Q1: What are the essential features of the device that would fulfill the ST goal? Note that the Panel at present considers the CTF concept as a set of integrated capabilities that are useful as reference information.

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The essential features of the device that would fulfill a full implementation of the ST goal are described below, using existing assessments of CTF as example. Also addressed here are the issues of magnets and remote handling, which are part of Q3b and Q4c, respectively, and are included here as essential features. These features of the device are likely to survive updates of the physics and engineering design assumptions wehn key scientific and technical questions regarding the ST goal are resolved.

The essential features of a D-D implementation to achieve a part of the ST goal that deals with "Taming the Plasma Material Interface," a separate input could be provided if requested by the FESAC TAP.

Availability/Duty Factor

The primary issue for the CTF example, similar to all future D-T fusion device, is availability/duty factor. We define duty factor here as the ratio of the mean time between failure (MTBF) to the sum of the MTBF and the mean time to repair/replace (MTTR). This clearly requires both high reliability and an effective maintenance scheme. Reliability/duty factor can be enhanced by simplifying design solutions, adequate R&D, testing, and prototyping of those design solutions, and adequate design margins. Maintenance can be enhanced by minimizing the activities required in-situ. This can be done by providing a modular design with simplified removal and replacement of modules that are repaired or refurbished away from and independently of the device itself. High maintainability enables fast growth of reliability by accelerating commissioning and repair.

Magnets

For a conventional tokamak, reliability of the coil set is problematic, because replacement of a toroidal or poloidal field coil is extremely difficult. Failures are almost always electrical in nature, usually brought on by coolant leaks. If the coil set is superconducting, high voltages can appear during a coil quench and cause failure. Significant shielding (1+ m) is required to protect insulation and limit nuclear heating. For the CTF example uses a single-turn TF coil; the voltage across the coil set is low (10-15 V). The issues, of course, include the design of the high current power supply system and feeder bus, the high current electrical joints between the central core and the return legs, cooling of the central core, and radiation damage. There is also an option of solenoid wrapped around the central TF leg to assist startup.

The magnet issues include:

- The high current (~10 MA) power supplies and feeder bus system is not a feasibility problem, but more of an optimization problem. Although it is difficult, the equipment is remote from the device core and reliability issues can be solved with adequate redundancy. The low voltage requires current control and feedback to balance the current among the feeds (~1MA) and in the return legs of the TF coil. Dissipation in feeds must be minimized. Recent progress in high-temperature superconductor power lines (0.6-GW, 140-kV, 2000-ft line in Long Island) encourages the consideration of this option for feeds. The option of homopolar generator also deserves consideration in defining potentially high-leverage R&D.

- The high current electrical joints (~10 MA through a ~1m diameter interface) in between the center core and the return legs will probably require a sliding interface. Standard mechanical sliding joints should be evaluated for this purpose. Another option is to use a liquid metal joint, which will require adequate seals and be configured such that the Lorentz forces on the liquid metal tend to trap it rather than expel it from the system. Clearly this aspect of the design will require rigorous prototyping and testing at full parameters.
- The current density in the central Cu conductor is expected to be high (e.g., ~5kA/cm^2, ~150W/cm^3 in Glidcop). Cooling of the central core will require careful optimization of passages to accommodate resistive power and nuclear heating (~20W/cm^3 at surface), while considering effects like corrosion, degradation of conductivity, etc. R&D is needed on construction techniques.
- Radiation damage to the central TF core will include hardening/embrittlement as well as reduction in electrical conductivity. These effects are well-known based on irradiation studies using fission nuclear environment, which indicated adequate lifetime. (See, ST community document.)
- The startup solenoid is assumed to be a thin-layer MIC design wrapped around the TF center post. This multi-turn coil will have relatively high voltage compared to the TF coil. However, since it is only used during the early phases of current ramp, the insulation does not have to function under neutron irradiation, so radiation assisted conductivity is not an issue. Degradation of the insulation due to radiation has been quantified up to 10 dpa in fission environment (See, ST community document). The MIC design using ceramic powder is expected to be the best solution for accommodating such radiation damage effects as swelling and embrittlement. The performance of this coil will require careful design of the cooling and first wall protection, and it will almost certainly have to survive relatively high temperatures during the burn phase. For this reason helium may be the best coolant option. Preliminary estimates indicate that a 1-cm thick solenoid (30% Cu, 50% He, 10% ferric steel, 10% MgO) provides ~0.4 Wb, capable of inducing ~0.5 MA plasma current under RF assist. Up to 2 cm is expected to be acceptable without major impact on the ST goal device.

<u>Divertor</u>

Another issue, related to the low aspect ratio of the ST, is the high power density on the divertor. To address this issue and as an option, a novel scheme called a "super-X" divertor may be employed to expand the diverted flux surface and lower the power density. This feature has not been included in the existing conceptual layouts, but should be added next. Initial estimates indicate that the divertor heat flux would be reduced by a factor of 5-6 from a conventional design, due to a combination of SOL area expansion, longer connection length, and increased radiation cooling. The configuration has the added advantage of increased space for nuclear shielding. Another R&D option would be power and particle control using liquid metal in a lower single-null divertor. (see, ST community document.)

Disruptions

Finally, as with all devices using plasma current to obtain rotational transform, there is the issue of plasma disruptions. Since the CTF example is estimated to have very high ideal beta limits (β_T up to 35-40%), it is possible to reduce disruptions by operating well below this limit (such as β_T ~18%, β_N ~3.8, q_{cyl} ~ 3.7). In addition, the toroidal peaking factor for the halo currents has been shown on MAST to be much lower and symmetric in the ST configuration, lowering mechanical loading and peak heat deposition due to disruptions. Due to the critical impact of disruptions on the reliability of the plasma operation, stability control to minimize disruptions under the conditions of substantial margins becomes an R&D of high leverage. (See, "2-pager" for Q4b.)

Maintainability

With respect to maintainability, the CTF example (see Figure) is based on a completely modular design, with full, vertical remote removal and replacement of the central core, the breeding blankets, the shield, and the PF coils using cranes from above the machine. The vacuum boundary is integrated with the TF return conductor to simplify the maintenance configuration. This approach would expedite remote handling, using linear motion of large components and allowing all welds to locate external to the shield boundary. The test blankets are installed in the tangential (which is nearly perpendicular due to low aspect ratio), equatorial ports, and removed and replaced radially without interference between adjacent modules. No components are intended to be repaired or modified in place. Instead, all scheduled or unscheduled maintenance uses a common scheme of removal of the old component and replacement with a spare, new component. Extensive hot cell laboratories would be built to enable investigation of the replaced modules and R&D to establish the knowledge base to harness fusion power. Since the ST is a compact device with components of relatively manageable sizes, it is necessary to have replacements on hand. This concept greatly simplifies tooling and reduces maintenance times (MTTR) to maximize the time available for operation. The design and performance of the remote maintenance equipment will be driven by the duty factor goal of 30%.

Design Assumptions and Parameter Choices

Design choices for the CTF example were made using a "systems code" that accounts for physics and engineering design requirements and constraints while achieving a figure of merit, such as minimum external power required. Such a systems code was developed [1] and includes the following assumptions.

Physics:

- Fraction of the beta limit, which is a function of aspect ratio, kappa, and q, accounting for the fast ion component
- Bootstrap current fraction and lower bound of Greenwald fraction of density
- NBI power deposition profile as function of n and T, which assume powers of parabolic profiles
- Thermal ion fusion plus beam-target fusion heating and neutron source profiles
- Separate ion (~neoclassical) and electron (~ITER H-mode) power balance, coupled through electron-ion thermal equilibration
- Global energy confinement time constrained by HH $\leq$ 1.5.

- Ion and electron heating by NBI and fusion alpha
- Radiation losses, assuming 1% Oxygen in plasma
- NBI heating and current drive for sustainment accounting for bootstrap current

Engineering:

- Radial build: TF conductor, thin solenoid (to be added), gap, inboard first wall & tile, inboard SOL, plasma, outboard SOL, mid-plane test module, shield, TFC return conductor/vacuum boundary, etc.
- TF coil stress and heating including flared TF center column toward the ends
- PF coil requirements
- TF coil, PF coil, NBI, and balance of plant electrical input power
- Distributed neutron wall loading including shadow effect of center stack
- Blanket coverage accounting for geometry for tritium breeding and for test blanket modules (TBM) after accounting for NBI, RF and diagnostic port requirements, and assuming the TBM height is proportional to the plasma height
- Divertor and first wall heat loads handled within a relative large divertor module, assuming double-null
- Nearly full tritium self-sufficiency assuming successful performance of the tritium breeding and the test blankets

A non-linear optimizer is used in the systems code to solve for design points that satisfy all physics and engineering constraints while meeting a set of performance criteria and maximizing or minimizing a figure of merit which can be a function of one or more variables. For the example, the typical performance requirement is average neutron wall loading (e.g. $1MW/m^2$) on the TBM with a minimum TBM area of $10m^2$. The typical optimization function is chosen to be the auxiliary heating and current drive power P_{aux}. Minimization of P_{aux} tends to lead to the most compact and electrically efficient design.

Scans in aspect ratio suggest that A=1.5 yields the most compact (smallest R_0) and electrically efficient design. At lower A the stress and heating in the TF center column would become limiting and forces a larger machine, whereas at higher A, R_0 increases to ensure minimum TBM area as the plasma elongation becomes lower. These results suggest that a $1MW/m^2$, $10m^2$ TBM requirement can be met by a machine as small as R_0=1.2m while limiting the total β_T to 18%, requiring a total electric power input of ~250MW (see Table). This conceptual choice provides the opportunity to deliver $2MW/m^2$ at the TBM, if β_T of 28% is reliably achieved. This device could be operated in DD with ~3MA current at minimal fusion powers, for example, to commission key systems such as the divertors, remote handling equipment, shielding integrity, etc.

[1] “Spherical Torus Design Point Studies”, C. Neumeyer, Y-K Peng, C. Kessel, P. Rutherford, Princeton Plasma Physics Laboratory Report PPPL-4165, June 2006.

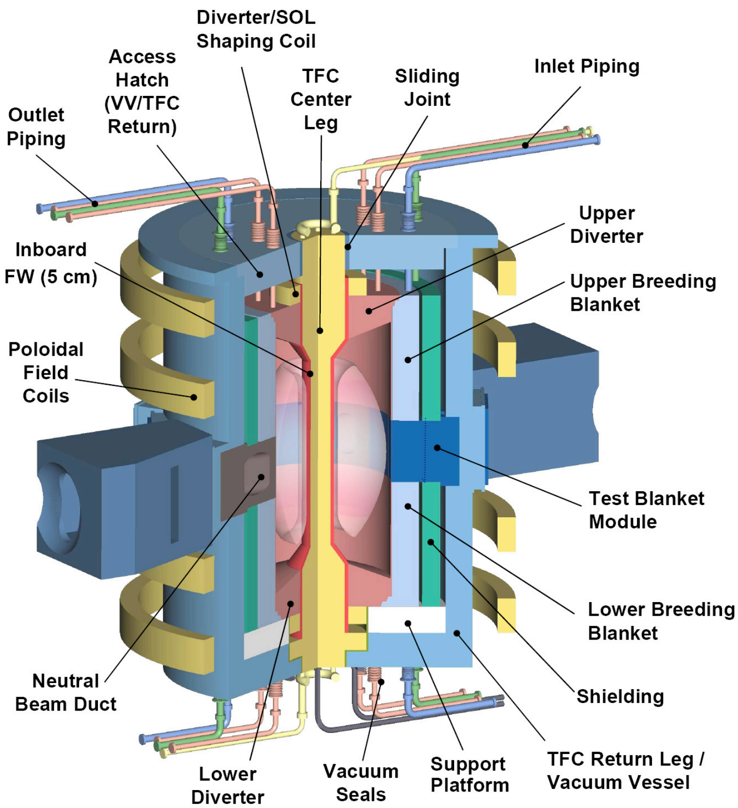

Figure. Example schematic concept of a modularized configuration for a full implementation of the ST goal.

W_L [MW/m^2]	**0.1**	**1.0**	**2.0**
R0 [m]		**1.20**	
A		**1.50**	
kappa		**3.07**	
qcyl	4.6	**3.7**	3.0
Bt [T]	1.13	**2.18**	
Ip [MA]	3.4	**8.2**	10.1
Beta_N	**3.8**		5.9
Beta_T	0.14	**0.18**	0.28
n_e [10^{20}/m^3]	0.43	**1.05**	1.28
f_{BS}	0.58	**0.49**	0.50
T_{avgi} [keV]	5.4	**10.3**	13.3
T_{avge} [keV]	3.1	**6.8**	8.1
HH98		**1.5**	
Q	0.50	**2.5**	3.5
P_{aux-CD} [MW]	15	**31**	43
E_{NB} [keV]	100	**239**	294
P_{Fusion} [MW]	7.5	**75**	150
T M height [m]		**1.64**	
T M area [m^2]		**14**	
Blanket A [m^2]		**66**	
$F_{n-capture}$		**0.76**	

Table. Approximate parameters estimated for this example.