Contribution ID: **856** Type: **Overview Poster**

Overview of the SPARC physics basis towards the exploration of burning-plasma regimes in high-field, compact tokamaks

Monday, 10 May 2021 16:00 (20 minutes)

Achieving net energy production in magnetic confinement fusion devices is a key milestone in the quest for fusion energy. With the mission of demonstrating net fusion energy, the SPARC tokamak is being designed jointly by the MIT Plasma Science and Fusion Center and Commonwealth Fusion Systems. Its study of reactor-relevant, alpha-heating-dominated scenarios and high power density regimes will help retire risk for ITER operations and for fusion power plants. A team of over 100 engineers and scientists is on track to deliver a toroidal field model coil using high-temperature superconductor (HTS) technology by 2021, with the engineering design of the tokamak progressing in parallel. Negotiations with potential host sites in the Northeast US are underway, with start of construction planned in 2021 and operation expected in 2025.
SPARC will be a pulsed machine operating with Deuterium-Tritium (DT) fuel and with ICRF auxiliary heating. The high strength of the magnetic field ($B_T > 12.0T$ on axis), will allow operation at high plasma current and high absolute density, leading to net fusion output in a device with a size comparable to current tokamaks ($R_0 < 2.0m$). In particular, the SPARC mission objective has been established as demonstration and study of Q>2 plasma conditions, where Q is the ratio between the total fusion power and the external power absorbed in the plasma. Figure 1 depicts the poloidal cross-section of the Version 1C (SPARC V1C) design iteration, and Figure 2 indicates main plasma parameters for the baseline DT H-mode plasma discharge.

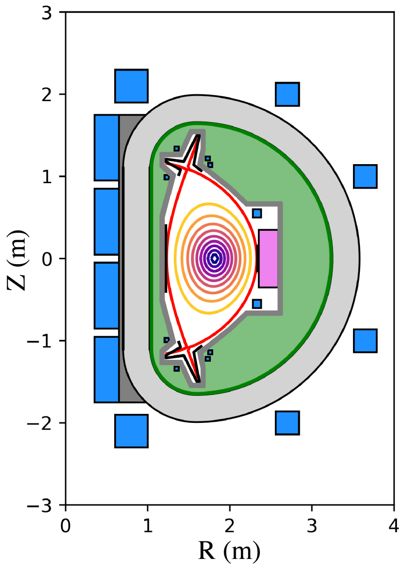
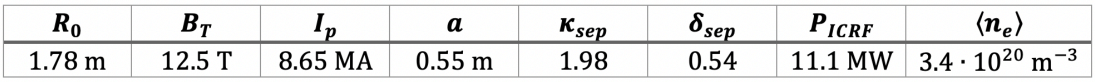


Figure 1: SPARC V1C poloidal cross-section.



R_0	B_T	I_p	a	κ_{sep}	δ_{sep}	P_{ICRF}	$\langle n_e \rangle$
1.78 m	12.5 T	8.65 MA	0.55 m	1.98	0.54	11.1 MW	$3.4 \cdot 10^{20}$ m^{-3}

Figure 2: SPARC V1C main plasma parameters for nominal DT H-mode operation.

Following a traditional design workflow (1), SPARC parameters are first selected using empirical scaling laws and plasma operation contour (POPCON) analysis. Figure 3 represents the operational space for SPARC V1C for its baseline scenario, demonstrating that $Q \approx 11$ can be reached with conservative assumptions ($H_{98} = 1.0$ confinement, and $\nu_{Ti} = 2.5$, $\nu_{ne} = 1.3$ profile peaking factors, consistent with empirical predictions).

Total fusion power remains below administrative limits for the machine ($P_{fus} < 140MW$), and safety factor ($q^* = 3.05$), normalized density ($f_G = 0.37$) and normalized pressure ($\beta_N = 1.05$) are at reasonably safe levels of operation.

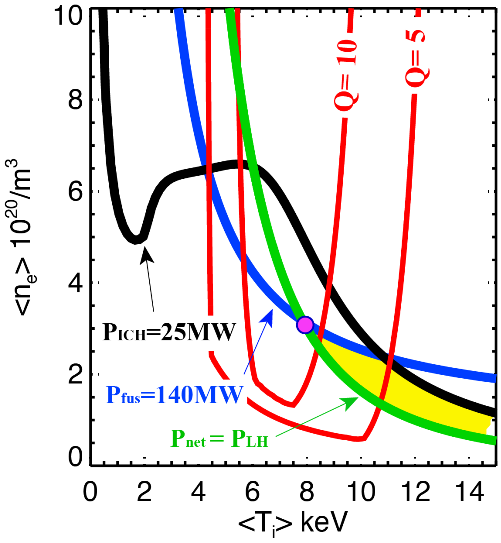


Figure 3: SPARC V1C operational space, bounded by LH threshold (green), maximum fusion power (blue) and available ICRF power (black). $Q_{max} = 11.4$ (circle).

The development and validation of theory-based reduced models allow integrated simulations to also inform the design of SPARC. To this end, simulations with the TRANSP code (2) coupled with the TGLF model (3) for turbulence and EPED (4) for pedestal stability are performed. Figure 4 depicts simulated temperature and density profiles. $H_{98} \approx 1.0$ is predicted, and fusion gain results in $Q \approx 8.2$. The good agreement between the two independent workflows (empirical and theory-based simulations) provides high confidence that SPARC will accomplish its $Q > 2$ mission. During nominal operation, D-T(3He) ICRF minority heating at 120 MHz will be utilized for on axis heating of both 3He and T. AORSA and CQL3D (5) simulations are in good agreement with TRANSP, which uses TORIC (6) to model ICRF. Single-pass absorption is excellent, and minimum losses of ICRF power (~1%) to alphas are predicted.

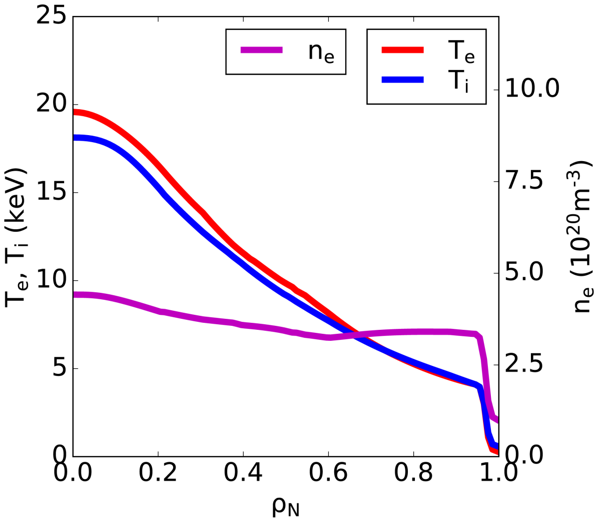


Figure 4: Predicted profiles for SPARC V1C.

Loss of fast ions (alphas and RF-tail ions) due to toroidal field (TF) ripple can be a major issue for the design of DT tokamaks, as it can lead to excessive localized wall heating. The effect of first-orbit, classical and TF ripple in SPARC has been studied with ASCOT (7). Simulations indicate that the total losses are small (<2%), due to the low edge TF ripple (0.15%) in the SPARC design. There is no concentration of losses toroidally and only modest concentration poloidally in the current TF design.

Managing divertor heat flux will be challenging in SPARC, but unmitigated levels are comparable to ITER. To ensure divertor survivability with conservative assumptions (i.e., not relying on partial or complete divertor detachment), the poloidal field coil set and central solenoid are being designed to ensure that a fast strike point sweep (~1 Hz) can be achieved. Thermal simulations of the divertor target indicate that sweeping during the flat-top is sufficient to ensure divertor survivability with only a moderate divertor radiation fraction. SPARC will be equipped with impurity gas injection to attain detached-divertor scenarios. The feasibility of an "advanced" divertor is also being assessed.

In summary, SPARC will be an important experiment to study burning plasma physics and will be a proof-of-principle for high-field, compact fusion power plants. The SPARC design is converging towards a self-consistent model of the machine with robust engineering and physics. Conservative estimates of fusion gain show significant margin for the Q>2 mission, leaving room for extensive exploration of burning plasma physics regimes.

This work was supported by Commonwealth Fusion Systems.

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Session Classification: OV-OV/P Overview Posters

Track Classification: Overview