Prospects for pilot plants based on the tokamak, spherical tokamak, and stellarator*

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Abstract - A potentially attractive next-step towards fusion commercialization is a pilot plant, i.e., a device ultimately capable of small net electricity production in as compact a facility as possible and in a configuration scalable to a full-size power plant. A key capability for a pilot plant program is the production of high neutron fluence enabling fusion nuclear science and technology (FNST) research. It is found that for physics and technology assumptions between those assumed for ITER and nth-of-a-kind fusion power plant, it is possible to provide FNST-relevant neutron wall loading in pilot devices. Thus, it may be possible to utilize a single facility to perform FNST research utilizing reactor-relevant plasma, blanket, coil, and auxiliary systems and maintenance schemes while also targeting net electricity production. In this paper three configurations for a pilot plant are considered: the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS). A range of configuration issues are considered including: radial build and blanket design, magnet systems, maintenance schemes, tritium consumption and self-sufficiency, physics scenarios, and a brief assessment of research needs for the configurations.

1. Overview

Recent studies in the United States [1] and European Union [2] have identified scientific and technological gaps that need to be closed to construct and operate a magnetic fusion power plant following ITER. A potentially attractive next-step towards fusion commercialization is a pilot plant, i.e., a device which produces a small amount of net electricity as quickly as possible and in as small a facility as possible in a configuration directly scalable to a power plant [3, 4, 5]. The pilot plant approach could accelerate the commercialization of magnetic fusion by targeting electricity break-even while also carrying forward a high neutron fluence fusion nuclear science and technology (FNST) and component testing mission needed to ultimately achieve high availability in fusion systems. This paper studies three configurations for a pilot plant: the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS). These configurations are considered because: the

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tokamak presently has the most well-developed physics basis, the ST offers the potential for simplified maintenance, and the CS offers disruption-free operation with low recirculating power. Overall, initial analysis indicates that the CS and AT are the most energy efficient electrically. Compared to the ARIES [6] series of power plant designs, preliminary scaling studies indicate that with similar engineering assumptions, the fusion power in a pilot plant with small net electricity production is approximately 20-30% of that of a power plant sized to produce 1GWe, and the major radius and neutron wall loading are approximately 0.6 times that of a full-scale power plant. Substantial peak outboard neutron wall loading of 2-5 MW/m^2 is achievable in the pilot plants studied indicating such devices are indeed suitable for consideration as FNST development devices [7]. The pilot plant devices investigated here are approximately 1.5 times larger in linear dimension than proposed ST [8, 9] and tokamak [10] facilities designed to achieve similar neutron wall loadings but without consideration of electricity production or extrapolation of the configuration to a power plant. The analysis, design, and mission considerations for pilot plants are described below.

2. Engineering efficiency analysis

The overall pilot plant engineering efficiency Q_{eng} is defined as the ratio of electrical power produced to electrical power consumed and can be expressed as:

$$Q_{eng} = \frac{\eta_{th}\eta_{aux}Q(4M_n + 1 + 5/Q + 5P_{pump}/P_{fus})}{5(1 + \eta_{aux}Q(P_{pump} + P_{sub} + P_{coils} + P_{control})/P_{fus})} \quad (1)$$

where η_{th} = thermal conversion efficiency, η_{aux} = auxiliary power wall plug efficiency, P_{fus} = total DT fusion power, P_{aux} = auxiliary power for heating and current-drive, $Q = P_{fus}/P_{aux}$, M_n = neutron energy multiplier, P_{th} = thermal power = $M_nP_n + P_\alpha + P_{aux}$, P_{pump} = coolant pumping power, P_{sub} = subsystems power, P_{coils} = power lost in normally conducting coils, and $P_{control}$ = power used in plasma or plant control that is not included in P_{aux}. Equation 1 illustrates that the leading terms in the engineering efficiency Q_{eng} involve a combination of technology and physics performance metrics. In particular, Q_{eng} depends to leading order on the thermal conversion and auxiliary system wall-plug efficiencies (η_{th} and η_{aux}) and the fusion gain Q. In this analysis, the value of η_{th} is varied to assess the impact on device size, a constant $\eta_{aux} = 0.4$ is assumed, the normalized current drive (CD) efficiency $\eta_{CD} = I_{CD}R_0n_e/P_{CD} = 0.3 \times 10^{20} A/Wm^2$, and $M_n = 1.1$. The coolant pumping power is assumed to be proportional to the total thermal power: $P_{pump} = 0.03P_{th}$. Similarly, the subsystem + control power is also assumed to be simply proportional to the total thermal power: $P_{sub} + P_{control} = 0.04P_{th}$. More accurate assessments of the power requirements for these and other auxiliary systems are needed and is a topic for future research. With the above assumptions, the plasma stability, confinement, magnet technology, tritium breeding ratio, and shielding requirements to avoid exceeding neutron damage limits together determine the device size needed to achieve $Q_{eng} \geq 1$ and support a FNST mission.

3. Assessment of Device Size

An important constraint on overall device size for the AT and CS is the maximum magnetic field strength allowed at the superconducting (SC) toroidal field (TF) coil, or equivalently the effective TF current density as determined by superconducting strand current density and the space needed for magnet cooling, quench protection, and structural support. To minimize device size, the AT and CS devices discussed here use effective TF current densities $\langle J_{TF} \rangle \approx$ 20-25 MA/m^2 and maximum fields B$_{max} \approx$ 13-14T above ITER values (10-12 MA/m^2, B$_{max} \approx$ 11-12T) and will therefore require advancements in SC TF coil technology and/or design. For the AT and CS cases, an inboard thickness of 1-1.3m was assumed for the first-wall + blanket + skeleton-ring + vacuum-vessel. This inboard shielding assumption is supported by 1D neutronics analysis described in Section 4. Figure 1 shows the aspect ratio A=4 AT pilot design space for $Q_{eng} \approx 1$, $\eta_{th} = 0.3$, thermal $\beta_N \leq 4$, B_{max} at the TF coil $\leq$ 14T, and average neutron wall loading $\langle W_n \rangle \geq$ 1MW/m^2. As shown in the figure, devices with major radius $R_0 \geq$ 3.5m at $B_T \approx$ 5-6T are possible. However, with ITER-like TF magnet parameters the pilot size increases to $R_0 \approx$ 6-7m.

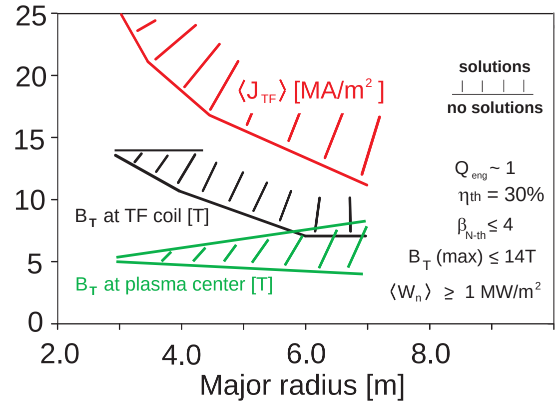

Figure 1: AT pilot $\langle J_{TF} \rangle$ and B_T at TF coil and plasma center vs. major radius.

Figure 2 shows the dependence of the required confinement multiplier H_{98} on density, device size, and blanket thermal conversion efficiency for the AT pilot. A $R_0 \leq$ 4m AT pilot requires H_{98} as low as 0.9 at Greenwald fraction $n/n_{Greenwald}$ near 1 for $\eta_{th} = 0.45$, while H_{98} increases to 1.4 at lower $n/n_{Greenwald} \approx 0.5$ for $\eta_{th} = 0.3$. The variation of H_{98} with density is a result of the density dependence of the ITER H-mode confinement scaling, the temperature dependence of the fusion cross-section, and the reduction of current drive efficiency at increased density (similar trends are observed for the ST as shown in Figure 3). Figure 2 also shows that for larger $R_0 \leq$ 7m AT pilots, sub-ITER H-mode confinement multiplier H_{98} of only 0.7-0.9 is needed for $n/n_{Greenwald} = 0.5$ to 1 for $\eta_{th} = 0.45$. There is a stronger dependence on density at lower $\eta_{th} = 0.3$, with H_{98} increasing from 0.8 to 1.3 as $n/n_{Greenwald}$ is lowered from 1 to 0.5.

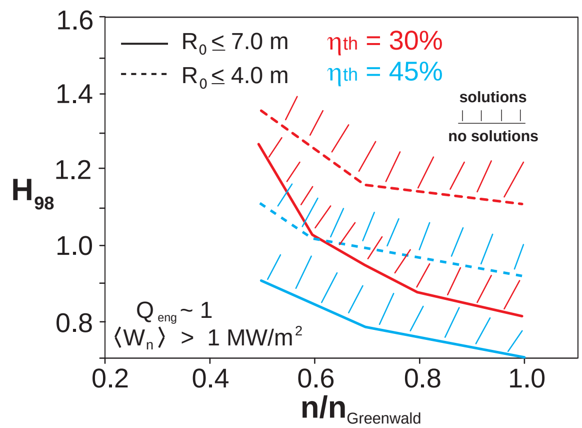

Figure 2: AT confinement multiplier H_{98} required for $Q_{eng} \approx 1$ vs. density, R_0, and η_{th}.

For an ST pilot it is important to minimize TF resistive losses to 150-200MW to enable access $Q_{eng} \approx 1$. To achieve this, the vacuum toroidal field at the plasma geometric center is limited to $B_T \leq 2.4$ Tesla, a flared TF copper central rod and large cross-section copper TF return legs are utilized, and SC PF coils (like ARIES-ST) are used for coils not attached to the central rod. The ST utilizes a 0.10m thick shield to reduce radiation damage and nuclear heating of the central Cu TF magnet and has a plasma aspect ratio A = 1.7.

The ST pilot operating points with $Q_{eng} = 1$ and $\eta_{th} = 0.45$ depend strongly on the normalized density $n/n_{Greenwald}$ as shown in Figure 3 for a range of major radii: $R_0 = 1.6$, 1.9. and 2.2m. Negative Neutral Beam Injection (NNBI) heating and current drive (CD) with injection energy = 0.5MeV is assumed for the ST pilot. Figure 3a shows that at fixed P_{NBI} = 30MW the β_N decreases by ≈ 0.5 for each 0.3m increase in major radius. These β_N values are above the no-wall limit, and resistive wall mode stabilization would be utilized [11]. Figure 3a also shows that the total β_N increases substantially at low density, whereas β_N varies only weakly with density for $n/n_{Greenwald} > 0.6$. Much of the increase in β_N at low density is due to increased fast-ion (NBI + alpha) stored energy fraction (due to increased slowing down time) as shown in Figure 3c. Figure 3b shows that the required H_{98} decreases rapidly with increased density as device size is increased and is almost independent of device size at fixed P_{NBI}. Figure 3d shows the strong increase in bootstrap fraction with increased density as the NBI-CD decreases at high density. Doubling the heating power at fixed R_0 = 2.2m from P_{NBI} = 30MW (orange) to P_{NBI} = 60MW (green) reduces the required H_{98} by a factor of 1.1-1.25 and increases the externally controlled NBI-CD fraction by a factor of 1.6-2.

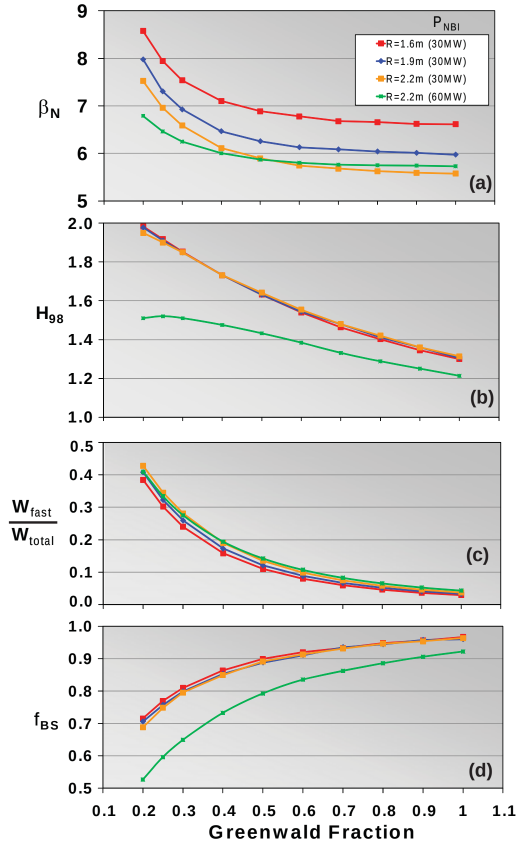

Figure 3: ST pilot $\beta_N, H_{98}, W_{fast}/W_{tot}$, and f_{BS} vs. density at Q_{eng} = 1.

Unlike the AT or ST, the CS pilot does not require auxiliary current drive and also has a much wider operating space with respect to plasma density. It is also possible to operate with comparatively low auxiliary power and higher fusion gain Q. The CS pilot device size is therefore determined largely by achievable magnetic field, confinement, and stability. As shown in Figure 4a, CS pilots with $B_T \geq 5$T and average major radius $\geq$ 4m can produce $Q_{eng} > 1$ provided the confinement is near H-mode levels (assumed to be $\leq 2\times$ the 2004 international stellarator scaling for L-mode = ISS04) and the total β is near $\leq 6\%$ which is assumed to be no-wall stable based on stability analysis for ARIES-CS [6]. Figure 4b shows that for reduced $H_{ISS04} \approx 1.25 - 1.5$, increased $B_T \geq 6$T and average major radius $\geq$ 5m are required to produce $Q_{eng} > 1$.

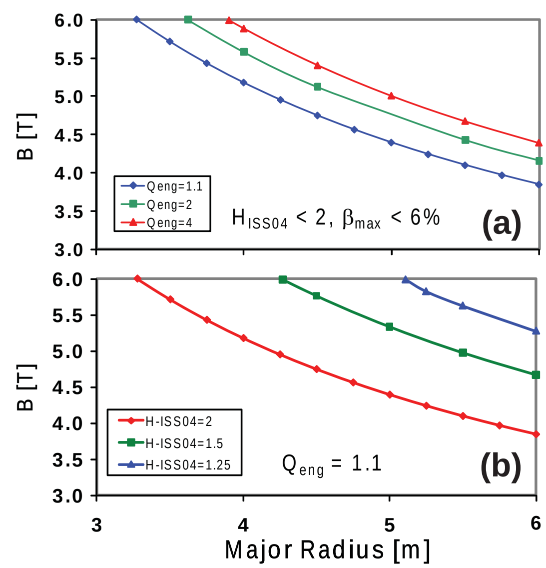

Figure 4: Toroidal field required in the CS pilot vs. average major radius for a range of (a) Q_{eng} and (b) H_{ISS04} values.

	η_{th}	A = R_0 / a	R_0 [m]	κ	B_T [T]	I_P [MA]	q_{95}	q_{cyl}	f_{BS} or iota from BS	n_e/n_G	H_{98} or H_{ISS04}	β_T [%]	β_N	P_{fus} [MW]	P_{aux} [MW]	Q_{DT}	Q_{eng}	<W_n> [MW/m^2]	Peak W_n [MW/m^2]
AT	0.30	4	4	2	6	7.7	3.8	2.4	0.59	0.9	1.2	4.6	3.6	553	79	7.0	1	1.8	2.9
AT	0.45	4	4	2	6	7.7	3.8	2.4	0.5	0.8	1.1	3.9	3	408	100	4.1	1	1.3	2.1
ST	0.30	1.7	2.2	3.3	2.4	20	7.3	2.8	0.89	0.7	1.35	39	6	990	50	19	1	2.9	5.2
ST	0.45	1.7	2.2	3.3	2.4	18	7.8	3.0	0.85	0.7	1.3	30	5.2	630	60	10.5	1	1.9	3.4
CS	0.30	4.5	4.75	1.8	5.6	1.7	1.5	-	0.2	-	2	6	-	529	12	42	2.7	2	4.0
CS	0.45	4.5	4.75	1.8	5.6	1.7	1.5	-	0.2	-	1.6	6	-	313	18	17	2.7	1.2	2.4

Table 1: Parameters for AT, ST, and CS pilot plants for thermal efficiency η_{th} *= 0.3 and 0.45.*

Based on the analysis above, Table 1 summarizes the parameters of pilot devices based on the AT, ST, and CS for two values of thermal efficiency $\eta_{th} = 0.3$ and 0.45. This range of thermal efficiencies is chosen to approximately span the range expected for candidate pilot plant blankets including He-cooled pebble-bed (HCPB) ceramic blankets and dual-coolant breeding blankets with flowing Pb-Li (DCLL) [2, 6, 12, 13].

Several noteworthy trends are evident from Table 1. First, the AT pilot plant with $\eta_{th} = 0.45$, $\langle J_{TF} \rangle \approx 21$ MA/m^2 (B$_{max}$ = 13T), and $Q_{eng} = 1$ has toroidal field, plasma current, β_N, and fusion power similar to proposed ITER fully non-inductive scenarios but with reduced $H_{98} = 1.2$ (versus 1.5-2 for ITER) and in a device 30% smaller in major radius. The ability to achieve similar fusion performance in a smaller device results from assumed improvements in TF magnet technology and from sizing the central solenoid to provide only enough flux-swing for plasma current ramp-up. Operation at lower $\eta_{th} = 0.3$ requires higher values of P_{fus}, Q_{DT}, H_{98}, and β_N at or above the no-wall stability limit. For comparison, due to inboard space constraints, the ST pilot (like other A $<$ 2 designs) has no inboard blanket or solenoid and utilizes single-turn normally conducting (Cu) TF coils. The use of Cu TF coils increases the recirculating power and thus the fusion power required to achieve $Q_{eng} = 1$. As shown in Table 1, the ST fusion powers are 1.5-1.8$\times$ the AT values with similar dependence on η_{th}. The ST pilot plant plasma current is 2.5$\times$ higher than in the AT, and the bootstrap fractions are also higher, although comparable bootstrap current fraction $f_{BS} \approx 0.6$ is achievable in the ST at lower $n_e/n_{Greenwald} = 0.3$ and higher $H_{98} = 1.5$. The ST pilot plasma has the highest average neutron wall loading (due to higher required fusion power) of all configurations assessed with average neutron wall loadings $\langle W_n \rangle$ comparable to those previously proposed for nuclear component testing [8]. Finally, for the stellarator pilot plant, to minimize device size, increase neutron wall loading for FNST, and utilize physics assumptions closest to the AT, a quasi-axisymmetric (QAS) CS design with low average aspect ratio $\langle R_0/a \rangle = 4.5$ is chosen. Favorably, for $\eta_{th} = 0.3$, the CS has fusion power 500MW similar to the AT but with much lower P_{aux}, 4-6$\times$ higher Q_{DT}, and $Q_{eng} = 2.7$ due to elimination of external current-drive power and the usage of H-mode like confinement ($H_{ISS04} = 2$). Higher η_{th} enables solutions with similar Q_{eng} at lower $P_{fus} \approx 300$MW.

4. Radial Build, Device Layout, and Maintenance

A critical aspect of pilot design is provision for adequate space for internal components including breeding blankets, neutron shielding, structural supports, and manifolds. The parameters summarized in Table 1 combined with lifetime assumptions and maintenance requirements enable estimation of the radial build requirements for each pilot.

The pilot plant lifetime is assumed to be 20 years with availability 10-50% (30% average) = 6 full power years (FPY). The vacuum vessel, manifolds, skeleton rings, and SC TF coils are assumed to be lifetime components as are the normally conducting TF coils for the ST (with the exception of the center stack). Damage to ferritic steel (FS) structure is limited to 80 dpa, and He production is limited to 1 He appm where reweldability is required. For the SC magnets (operating at 4K), the peak fast neutron fluence to Nb_3Sn ($E_n > 0.1$ MeV) is limited to 10^{19} n/cm^2, peak nuclear heating ≤ 2 mW/cm^3, peak dpa to Cu stabilizer $\leq 6 \times 10^{-3}$ dpa, and peak dose to electric insulator $\leq 10^{10}$ rads. Finally, the overall tritium breeding ratio (TBR) for all blanket systems is required to be approximately 1.1 in order to achieve net TBR of 1.01 including TBR reductions from test modules and/or large penetrations.

Preliminary inboard and outboard radial builds for DCLL blankets for the AT, ST, and CS pilots are shown in Figure 5 based on these specifications. For the AT, an inboard (IB) blanket thickness of 40cm is used and would be replaced every 2.5 FPY. The OB blanket would be 76cm thick, and would also be replaced every 2.5 FPY. For the ST, on the inboard side, there is a He-cooled FS shield+VV to reduce nuclear heating and radiation damage of the Cu magnet. Because of the higher neutron wall loading in the ST, the inboard midplane shielding structure would require replacement every 1.8 FPY, and the central Cu TF magnet would be replaced concurrently. The Cu of centerstack TF becomes embrittled at 0.1 dpa (2-3 days of full power operation), but central TF conductor designs are possible that keep stresses below allowable limits [14]. On the outboard side, a 1m thick blanket is used and would require replacement every 1.4 FPY. Finally, for the CS, a uniform 53cm thick blanket is used everywhere (except behind the divertor) and would require replacement every 1.7 FPY.

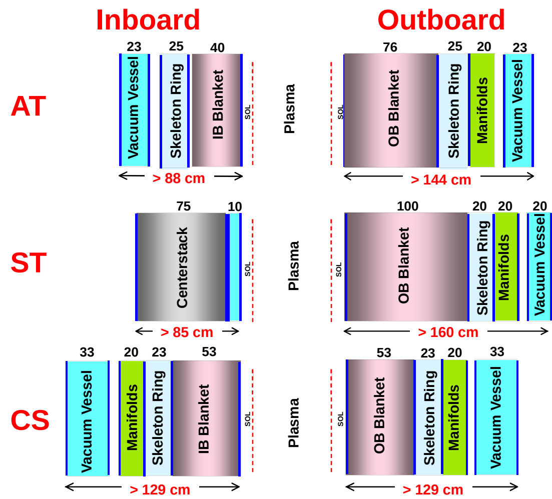

Figure 5: Radial builds of pilot blankets, skeleton rings, manifolds, vacuum vessels.

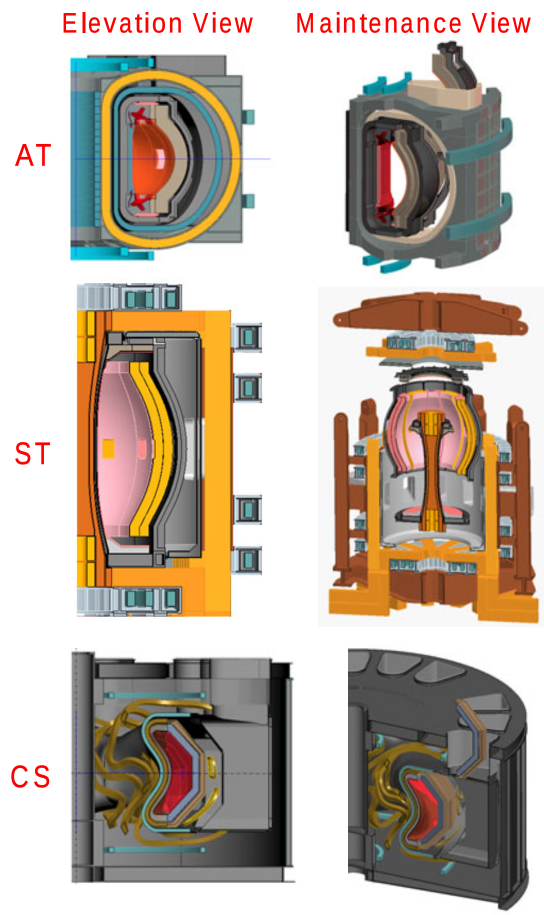

Figure 6: Pilot plant elevation views and 3D views of vertical maintenance schemes.

Based on the pilot parameters in Table 1 and radial build information in Figure 5, 3D conceptual designs have been developed for each configuration incorporating free-boundary equilibrium calculations and conventional divertor designs for the AT and ST. The AT design incorporates a central solenoid sufficient for plasma current ramp-up, whereas the ST requires solenoid-free ramp-up using NBI and bootstrap current-drive. Special attention has been paid to maintenance, and a vertical maintenance scheme has initially been adopted for each of the pilot configurations. As shown in Figure 6, for the AT, individual toroidal segments are translated radially, then lifted vertically through ports above the pilot core. For the ST, the upper support structure, PF coils, and TF horizontal legs are removed as a unit, and then the core (centerstack + blankets + skeleton) is removed vertically either as a unit or in a small number of components. For the CS, the modular coils have been straightened in the outboard sections (not yet re-optimized to satisfy plasma requirements) enabling the incorporation of vertical ports similar in design to the AT. Design strategies to simultaneously satisfy physics requirements and criteria for favorable maintenance are being investigated [15]. As shown in Figure 6, CS individual toroidal segments are also translated radially, then lifted vertically through the ports.

5. Tritium Consumption

The pilot plants described here are projected to produce 0.3-1 GWth fusion power to achieve $Q_{eng} \geq 1$. Since the DT fusion T burn-up rate is 56 kg / GWthy, full-power pilots would consume 17-56 kg of T per FPY. The world maximum T supply (from CANDU) over the next 30-40 years is 25-30 kg, and ITER is projected to consume roughly one half of this amount assuming 0.5 GWth operation at 2% duty factor for 10 years and including T decay at 5.5% per year. Thus, blanket technology development programs aiming to achieve T self-sufficiency would have available approximately 5-15 kg of T, or would have to purchase T from external sources at an estimated cost of 30-100M USD/kg.

The requirements and strategy for establishing the feasibility, operability, and reliability of blanket and PFC systems has been established over several decades [7]. The fusion testing requirements for blanket development include: local neutron wall loading $\geq$ 1 MW/m^2, steady state operation, test area $\geq$ 10 m^2, and testing volume $\geq$ 5 m^3. The development program is envisioned to have three phases: (I) fusion break-in for initial exploration of performance in a fusion environment with neutron fluence of 0.3 MWy/m^2, (II) engineering feasibility phase for concept performance verification and selection with $1-3$ MWy/m^2, and (III) engineering development and reliability growth with $\geq$ 4-6 MWy/m^2 accumulated test-time utilizing multiple improved blanket versions.

All three pilots have sufficient outboard testing surface area and volume to incorporate test blanket modules. The AT and ST pilots require a fusion power of roughly 200 MW to produce a peak outboard neutron wall loading of 1 MW/m^2, while a CS (due to the 3D dependence of the neutron production rate) would require $\approx$ 130 MW of fusion power. In the conservative limit of minimal T breeding, achieving 1 MWy/m^2 peak neutron fluence would require 8-12 kg of T - consistent with the expected availability of T following ITER DT operation. Thus, it should be possible to establish the engineering feasibility of blanket systems during initial pilot plant operation. Assuming TBR near 1 is established during phase I of operation, it should be possible to extend operation to higher fluence in Phases II and III of development without requiring additional T from external sources.

Smaller FNST devices [8, 9, 10] designed specifically to maximize neutron wall loading per unit fusion power will consume less T per MWy/m^2 than a pilot, but will nevertheless need to breed or purchase T during Phase II of the blanket development program. A potential advantage of the pilot is that if the blanket development program is successful, the pilot is (by design) more able to achieve conditions for net electricity production in a configuration and with a maintenance scheme representative of a full-scale power plant.

6. Research Needs and Future Work

Many research needs remain for the pilot plant concept. Improved magnet technology is needed for all configurations - in particular higher effective current density SC magnets for the AT and CS and the development and fabrication of large single-turn radiation-tolerant Cu TF magnets for the ST. For the CS, additional engineering and physics analysis is needed to assess passive shaping by superconducting tiles proposed to simplify CS coils and improve maintainability and availability. For physics needs, the AT and ST require fully non-inductive operation at high elongation operating near or above the n=1 no-wall limit with very low disruptivity. The ST additionally requires non-inductive plasma current ramp-up. While both the ST and QAS CS share a substantial physics basis with the conventional aspect ratio tokamak, additional development of the ST and CS physics bases is needed to reduce risks in extrapolation to larger device size. The AT and ST clearly challenge the steady-state and transient power handling capabilities of existing technologies with $P_{heat}/S \approx 1\text{MW/m}^2$, $P_{heat}/R \approx 30 - 60\text{MW/m}$, and $W/S \approx 0.5 - 1\text{MJ/m}^2$ which are at or above ITER values and comparable to values projected for FNST devices [8, 9, 10]. More generally, plasma-material interface capabilities supporting long-pulse, high duty-factor (10-50% availability), high power-loading, and high-temperature first-wall and divertor operation remain to be developed. Finally, if a pilot plant is to have an FNST mission and is to extrapolate to a commercial power plant, device maintainability will be paramount. A vertical maintenance scheme is proposed for all three configurations enabling segment removal and replacement of major internal components (possibly the entire core for the ST). Additional research is needed to assess the advantages and disadvantages of this approach, and to determine approaches for maintaining smaller internal components. These issues will be addressed in future work.

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