0.69}\right]$$

Here M is the average plasma ion mass, and P_{Tot} the total plasma heating power. Results of analysis of a number of such H-mode plasmas indicate that H-factors up to 1.3 can be obtained.[31] Since χ_i can be substantially different from χ_e, it is necessary to separate the energy confinement times of the electrons and ions in order to make basic projection to CTF. By using the measured profiles (Figure 6), accounting for energy transfer between electrons and ions, and subtracting the stored energy of the NBI ions, we arrive at the following approximate partition of the energy loss channels in the plasma core (Table II).

Table II. Estimates of plasma electron and ion energy confinement factors for an NBI driven H-mode plasma with relatively long pulse on NSTX (#109070)

Major radius, R_0 (m)	0.85
Plasma aspect ratio, A	1.4
Plasma elongation, κ	2
Applied toroidal field, B_{T0} (T)	0.45
Plasma current, I_p (MA)	0.8
Safety factor, q_{cyl}	3
Normalized beta, β_N	5.6
Global H-factor, H_{98}	1.28
Electron energy confinement H-factor, H_{98e}	0.7
Ion energy confinement H-factor, H_{98i}	4.0
Ion neoclassical energy confinement factor, H_{NC}	0.7

The global energy confinement time τ_E and the separate energy confinement times, τ_{Ee} and τ_{Ei}, are related 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 energies, respectively. The separate factors relative to $\tau_E^{98[y,2]}$ and τ_{NCi} will provide a basis for making projections to CTF.

IV.C. Plasma pressure gradient driven Current

An important component of sustained current arises from the plasma pressure gradient, somewhat akin to the thermal electric current observed in solid conductors. The commonly called "bootstrap" current I_{BS}[32] has been estimated to be substantial on NSTX owing to the relatively high β_N and q_{cyl}. Figure 7 shows the estimated bootstrap current fraction $f_{BS} = I_{BS}/I_p$ as a function of β_T.

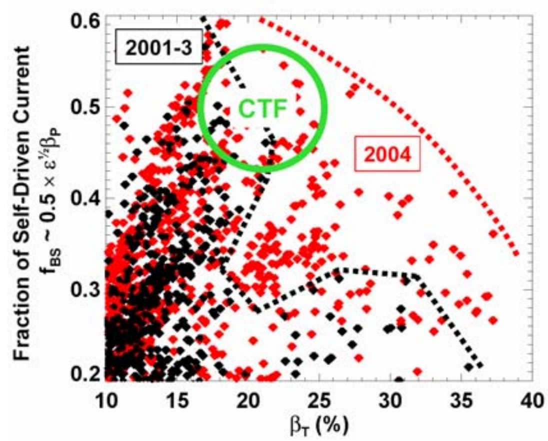

Fig. 7. Progress of bootstrap current fraction versus β_T on NSTX for 2001-2003 and 2004

The regime of interest to the CTF is located around $f_{BS} \sim 0.5$ and $\beta_T \sim 20\%$, which is within the range of parameters already produced in NSTX. In contrast, the regime of interest to the ST DEMO is near $f_{BS} \sim 0.9$ and $\beta_T \sim 50\%$, which indicates an important direction of longer term ST research in fusion plasma science.

IV.D. Sustainment of Driven Fusion Conditions

The H-mode plasmas on NSTX are aided by substantial f_{BS} 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-m^2 is approximately given by:[33]

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

It is seen from this that I_{NB} can be in the range of 0.1 – 0.4 MA on NSTX for the given values of P_{NB} up to 7 MW, n_{20} = 0.25 – 0.65, 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. Research on NSTX is continuing to understand the remaining inductive drive requirements and test operating scenarios for their elimination.[34]

To sustain a driven burn (Q ~ 2) 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/m^3. The normalized fusion product $\beta_N H_{89P}$ represents an equivalent plasma science condition that can be tested on NSTX. Here H_{89P} is the confinement time factor relative to the so-called "L-mode" plasma.[30] Recent progress of this test[35] is presented in Figure 8 in contrast with the CTF and ST DEMO requirements.

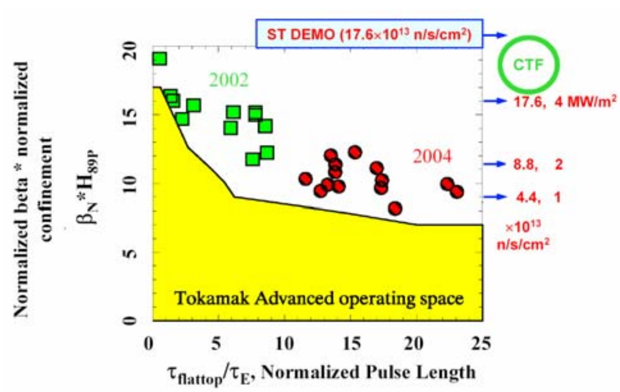

Fig. 8. Progress on NSTX in the normalized fusion product $\beta_N H_{89P}$ versus the plasma flattop time normalized to τ_E, in contrast with the equivalent conditions obtained in tokamaks so far. The flattop times have also reached beyond the plasma current redistribution times.

Also shown are the fusion neutron fluxes that can be produced in the CTF for a range of normalized fusion products. It is seen that the results on NSTX, where $\beta_N H_{89P} \geq 10$, is encouraging for the CTF baseline conditions of producing a fusion neutron flux Γ_n up to 8.8×10^{13} n/s/cm^2. To double Γ_n in CTF toward the level of DEMO in CTF would require a substantially higher $\beta_N H_{89P}$, which is nevertheless substantially below the ST DEMO requirement.[13]

IV.E. Determination 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 a moderate amount of RF for profile tailoring if necessary. Without assuming active feedback control of global MHD modes, it is further necessary to determine if the plasma profiles so determined would be stable. The TSC[36] and PEST-II[37] codes are used in these calculations, for the baseline case producing $\Gamma_n = 4.4\times10^{13}$ n/s/cm^2, at a density $\langle n_e\rangle = 0.69\times10^{20}$ m^{-3} and E_{NB} = 110 kV D^0. TFTR-type positive ion beam system[38] is assumed.

By spreading the neutral beam cross section vertically to span the height of the mid-plane access (Figure 1), the NBI driven current profiles in the CTF can be relatively broad, as shown in Figure 9.

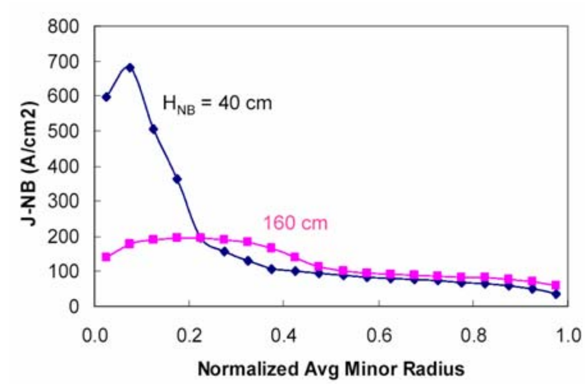

Fig. 9. NBI driven current profile J_{NB} for the baseline CTF operation, using 40-cm and 160-cm heights (H_{NB}) for the beam cross section.

Here P_{NB} = 30 MW is applied to produce a total I_{NB} ~ 5 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 (~ 175 cm) of the mid-plane radial access. 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 q0, high bootstrap current, and plasma stability.

Free-boundary equilibrium calculations (Figure 10) indicate that plasma elongations up to 3.2 can be produced with the distant PF coils for $\ell_i(1) < 0.5$, at $3.0 \leq \beta_N \leq 4.5$. In the case of inboard limited plasma during Phase-I operation (Tables III, IV), 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 reaches 0.45 only at the lower $\ell_i(1)$ values about 0.3, progressively decreasing to 0.2 as $\ell_i(1)$ rises to 0.5. Ideal MHD stability of the n=1 kink mode,

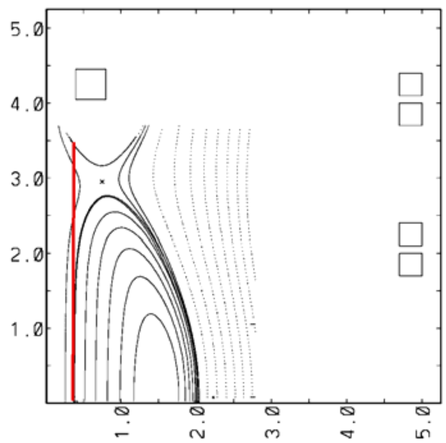

Fig. 10. Inboard limited CTF plasma with $\ell_i(1) = 0.25$, $\kappa = 3.2$, $\delta = 0.4$, $\beta_N = 4.0$, and $\beta_T = 20\%$.

without a wall, shows that the reference shape $\kappa = 3.2$ and $\delta = 0.4$ is stable in the target range of $3.0 \leq \beta_N \leq 4.5$, required for CTF, with $\ell_i(1) < 0.5$. Ideal MHD stability of lower elongations and triangularities are also examined and can be made stable.

The broad NB deposition and driven current profiles (Figure 9) are combined with bootstrap current and an assumed off-axis current produced by EBW to enable a range of $0.25 \leq \ell_i(1) \leq 0.5$. The consistency of the current profile, pressure profile, plasma shape, PF coil capability, and ideal MHD stability without active feedback, is being determined. The free-boundary evolution code TSC is used to examine the flattop plasma with extrapolated NSTX thermal diffusivities (Table II), and to examine the solenoid-free rampup requirements. Figure 11 shows theCTF plasma profiles for the Phase-I operation conditions indicated in Table III.

A point model approximation of the plasma ramp-up to steady state operation is also prepared[39] to assess the global plasma behavior and requirements of the CTF plasma. The model accounts for the plasma current circuit equations including the poloidal coil currents and the non-inductive currents from external current drive and internal bootstrap effect, plus 0-D plasma energy and particle balance. A representative result is provided in Figure 12, which shows that an appropriate combination of β rise, poloidal field coil induction, fueling, heating, and the external and internal driven currents can successfully bring a modest initial I_p of 100 kA to the full level (~10 MA), producing a full fusion power P_{DT} of ~300 MW.

V. CHOICES OF CTF PARAMETERS

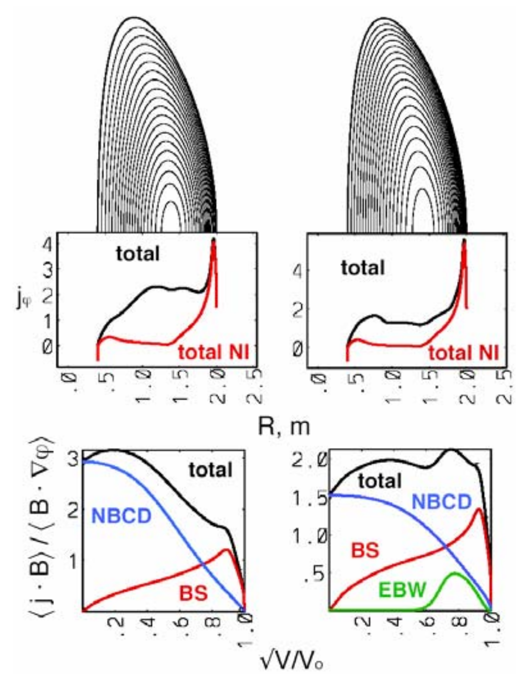

Fig. 11. CTF plasma current profiles calculated by the JSOLVE code for the steady-state TSC simulation. A profile with $\ell_i(1) = 0.5$ & $q_0 \sim 2$ can be maintained by I_{NB} and I_{BS} (left-hand side) using $P_{NB} = 30$ MW, while adding $I_{EBW} = 1$ MA would allow $\ell_i(1) = 0.25$ & $q_0 \sim 4$.

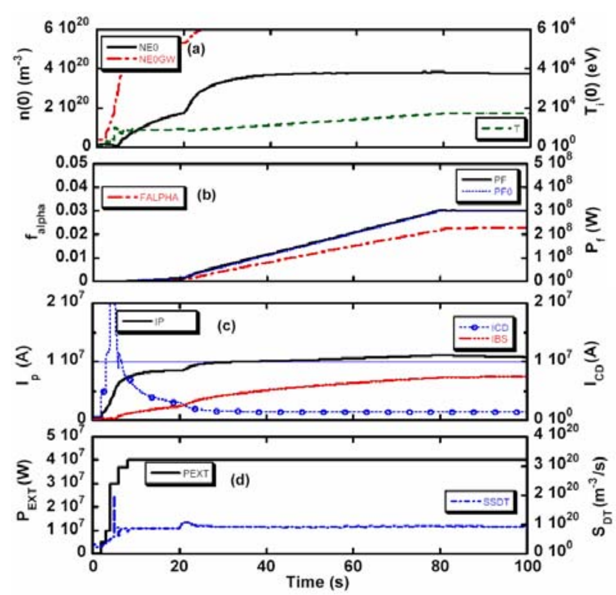

Fig. 12. Point model simulation of CTF plasma ramp-up to steady state operation, showing evolutions of (a) $n_e(0)$ and $T_i(0)$; (b) P_{DT} and fraction of α-ash f_α; (c) I_p, I_{CD}, and I_{BS}; and (d) P_{aux} and particle fueling rate S_{DT}.

"Systems Codes" have been developed and used[40] to estimate the major parameters and their tradeoffs of toroidal device designs. For ST devices, a new code[43] has been developed to capture in approximation the unique features of the ST plasma and device configuration for this purpose. The parametric survey is to minimize the total auxiliary power while producing a prescribed Γ_n for a given CTF device design (Section III), subjected to a set of physics and engineering limits.

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 IV:

- Electron and ion power balance accounting for heating, energy loss, and electron-ion power transfer;
- Fusion power from the slowing down beam ions and the thermal plasma assuming equal deuterium and tritium concentrations;
- Positive ion ($E_{NB} \leq 160$ kV) and negative ion ($E_{NB} >$ 160 kV) NBI energy species;
- Volume integration of NBI heating and drive current profiles, and fusion power using plasma profiles similar to the NSTX profiles shown in Figure 6;
- Plasma pressure from electrons, thermal ions, fast ions, alpha particles, and impurity ions;
- MHD pressure stability limits accounting for plasma geometry and profiles.[41,42]
- Bootstrap current, in conjunction with the NBI driven current, making up the total I_p;

Engineering features described in Section III are also modeled in approximation and included in the code.

- TF center leg conductor assumes Glidcop AL-25 material, σ=87% IACS, similar to the ARIES ST choice.[13] The shape of the center leg is shown in Figure 1. The water inlet temperature is 35^{o}C, flow velocity 10m/s, with resistive dissipation and nuclear heating included. The copper and water temperatures are limited to below 150^{o}C; the VonMises stresses are limited to 100Mpa;
- TF current returned through an aluminum outer shell with horizontal sections typically 1.0m thickness and vertical section 0.75m thick, which also serves as the vacuum boundary;
- Radial builds between the outer edge of the plasma and the TF return as follows:
  - SOL and gap 0.10m
  - Blanket 0.55m
  - Shield 0.7m
  - Gap 0.1m;
- Inboard-limited or double null divertor geometry with models for magnetic flux expansion as a function of aspect ratio. Radiation fraction from the core is set at 50% of total heating power. At the divertor it is adjusted to reach the allowable average peak flux at divertor = 15 MW/m^2;
- Allowable average peak flux at first wall = 1.0 MW/m^2;
- Fusion neutrons incident on the center leg, and NBI, diagnostic, and RF ports are lost;
- Fusion neutron flux distributed on wall according to the 1/r^2 from source;
- Local tritium breeding ratio = 1.25 for captured fusion neutrons;
- Algorithms included for thermal power conversion and electric power generation and consumption.

The Systems Code is implemented using EXCEL and use the non-linear optimizer SOLVER to find solutions. A typical set of independent variables adjusted by SOLVER include impurity radiation level, f_{GW}, β_{Ni}, β_{Ne}, T_{0i} , T_{0e}, 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 III summarizes the key parameters of the CTF assuming three levels of fusion neutron flux, designated as Phases I, II, and III, assuming ion and electron energy confinement times scale as the ITER H-mode.

Table III. Key science and engineering conditions for the CTF with R_0 = 1.2m, a = 0.8m, κ = 2.8, B_{T0} = 2.5T, I_{TF} = 15MA, $n_D = n_T$, H_{98e} = 0.7, and H_{98i} = 4.0, for Γ_n = 4.4, 8.8, and 17.6×10^{13} n/s/cm^2

Operation Phase	**I**	**II**	**III**
Γ_n (10^{13} n/s/cm^2)	**4.4**	**8.8**	**17.6**
Ip (MA)	9.1	12.8	16.1
q_{cyl}	4.2	3.0	2.4
β_N (%-m-T/MA)	3.1	3.9	5.0
β_T (%)	14	24	39
$\langle n_e \rangle$ (10^{20} /m^3)	0.70	1.0	1.5
n_{GW} (%)	16.4	16.8	20.3
$\langle T_i \rangle$ (keV)	20	22	21
$\langle T_e \rangle$ (keV)	8.1	10.7	12.6
Equivalent H_{98}	1.6	1.5	1.4
f_{BS} (%)	52	43	44
P_{NB+RF} (MW)	36	47	65
E_{NB} (kV) for D^0	112	160	247
P_{DT} (MW)	77	154	308
$P_{Beam-Plasma}/P_{DT}$ (%)	38	31	24
f_{Rad} (%) for $\Gamma_{Div} \leq 15$ MW/m^2	65	79	89
Achievable f_{BR} (%)	100	95	89

It is seen that the Phases I & II operation of CTF requires plasma conditions that are substantially within the well established limits in q_{cyl} ($\geq$2.3 for current driven mode stability), β_N ($\leq$5.6 for pressure-driven mode stability without assuming plasma rotation and conducting wall), and n_{GW} (≤1 for edge density stability). The modest plasma density in this case also allowed $\langle T_i \rangle / \langle T_e \rangle > 2$, leading to an apparent enhancement of H_{98} to > 1.5 without changing H_{98e} and H_{98i}. The required E_{NB} of ≤ 160 kV will permit the use of the TFTR-type positive-ion NBI system.[38] The modest density further leads to a substantial level of beam-plasma fusion fraction close to 40%.

As Γ_n is doubled in Phase-III operation, β_N is increased up to ~ 6, which may require active feedback control of the Resistive Wall Modes (RWMs),[46] while n_{GW} still remains modest. The density is increased so that $\langle T_i \rangle / \langle T_e \rangle \sim 1.6$ and $E_{NB} \sim 300$ kV, which will require JT-60U[55] and LHD-type[56] of negative-ion NBI system. In all three cases, f_{BS} remains in the range of 40% - 50%. As additional mid-plane ports are utilized by increased auxiliary heating power, the fraction of fusion neutrons captured $f_{Breeding}$ by tritium breeding blankets also decreases, from 95% to 89%, resulting in substantially increased rate of tritium consumption. Detailed neutron scattering and absorption analysis, accounting for the various materials in the chamber systems, will be required to determine the achievable $f_{Breeding}$ adequately.

It is worth noting that a substantial improvement in the CTF plasma conditions would result if the ion energy confinement can remain substantially neoclassical by maintaining the plasma profiles shown in Figure 6, so that $H_{NC} = 0.7$ can be maintained. This would effectively remove the ion energy lose channel in CTF. Table IV summarizes the major effects of this on the CTF parameters, in comparison with those given in Table III.

Table IV. Key science and engineering conditions for the CTF with $H_{NC} = 0.7$, instead of $H_{98i} = 4$

Operation Phase	**I**	**II**	**III**
Γn (10^{13} n/s/cm^2)	**4.4**	**8.8**	**17.6**
Ip (MA)	9.6	12.3	15.0
q_{cyl}	4.0	3.1	2.6
β_N (%-m-T/MA)	4.1	4.9	6.1
β_T (%)	19	30	45
$\langle n_e \rangle$ (10^{20} /m^3)	0.66	1.0	1.5
n_{GW} (%)	15	17	21
$\langle T_i \rangle$ (keV)	34	31	28
$\langle T_e \rangle$ (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)	77	153	308
$P_{Beam-Plasma}/P_{DT}$ (%)	23	20	15
f_{Rad} (%) for $\Gamma_{Div} \leq 15$ MW/m^2	49	71	85
Achievable f_{BR} (%)	100	100	95

It is seen that the fusion amplification Q can be increased from 2-5 in Table III to 4-8 in Table IV. The reduced auxiliary power results in increased β_N, which in turn leads to a higher f_{BS}. The reduced auxiliary power also eases the plasma radiation cooling requirement, makes available a mid-plane port for an addition test module, which increases the captured neutrons for tritium breeding. In principle a net tritium breeding fraction f_{TB} of ~100% can become possible if the fusion blankets are capable of a local tritium multiplication near 125%. These lead to a closer approach to the requirements of DEMO.[13,14] The science of energy confinement in NBI dominated ST plasmas therefore has high leverage in determining the CTF and DEMO performance.

These results show that the CTF has the potential for reliable plasma operations for Γ_n in the range of 4.4-8.8×10^{13} n/s/cm^2 without active feedback control of MHD modes. It further has the potential to achieve Γ_n up to 17.6×10^{13} n/s/cm^2, which is at the DEMO level, if active feedback control of field errors and RWMs could reliably allow access to plasma conditions of very high β_N and f_{BS}.

VI. FUSION PLASMA AND ENGINEERING SCIENCES KNOWLEDGE BASE FOR CTF

With worldwide preparation of the physics basis for ITER[44] and the anticipated ITER construction beginning in 2006, the burning plasma (Q ~ 10) science knowledge base, for tokamak with a large plasma size scale ($\rho_i{}^{*-1} = a/\rho_i \sim 10^3$) with $\langle T_i \rangle \sim 15$ keV, is expected to be completed in the 2020 time scale. Here ρ_i is the average plasma ion gyro-radius. This, coupled to progress in the USDOE strategic goal for fusion[5] 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 ~ 2-4) knowledge base for CTF, which is characterized by a moderate plasma size scale ($\rho_i{}^{*-1} \sim 10^2$) with $\langle T_i \rangle \sim 20$ keV. However, owing to the large extensions in the ST of the fusion plasma science regimes,[2] it is necessary to establish the extended knowledge base prior to the CTF operation, in the parameter ranges suggested in Section V. Further, solenoid-less initiation, ramp-up and sustainment of I_p is needed and uniquely important to CTF.

VI.A. Plasma Science Knowledge Base for ST CTF

Section IV presented several important advances in the ST plasma science that have guided the selection of the basic CTF parameters. Though the presented parameters of the CTF indicate relatively attractive cost-effectiveness, the results are subject to the rather unique plasma science regimes being investigated in today's ST plasmas.[4,15,16] The conditions, which define the intensive and extensive characteristics of the ST plasma, are provided in Table V. It is shown that the ST extends the plasma science regimes beyond those of ITER. It will therefore be important to answer the key questions of fusion plasma science that stem from the extended plasma regimes.

Table V suggests the following plasma physics questions of importance to the determination of the CTF plasma:

Table V. Fusion plasma science regimes revealed in NSTX and projected for CTF, compared to those of ITER

Plasma Science Conditions	NSTX	CTF	ITER
Toroidicity, $\varepsilon = a/R_0$	≤0.71	≤0.67	≤0.3
Elongation, κ	≤2.5	≤3.2	≤2
B_p/B_T in large-R region	~1	~1.5	~0.2
β_T/β_0 (central local β)	≤0.4/~1	≤0.45/~1	~0.02/0.06
Normalized size, $\rho_i{}^{*-1}$	~40	~80	~800
Alfvén Mach number, M_A	~0.3	~0.3	~0.01
Flow shearing rate (s^{-1})	~10^6	~10^6	Small
V_{NB} or $V_\alpha/V_{Alfvén}$	~4	~4	~4
Dielectric constant, ε_e ($= \omega_{pe}{}^2/\omega_{ce}{}^2$)	~10^2	~10	~1
Edge mirror ratio, M_B	≤4	≤4	≤2
Internal poloidal flux, ~$\ell_i R_0 I_p$ (MA-m)	~0.3	~4	~60

- How do the large ε, κ, B_p/B_T, and M_B at the plasma edge affect the properties of the Edge Localized Modes (ELMs)? Recent measurements of the ELM properties on NSTX[45] suggest a rich variety of ELMs can exist, with widely varied potential impact on the plasma core properties and the plasma-wall interactions.

- How does the large flow affect the plasma equilibrium and the global pressure-driven MHD modes? Recent measurements and modeling on NSTX[46] show that this leads to large modifications in the plasma equilibrium profiles and the properties of RWM.
- How do the strong shaping and the large flow affect the MHD mode locking as a function of the error field magnitude? Recent studies on NSTX[47] show that a substantial reduction in mode locking can result from adjustments of the field errors, leading to an increased range in density for stable operation of the plasma.
- How do the supra-Alfvénic fast ions affect the various Alfvén modes in the plasma, particularly at modest β values? Recent measurements and analysis on NSTX and comparisons with DIII-D studies[48] indicate that Compression Alfvén Mode (CAE)-like modes can also be excited at low field and NBI power on DIII-D. This suggests a strong dependence of such modes on the presence of supra-Alfvénic fast ions, which is anticipated in the CTF.
- How do strong flow and current ramp affect the electron energy confinement in low density L-mode plasmas? Can the L-mode plasmas with good confinement be sustained for long durations? Recent measurements and analysis[49] suggest that a strong magnetic shear reversal can be produced in NSTX to reduce χ_e toward the level of χ_i in the plasma core, in contrast to the conditions of NBI-driven H-mode (see Figure 6).
- How do the large dielectric constant ε_e and large particle trapping fraction (low aspect ratio) affect the edge conversion and propagation of the Electron Bernstein Wave (EBW)? Recent EBW emission measurements and current drive calculations[50] indicate potentially high current drive efficiency, taking advantage of the Ohkawa current.[51]
- How does the low aspect ratio affect the solenoid-less current initiation using a combination of RF electron heating and vertical field with a positive decay index? Recent results from JT-60U[52] and LATE[53] indicate that I_p/I_{TF} up to 0.3% and 10% can be produced this way, forming toroidal plasmas with $R_0/a \sim 4$ and 1.4, respectively.

Whereas present-day experiments[15,16] will shed much light on the answers to these questions, ST fusion plasma science knowledge base at the multiple MA level[54] will be needed. New experimental results, when adequately studied and understood, will help determine realistic conditions and requirements for reliable operations in CTF.

VI.B. Engineering Science Knowledge Base for CTF

Successful ITER plasma operations through 2020 are expected further to establish the fusion engineering science knowledge base for long pulse (~ 10^3 s) burning plasmas producing a fusion neutron wall flux $\Gamma_n \sim 2.6\times10^{13}$ n/s/cm^2. The systems used to heat, fuel, pump, and confine the ITER plasmas would establish the basis for the initial operation of CTF at $\Gamma_n \sim 4.4\times10^{13}$ n/s/cm^2. The relatively moderate E_{NB} determined in Section V suggests that present-day positive-ion[38] and negative-ion NBI techniques[55,56] need to be extended to steady state operations. However, the engineering science knowledge base for the water-cooled, single-turn, normal conducting center leg of the TF coil is needed and uniquely important to CTF.

The fusion plasma and engineering science conditions so provided in CTF, together with the full remote handling capabilities indicated in Section III, would introduce the reactor-like conditions in which all chamber systems can be tested effectively. Further, the test modules would be provided by the fusion engineering and technology 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 become the initial systems to be tested on CTF.

VII. CONCLUSIONS AND DISCUSSION

Our work, though preliminary in nature, has brought to light the following conclusions:

- The engineering science knowledge base to be established through the use of a CTF that produces a steady-state fusion neutron flux of Γ_n in the range of $4.4\text{-}17.6\times10^{13}$ n/s/cm^2, will bridge between the conditions to be achieved in ITER to those required by a DEMO that endures 60 dpa before scheduled maintenance biannually. The fusion engineering science conditions to be achieved by the CTF is identified and discussed in Section II.
- A much simplified, modest-size ($R_0 \sim 1.2$m) ST CTF configuration becomes possible, when a single-turn, normal conducting center leg of the TF coils are used in the absence of a central solenoid. Such a configuration is shown in Section III to allow full remote handling during the assembly and disassembly of all activated chamber systems, including mid-plane test modules, cylindrical test modules, divertor, center post, diagnostic module, RF module, and NBI module. Such an approach is deemed required to achieve the testing mission of the CTF, through the achievement of a full-performance duty cycle to 30%, which would be an order of magnitude increase beyond the anticipated ITER operation.
- Progress in the ST plasma science knowledge base in past years has indicated that the required plasma parameters in CTF can be produced in ST plasmas with normalized stability, confinement, and bootstrap current conditions already achieved in NSTX, in a regime substantially away from the known limits against reliable ST plasma operation. Key examples of such plasma conditions are summarized in Section IV. Though plasma flattop times have reached beyond the current redistribution times, numerical simulation of the CTF plasma conditions suggest that a combination of NBI and RF (likely EBW) heating and current drive at a total power level below 40 MW would provide the flexibility needed to achieve the full steady state conditions while producing the Phase-I level of Γ_n.
- The CTF fusion plasma and engineering science conditions are modeled in approximation in a systems optimization code to determine the landscape of the CTF designs. The results are presented in Section V, and show that a CTF with $R_0 = 1.2$m has the potential capability to deliver Γ_n in the range of $4.4\text{-}17.6\times10^{13}$ n/s/cm^2. For the lower half range of this flux, the normalized CTF plasma conditions are within the stability, confinement, and bootstrap current limits already produced in NSTX without active feedback control of field errors and MHD modes. The higher half of the range reaches the DEMO-level Γ_n, and will require research progress into ST plasma conditions achievable using active feedback control of field errors and MHD modes.
- While ITER is anticipated to establish the burning plasma science knowledge base, the ST plasma extends the plasma science regimes beyond those of ITER in several important topical areas. Key questions that address the scientific issues of the extended regime are discussed in Section VI. The scientific knowledge base for solenoid-less initiation, ramp-up, and sustainment of the ST plasma is identified as the most critical among the remaining physics issues of CTF and DEMO. Also identified as critically important is the fusion engineering science knowledge base for the center leg of the TF coil.

In view of these results, the following discussions have become appropriate:

- This study indicates that the CTF has high potential to produce reliably Γ_n levels that are 2-3 times those anticipated in ITER and two orders of magnitude in sustained operation. However, the level of fusion engineering science knowledge base to be established by ITER available in the 2020 time scale will likely refine the plasma operation conditions of the Phase-I level (Tables III and IV). The progress in fusion plasma science and that in fusion engineering science will therefore need to be advanced in tandem using the CTF; advances in one will motivate and require those in the other, eventually reaching the level of DEMO. A design with full remote handling assembly and disassembly will therefore be indispensable for this progress to be achieved in a timely 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 successfully. With a 2-year maintenance

cycle, a DEMO delivering 4 MW/m^2 flux and 75% duty factor would accumulate 60 dpa between maintenance. As a result, CTF with Phase II capability (Tables III and IV) and 30% duty factor would deliver in 10 years the engineering science knowledge base required 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 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 of interest to an effective development of Fusion Energy Sciences. The lower end would be a small fusion unit with $R_0 \leq 1$m producing reduced P_{DT} (~10 MW) and Γ_n (~10^{13} n/s/cm^2) for extended plasma and engineering science studies. The higher end could be a Pilot Plant[57] with $R_0 \sim 1.5$m capable of testing the integrated operation of fusion electricity production at substantial P_{DT} (~300 MW), assuming reliable ST plasma conditions similar to those assumed for the Phases I & II of the CTF.
- The ST allows extended fusion plasma science regimes and simplified configurations with reduced size. The potential benefits of this special combination are only beginning to be examined. 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 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; \$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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