Physics and Engineering Assessments of Spherical Torus Component Test Facility

Y.-K. M. Peng 1),[a] C. A. Neumeyer 2), P. J. Fogarty 1), C. Kessel 2), D. J. Strickler 1), P. Rutherford 2), D. Mikkelsen 2), T. W. Burgess 1), R. Bell 2), J. Menard 2), D. Gates 2), S. Sabbagh 3),[a] O. Mitarai 4), J. Schmidt 2), E. Synakowski 2), J. Tsai 1), L. Grisham 2), B. E. Nelson 1), E. T. Cheng 5), L. El-Guebaly 6)

1) Oak Ridge National Laboratory, UT-Battelle, Oak Ridge, Tennessee 37831, U.S.A.
2) Princeton Plasma Physics Laboratory, Princeton, New Jersey 08543, U.S.A.
3) Columbia University, New York, New York 10027, U.S.A.
4) Kyushu Tokai University, Kumamoto, Japan
5) TSI Research, Solano Beach, California 92075, U.S.A.
6) University of Wisconsin, Madison, Wisconsin 53706, U.S.A.
[a]on assignment at Princeton Plasma Physics Laboratory

e-mail contact of main author: mpeng@pppl.gov

Abstract. A broadly based study of the fusion engineering and plasma science conditions of a Component Test Facility (CTF), using the Spherical Torus or Spherical Tokamak (ST) configuration, 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. The desired chamber conditions can be provided by a CTF with $R_0 = 1.2$ m, A = 1.5, elongation ~ 3.2, $I_p \sim 9$ MA, $B_T \sim 2.5$ 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.

1. Introduction

Successful development of practical fusion energy will require research and development that combine fundamental and applied science. Component Test Facilities (CTF) [1] based on ST [2], aimed at advancing the required fusion engineering science [3] and encouraged by recent scientific progress [4], will necessarily entail similarly combined efforts. A recent USDOE Office of Science plan [5] identified a CTF that would be created following the International Thermonuclear Experimental Reactor (ITER) [6] construction to address this goal. Data from CTF will determine how full steady state fusion conditions affects plasma chamber materials and components, and limits their operating life. This will in turn establish the engineering science knowledge base needed to support a decision to build a demonstration power plant (DEMO) to produce net electrical output. The plan of this paper is as follows: the CTF fusion engineering science mission and required conditions are summarized in Section 2. The engineering design features to achieve these with an ST CTF is presented in Section 3. The most recent results from ST research, as a basis for the present CTF concept, are summarized in Section 4. The fusion plasma and engineering science landscape of the compact ST CTF, calculated using a new ST systems code, will be presented in Section 5. The CTF scientific database needs are identified in Section 6 in reference to the latest progress in ST research [7,8] . 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 6. The paper closes with a conclusion in Section 7 of the key results of the study, and a discussion of the scientific and engineering implications of CTF.

2. CTF Fusion Engineering Science Mission and Required Conditions

A comprehensive assessment of the required knowledge base of the fusion chamber systems were reported by Abdou et al [3]. The key ingredients of the full conditions can be restated in Table I in terms of engineering and material science, in comparison with the ITER design and those anticipated for a DEMO [9,10] that assumes a 2-year maintenance cycle with a fully remote-maintainable design (see, Section 3).

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.

Fusion Engineering Science Conditions	ITER	CTF	DEMO
14-MeV neutron flux through chamber surface, Γ_n (10^{13}n/s/cm^2)	~3.7	>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 diension 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
Integrated duty factor, F_D (%)	2.5	30	75

It is seen that CTF bridges the gap between the ITER and the DEMO chamber conditions in all aspects except in fusion neutron and neutron heat fluxes. There is thus a high value to enhance the CTF conditions toward those of DEMO by increasing these fluxes. ITER provides adequate conditions in the spatial scales of materials depth and transverse dimension 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.

3. A ST Design to Achieve CTF Mission

To achieve the CTF fusion engineering science conditions, including an integrated 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/Repair (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 [11]. 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.

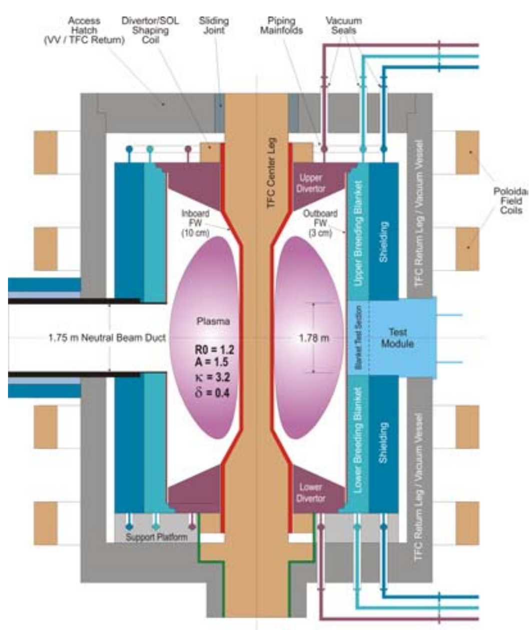

FIG. 1. Vertical cross section view of CTF for full remote handling of all chamber systems.

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 [12] 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 a "daisy-wheel" of 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.

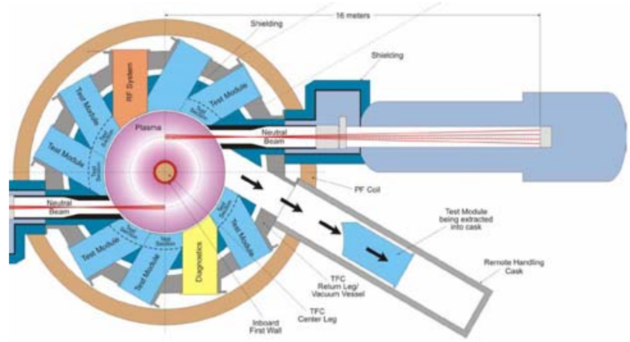

FIG. 2. Mid-plane view of a CTF configured for full remote handling of all chamber systems.

Other chamber systems would acquire vertical access for remote handling. Figure 3 depicts the arrangement that makes this possible. A sizable shielded maintenance enclosure could 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 200 metric tons. The chamber systems could be accessed vertically following hands-on evacuation and disconnection of all services from outside of the shield boundary of the CTF. A complete remote disassembly of all the chamber systems would become possible. The mid-plane modules, including the neutral beam liner, diagnostic systems and radiofrequency launchers would be removed horizontally to facilitate the disassembly. The entire procedure of disassembly (and assembly in reversed order), depicted in Figure 4, is estimated to require about 45 days [13], if adequate capabilities in transportation of the evacuated and shielded transfer casks and hot cell facilities are also provided. This approach, which is suitable for the ST configuration, would permit a high operational duty factor (~30%).

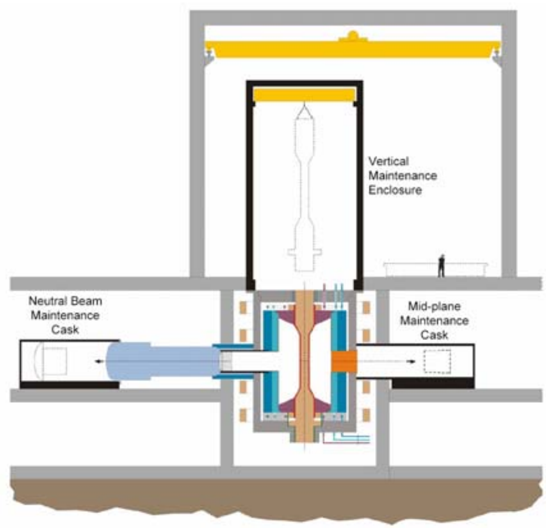

FIG. 3. Shielded maintenance cask systems are envisioned to allow horizontal remote replacement of mid-plane modules and the neutral beam systems, and vertical remote replacement of other chamber systems.

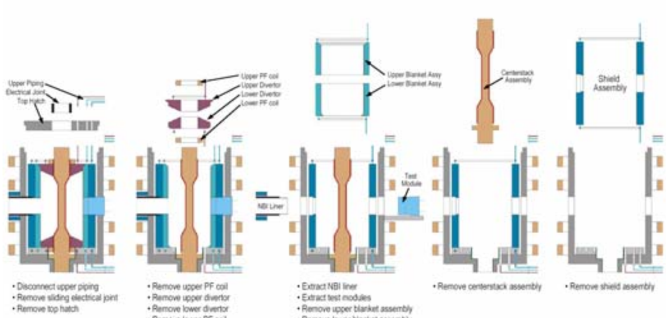

FIG. 4. Vertical remote disassembly procedure envisioned for CTF chamber systems.

4. Recent Progress is ST Plasma Science Knowledge Base for CTF

Recent studies of the global plasma stability beta limits in ST [14,15] and comparisons with the experimental results [4,7,8] have shed additional light on how a substantial range of plasma parameters of interest to the CTF can be produced while staying substantially below the plasma operational 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 control of field errors and MHD modes. Also indicated are the parameter regimes of interest to the CTF under consideration (Section 5) and the ST DEMO [9,10].

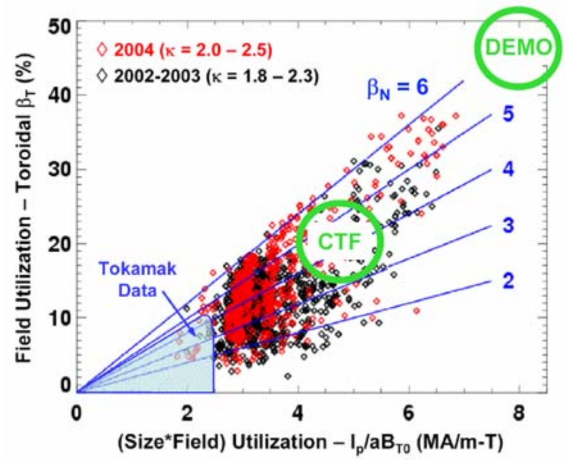

FIG. 5. Toroidal betas (β_T) and the normalized plasma current ($I_N = I_p/aB_{T0}$) obtained so far on NSTX, relative to the regimes of interest to CTF, DEMO, and the normal aspect ratio tokamak.

Under neutral beam injection (NBI) alone, relatively long-pulse plasmas have been routinely obtained that have properties of interest to the CTF. The plasma thermal energy confinement times τ_E compares favorably with the standard ITER H-mode scaling [16]. Results of analysis of a number of such H-mode plasmas indicate that H-factors up to 1.3 can be obtained [17]. Since χ_i can be substantially different from χ_e, it is necessary to separate the energy confinement times of the electrons and ions [18] in order to make projection to CTF. By using the measured profiles, accounting for energy transfer between electrons and ions, and subtracting the stored energy of the NBI ions, we arrive at an partition of the energy loss channels between the electrons and ions, respectively, with $H_{98e} \sim 0.7$ and $H_{98i} \sim 4.0$.

The "bootstrap" current I_{BS} [19] has been estimated to be substantial on NSTX owing to the relatively high β_N and q_{cyl}. Figure 6 shows a summary of the results in bootstrap current fraction $f_{BS} = I_{BS}/I_p$ and β_T. The regime of interest to the CTF is near $f_{BS} \sim 0.5$ and $\beta_T \sim 20\%$, 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 $\beta_T \sim 50\%$.

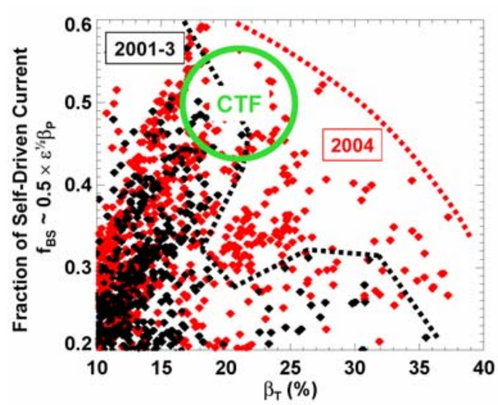

FIG. 6. Progress of bootstrap current fraction versus β_T on NSTX for 2001-2003 and 2004.

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 condition that can be tested on NSTX. Here H_{89P} is the confinement time factor relative to the so-called "L-mode" plasma [16]. Recent progress in this direction [20] is presented in Figure 7 together with the CTF requirements. 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 $\beta_N H_{89P}$ = 9-11, can be projected to fusion neutron fluxes Γ_n up to 8.9×10^{13} n/s/cm^2 in the CTF. To double Γ_n in CTF toward the level of DEMO in CTF would require a substantially higher $\beta_N H_{89P}$ (= 16).

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 if necessary for profile tailoring. 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 [21] and PEST2 [22] 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, using a TFTR-type positive ion beam system [23].

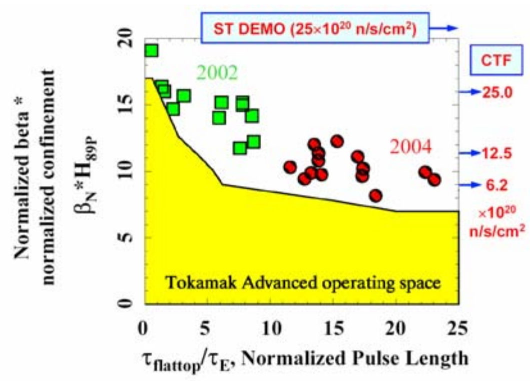

FIG. 7. Progress on NSTX in $\beta_N H_{89P}$ and the plasma flattop time normalized to τ_E, compared with the equivalent conditions on tokamaks.

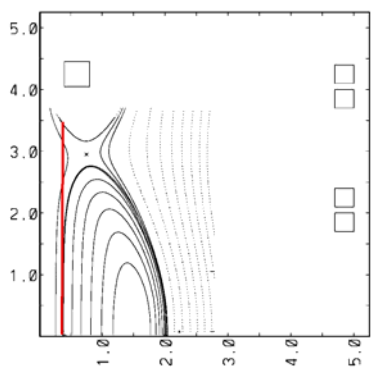

FIG. 8. Inboard limited CTF plasma with $\ell_i(1) = 0.25$, $\kappa = 3.2$, $\delta = 0.4$, $\beta_N = 4.0$, and $\beta_T = 20\%$.

Free-boundary equilibrium calculations (Figure 8) indicate that plasma elongations up to 3.2 can be produced using the distant PF coils in CTF for $\ell_i(1) < 0.5$, at $3.0 \le \beta_N \le 4.5$. In the case of inboard limited plasmas during Phase-I operation (see, Table II), 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 calculations of the n=1 kink mode, without a wall, show that the reference shape $\kappa = 3.2$ and $\delta = 0.4$ is stable in the target range of $3.0 \le \beta_N \le 4.5$ required for CTF, with $\ell_i(1) < 0.5$. Ideal MHD stability of lower elongations and triangularities are also examined. Broad NB deposition and driven current profiles are combined with bootstrap current and an assumed off-axis current produced by EBW to enable a range of $0.25 \le \ell_i(1) \le 0.5$. 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, and to examine the solenoid-free ramp-up requirements. Figure 9 shows the CTF plasma profiles for the Phase-I operation conditions indicated in Table II.

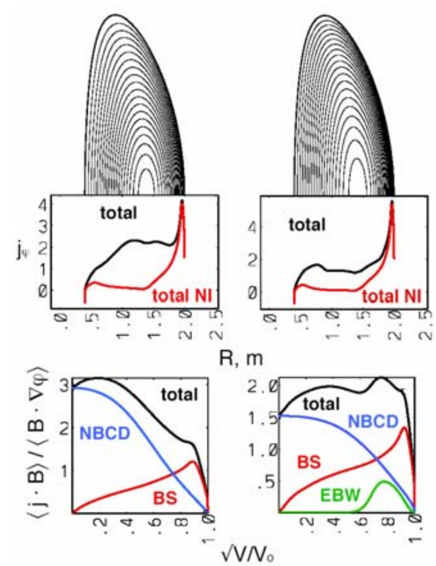

FIG. 9. CTF plasma current profiles calculated by the JSOLVE code for the steady-state TSC simulation. Profiles with $\ell_i(1) = 0.5$ & $q_0 \sim 2$ can be maintained by I_{NB} and I_{BS} (left-hand side), while adding $I_{EBW} = 1$ MA would allow those with $\ell_i(1) = 0.25$ & $q_0 \sim 4$.

5. Choices of CTF Parameters

A new ST systems code [26] has been developed to capture the properties of the ST plasma and the CTF device configuration, as guided by the engineering features and the physics results summarized in Sections 3 & 4. Solutions are constrained by power balance and various physics and engineering limits. Table II provides the key CTF parameters determined by the code, covering three levels of fusion neutron flux, as Phases I, II, and III. The results show that 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$4.5 for pressure-driven mode stability without assuming active feedback mode control), and n_{GW} ($\leq$1 for edge density stability). The modest plasma density in this case also allowed $\langle T_i \rangle / \langle T_e \rangle \sim$ 2-3, leading to an equivalent H_{98} of 1.5-1.6 while maintaining H_{98e} = 0.7 and H_{98i} = 4.0. The required E_{NB} of 110-160 kV will permit the use of the TFTR-type positive-ion NBI system [23]. The modest density further leads to a substantial level of beam-plasma fusion fraction >30%. As Γ_n is doubled again in Phase-III operation, β_N is increased to ~ 5, which will most likely require active feedback control of the field errors and the Resistive Wall Modes (RWMs) [27,28], while n_{GW} still remains modest. The density is increased so that E_{NB} ~ 250 kV, requiring JT-60U [24] and LHD-type [25] of negative-ion NBI system. In all three cases, f_{BS} remains in the range of 40% - 50%.

TABLE II. Key science and engineering conditions for the CTF with R_0 = 1.2m, a = 0.8m, κ = 3.2, B_{T0} = 2.5T, I_{TF} = 15MA, H_{98e} = 0.7, and H_{98i} = 4.0, for Γ_n = 4.4, 8.9, and 18×10^{13} n/s/cm^2.

Operation Phase	**I**	**II**	**III**
Γ_n (10^{13} n/s/cm^2)	**4.4**	**8.9**	**18**
Fusion neutron heat flux (W/cm^2)	100	200	400
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	110	160	250
P_{DT} (MW)	72	144	288
$P_{Beam\text{-}Plasma}/P_{DT}$ (%)	38	31	24
f_{Rad} (%) for $\Gamma_{Div} \leq$ 15 MW/m^2	65	79	89

6. Fusion Plasma and Engineering Sciences Knowledge Base for CTF

With worldwide preparation of the physics basis for ITER [29] and the anticipated ITER construction beginning in 2006, the burning plasma (Q ~ 10) science base is expected to be completed in the 2020 time scale, thereby establishing the low-Q (~2-4) database for CTF. However, owing to the large extensions (Table III) in the ST of the fusion plasma science regimes [2], it is necessary to establish the ST knowledge base prior to CTF operation, particularly on supra-Alfvénic fusion α particle physics [30].

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

Plasma Science Conditions	**NSTX**	**CTF**	**ITER**
Toroidicity, $\varepsilon = a/R_0$	$\leq$0.71	$\leq$0.67	$\leq$0.3
Elongation, κ	$\leq$2.5	$\leq$3	$\leq$2
B_p/B_T in large-R region	~1	~1.5	~0.2
Toroidal beta, β_T	$\leq$0.4	$\leq$0.4	~0.02
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	~1
Dielectric constant, $\omega_{pe}{}^2/\omega_{ce}{}^2$	~10^2	~10	~1
Edge mirror ratio, M_B	$\leq$4	$\leq$4	$\leq$2
Internal flux, ~$\ell_i R_0 I_p$ (MA-m)	~0.3	~4	~60

Table III also identifies the plasma physics issues of interest to the CTF plasma. Potential contributions from a ST experiment at multiple MAs were discussed earlier in [31]. More recent progress in understanding the new ST physics are available on electron energy confinement [32], EBW heating and current drive [33], and solenoid-free initiation and ramp-up of plasma current [34,35]. An extended description of the CTF and its physics basis will be submitted for publication under separate cover [36].

ITER plasma operations through 2020 are expected also to establish the engineering science 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 [23] and negative-ion NBI techniques [24,25] can be extended to support the CTF steady state operations. However, the engineering science base for the water-cooled, single-turn, normal conducting, copper alloy center leg of the TF coil is needed and uniquely important to CTF.

7. Conclusions and Discussion

In this work, the fusion engineering science conditions to be achieved by the CTF are described. A highly simplified, modest-size ($R_0 \sim 1.2$m) ST CTF configuration is shown to be possible and to allow full remote handling assembly and disassembly of all activated chamber systems. Such an approach is deemed required by the testing mission of the CTF, achieving an integrated duty factor of 30%, which would be an order of magnitude increase beyond ITER. Rapid progress in the ST plasma science knowledge base in recent years has provided important new information for the selection of a robust CTF plasma physics regime that is within the known physics limits. A systems code analysis shows that such a CTF has the potential capability to deliver Γ_n in the range of 4.4-18$\times10^{13}$ n/s/cm^2. For the lower half of this range ($\Gamma_n \leq 8.9\times10^{13}$ n/s/cm^2), the required CTF plasma conditions are substantially below the physics limits and are readily produced in NSTX without active feedback control of field errors and MHD modes. 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 requirements of CTF. Also identified as critically important is the fusion engineering science knowledge base for the center leg of the TF coil.

It is important to note that progress in fusion plasma and engineering sciences encompassing this range of Γ_n using the CTF will need to be made in concert. A design with full remote handling assembly and disassembly will therefore be indispensable in achieving this progress in a deliberate manner. The results further suggest that a wider range of parameters and performance of CTF would be possible. The lower end would be a smaller fusion unit with $R_0 < 1$ m producing modest P_{DT} (~10 MW) and Γ_n (~0.6×10^{13} n/s/cm^2). The higher end could be a Pilot Plant [37] with $R_0 \sim 1.5$m, capable of testing the integrated operation of fusion electricity production at substantial P_{DT} (~300 MW), while still remaining within the robust ST plasma regime. The CTF parameters presented in Section 5 result from assuming H_{98e} = 0.7 and H_{98i} = 4 relative to the ITER H-mode scaling. In the event that the ion thermal conductivity in ST using large tangential NBI power can approach the neoclassical level, the projected CTF plasma conditions are likely to improve further, suggesting the importance of this scientific question. With a 2-year maintenance cycle, a DEMO capable of full remote handling of all chamber systems would deliver 4 MW/m^2 flux at 75% duty factor, to an accumulated dose of 60 dpa between maintenance. Material testing to this level will be needed during the next three decades to support the effort to deliver fusion electricity via DEMO. Finally, the constructed cost for the CTF capable of Phase-I operation is estimated, by scaling from those of the major systems of ITER capable of Phase-I operation, is of the order of $1.05B in 2002 dollars, without contingency.

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