Divertor heat flux challenge and mitigation in SPARC


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Divertor heat flux challenge and mitigation in SPARC

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The SPARC experiment, because of its high magnetic field, high power, and compact size, will operate with divertor conditions at or above those expected in reactor-class tokamaks. Power exhaust at this scale remains one of the key challenges for practical fusion energy. Based on empirical scalings, the peak unmitigated divertor parallel heat flux is projected to be greater than 10 GW/m\textsuperscript{2}. This is nearly an order of magnitude higher than has been demonstrated to date. Furthermore, the divertor parallel ELM energy fluence projections (~11-34 MJ/m\textsuperscript{2}) are comparable to those for ITER. However, the relatively short pulse length (~25 second pulse, with a ~10 second flat top) provides the opportunity to consider mitigation schemes unsuited for long pulse devices including ITER and reactors. The baseline scenario for SPARC employs a ~1 Hz strike point sweep to spread the heat flux over a large divertor target surface area to keep tile surface temperatures within tolerable levels without the use of active divertor cooling systems. In addition, SPARC operation presents a unique opportunity to study divertor heat exhaust mitigation at reactor level plasma densities and power fluxes. Not only will SPARC test the limits of current experimental scalings and serve for benchmarking theoretical models in reactor regimes, it is also being designed to enable the assessment of long-legged and X-point target advanced divertor magnetic configurations. Experimental results from SPARC will be crucial towards retiring risk for a fusion pilot plant divertor design.

1. Introduction

The interface/interaction between the high plasma power/particle fluxes and plasma facing components in the presence of reactor-level power flows has long been recognized as a critical issue for fusion energy ("Report on Science Challenges and Research Opportunities in Plasma Materials Interactions", US Department of Energy 2015). SPARC, a compact, high magnetic field tokamak, aiming to demonstrate net energy from fusion (Creely Submitted), will be prototypical of divertor power exhaust requirements. Empirical scalings (Eich 2013, Brunner

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2018, Eich 2020) for the scrape-off layer (SOL) heat flux width, indicate that unprecedented high unmitigated peak parallel heat fluxes ($q_p > 10$ GW/m$^2$) are anticipated in the SPARC divertor. Furthermore, transient events such as edge-localized modes (ELMs) and disruptions impose heat flux and structural loading conditions comparable to ITER and will pose a significant challenge for SPARC (Hughes Submitted, Sweeney Submitted). At the same time, it is important to note that these predictions for SPARC rely on extrapolations from the experimental databases. Measurements in SPARC would be valuable for benchmarking these experimental scalings as well as validating first principles computational models, for example XGC (Chang 2017), for reactor conditions for the first time, thus increasing our confidence in subsequent predictions for ITER and a future fusion pilot plant like ARC (Sorbom 2015).

In present day pulsed devices, heat exhaust management is achieved by keeping heat fluxes and pulse lengths at levels that are low relative to material limits and/or employing engineering designs with intra-shot cooling. In some high-power cases, for example in JET (Telesca 2016) and ASDEX-Upgrade (Kallenbach 2015), control of heat flux to plasma facing components (PFC) is necessary and low-Z divertor impurity seeding is needed to access dissipative divertor regimes. How these impurity seeding experiments might scale to the SPARC device and its impact on the ability to achieve the primary mission goal of $Q > 2$ is unclear because of the sizable extrapolation to SPARC divertor parameters from these devices. Alternatively, active feedback control systems have also been developed that can reduce the heat flux to the divertor target surface without negatively impacting core plasma performance (Kallenbach 2012, Brunner 2017, Guillemaut 2017, Courtois 2019). Whether any of these active-feedback systems can be implemented in the SPARC device has yet to be assessed, but it will be challenging given the available diagnostic access in the nuclear environment and the actuator response times. Since we cannot ensure, from a physics and control perspective, that SPARC will achieve a dissipative divertor in its net-energy operation, attached divertor conditions have to be assumed in the design of the SPARC divertor.

Similarly, while the SPARC magnets and PFCs are being designed to be up-down symmetric and the primary reference discharges are all double-null equilibria, there are concerns about the ability to control and maintain double-null to the precision needed for power sharing across the upper and lower sets of divertors (De Temmerman 2011, Brunner 2018). Hence, the baseline design scenario will also have to assume power sharing fractions consistent with a single null equilibrium. To avoid melting or sublimating of the inertially cooled divertor, the strike point will be swept at ~1 Hz over the course of the 10 second discharge flat top, with the full sweep extent for SPARC (presently at a “V2” design) shown in figure 1a. Strike point sweeping for heat flux mitigation was a routine part of high-power operations on Alcator C-Mod (Reinke 2019) and JET (Silburn 2017).

With those conservative assumptions as a baseline, SPARC is also being designed to maximize the possibilities of accessing high power dissipative divertor regimes. Simulations of SPARC V2 are being developed using UEDGE (Rognlien 1992) and SOLPS-ITER (Wiesen 2015, Bonnin 2016). The simulation results are being used to bound divertor performance expectations, inform design requirements for the divertor control systems (e.g. gas injection and sweeping) and finally to predict wall particle and power fluxes on main chamber and divertor surfaces. Divertor heat flux management will be a key focus as the tokamak is commissioned to full power operations. Even with the limited diagnostics expected to be available, SPARC will uniquely open up new insights into dissipative divertor access and controllability at reactor level power and particle fluxes.
SPARC will be an important testbed for studying reactor level divertor power and particle densities and the potential solutions for a future fusion pilot plant. Looking beyond the techniques used for high-powered discharges today – impurity injection and divertor neutral pressure control, SPARC will have a tightly baffled divertor for high neutral compression and inclined target plates to assist access to dissipative divertor regimes. Furthermore, “advanced divertor” magnetic geometries present an interesting opportunity for SPARC to reduce heat loads, enhance access to detachment and improve detachment control. Therefore, the SPARC magnet coil set is being designed to produce a sub-class of “advanced divertors”, including long-legged and X-point target (XPT) magnetic geometries (figure 1b). The XPT magnetic geometry was first proposed as part of the ADX divertor test tokamak (LaBombard 2015) design process and has since been incorporated into the ARC pilot plant design (Kuang 2018). Although the XPT geometry can only be achieved in SPARC at reduced core plasma current, \( I_p < 5.7 \) MA, the core scenarios still project to a fusion power of up to 37 MW and \( Q \sim 2 \) (Creely Submitted). SPARC will be a unique and useful platform for both answering key questions about the physics of such advanced divertor geometries and providing data that would be crucial for designing the divertor for ARC. The limitation on what questions can be answered and to what precision will be dictated by the divertor diagnostic set, which is currently under consideration.

This paper will focus on the ongoing characterization and assessment of the current SPARC divertor design. The paper is organized as follows: Section 2 outlines the design guidelines for the SPARC divertor based on projections for SPARC derived from current empirical and theoretical models; Section 3 presents simulations of the divertor surface material temperature response to a strike point sweep, the conservative and primary heat flux mitigation scenario for the SPARC divertor if enough divertor power dissipation is not achieved; Section 4 introduces the development of next step models being developed to assess the access to dissipative divertor regimes and assist in refining the design guidelines; Section 5 provides a preliminary outline of the divertor XPT scenario being explored for SPARC; and lastly Section 6 discusses the uncertainty in the design guidelines resulting from the significant extrapolation of present devices to SPARC and highlights the potential for SPARC to be able to provide valuable data to increase the confidence in the design of a future fusion pilot plant.
2. Divertor design guidelines for SPARC

The empirical and heuristic models that have been developed to date are used to develop the heat loading design scenarios for the SPARC V2 divertor. There is a sizeable uncertainty in trying to determine the divertor target surface heat fluxes in a new device, especially since it requires extrapolation of present-day datasets. Therefore, conservative assumptions are used in order to ensure that SPARC can survive to achieve its primary scientific missions of $Q>2$.

Full field (12.2 T), full current (8.7 MA), DT H-mode discharges in SPARC V2 (Creely Submitted) are the reference discharge for designing the divertor. Based on the multi-machine scalings by Eich (Eich 2013) for the steady state heat flux width ($\lambda_q$) under attached H-mode conditions, the parallel heat flux entering into the divertor is proportional to $P_{\text{SOL}}B_0/R_0$ (LaBombard 2015). Where, $P_{\text{SOL}}$ is defined as the conducted power crossing into the scrape-off layer (SOL), $B_0$ is the on-axis magnetic field, and $R_0$ is the major radius. $P_{\text{SOL}}B_0/R_0 \sim 30 \text{ MW} \cdot \text{T/m}$ has been achieved on Alcator C-Mod (Hughes 2018) and ASDEX-Upgrade (Kallenbach 2015). While in preparations for DTE2, JET has also started to push $\sim 31 \text{ MW} \cdot \text{T/m}$, assuming 0% core radiation (Garzotti 2019). For the SPARC V2 full field, full current, DT H-mode, the normalized power of $P_{\text{SOL}}B_0/R_0 \sim 191 \text{ MW} \cdot \text{T/m}$ (assuming 39% core radiation fraction) is thus significantly higher than in present day devices and approaches levels for proposed fusion pilot plants, like ARC – where $P_{\text{SOL}}B_0/R_0 \sim 256 \text{ MW} \cdot \text{T/m}$ (assuming the nominal 35% core radiation fraction) (Kuang 2018). The scaling developed by Eich (Eich 2013) is for the period between ELMs and attached divertor H-modes only. While the full field L-mode discharge (Creely Submitted) has a higher projected value of $P_{\text{SOL}}$ due to the lower core radiation fraction, the wider $\lambda_q$ projected for SPARC L-mode relative to H-modes (Brunner 2018) at the same engineering parameters (e.g., poloidal magnetic field and $P_{\text{SOL}}$) would result in a reduced parallel heat flux, relative to H-mode, entering into the divertor. Therefore, unless otherwise stated, for the rest of this paper the full field H-mode reference discharge parameters summarized in table 1, will be used for modelling purposes, while the magnetic equilibriums used in this paper were generated using the FreeGS Grad-Shafranov solver (Dudson 2019).

The decision for divertor target material has yet to be made - with a current focus of analysis on tungsten or carbon. Important trade-offs are still being assessed. Carbon based PFC materials are preferred from a divertor heat flux handling standpoint. Decades of experience have shown that sublimation of carbon-based, temperature limited PFC designs allow failures from unintended overheating to avoid permanent damage that impacts future operations. Conversely, a high Z PFC material like tungsten could just crack or melt. Melted tungsten in a tokamak environment has been shown to exhibit a positive feedback loop where the melt layer will pool and protrude from the PFC surface (Coenen 2015, Pitts 2017), creating large leading edges that further concentrate plasma heat flux. The core plasma is also significantly less tolerant to high Z core impurities, small quantities can easily lead to the radiative collapse of the plasma resulting in a disruption which will place significant structural and thermal loads on the SPARC device (Sweeney Submitted). However, carbon as a PFC material does not project towards a reactor due to the high erosion rates and tritium retention concerns (Andrews 1999, Tanabe 2003, Skinner 2006). Due to the relatively short integrated discharge time over the course of the device lifetime, these issues do not immediately preclude the use of carbon PFCs in SPARC but they are nonetheless still a concern, e.g. the impact of dust production on high power ICRF antenna operations, and is being carefully evaluated by the team. In this paper, the assumed PFC material for each analysis is specified.
Table 1. Full field H-Mode projections for SPARC V2. These are based on 1D calculations by Creely (Submitted).

2.1 Divertor power exhaust

$P_{SOL}$ is the amount of conducted power crossing the last closed flux surface (LCFS) and entering the SOL. It is calculated as the sum of power from auxiliary (11.1 MW), ohmic (1.7 MW), and alpha heating (~20-28 MW) less the core radiated power (~10.4-13.2 MW). Estimates of core plasma impurity content assumed in 1D calculations by Creely (Submitted), predict 29 MW of conducted power through the LCFS. While 1.5D time-dependent core modelling simulations by Rodriguez-Fernandez (Submitted), using different physics models and slightly different assumptions about impurities, yield $P_{SOL} = 19-21$ MW, due to reduced fusion power predictions and increased core radiation. Although the core impurity assumptions used are based on the existing tokamak database, they have wide uncertainties. Therefore, for the design of the divertor targets, $P_{SOL} = 29$ MW is assumed as the upper limit. While some of the power flowing into the SOL will flow across SOL flux surfaces to the main chamber limiter surfaces, or radiated in the main chamber SOL, we ignore these loss channels and make the conservative assumption that all power goes to the divertors. Since maintaining perfect double null during a discharge will likely be challenging with the narrow SOL in SPARC, each set of divertors (upper and lower) will be designed to handle the full $P_{SOL}$ and no benefits from double-null power sharing is assumed. In single null scenarios ELM-suppressed EDA H-modes, the power sharing between the inner and outer divertor has been measured in Alcator C-Mod to be approximately 40:60 (Brunner 2018). In comparison, measurements of MAST for single null H-mode discharges with off-axis neutral beam heating, recorded the inter-ELM integrated energy to the outer divertor to be 6.5 times higher than at the inner divertor. The ELM integrated energy deposition was found to be approximately 3 times higher on the outer divertor (De Temmerman 2011). This suggest that the inter-ELM divertor IN:OUT sharing is approximately 15:85, and for the ELMs, 25:75. Due to the large uncertainty in these measurements, 40% of $P_{SOL}$ is assumed to go to the inner divertor and 70% to the outer divertor, an additional 10% power has been included to provide some margin. Thus, the inner divertors on SPARC are being designed to exhaust 11.6 MW and the outer divertor 20.3 MW for a 10 second pulse.
2.2 SOL heat flux width

<table>
<thead>
<tr>
<th>Scaling</th>
<th>OMP Heat Flux Width [mm]</th>
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</thead>
<tbody>
<tr>
<td>Eich (2013) Regression 14</td>
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<tr>
<td>Eich (2013) Regression 15</td>
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<tr>
<td>Eich (2020)</td>
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<tr>
<td>Goldston (2012)</td>
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<tr>
<td>Brunner (2018)</td>
<td>0.35</td>
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TABLE 2. Projections for the heat flux width mapped to the outer mid-plane in SPARC V2. Note that for Eich (2020), the dimensionally correct regression is used.

To convert the power to the divertors into a surface heat flux, a key unknown quantity is the level of cross-field energy transport in the SOL. Without a validated model to make predictions, empirical and heuristic scalings for the heat flux e-folding width ($\lambda_q$) are used for the design of SPARC. Table 2 summarizes the projections for the heat flux width mapped to the outer mid-plane (OMP) based on a set of recent scalings. The calculations are based on the parameters shown in table 1. However, some additional assumptions are required: the poloidal magnetic field at the outer midplane derived from the magnetic equilibrium of the primary reference discharge is 2.83 T; the core plasma volume average ion temperature is assumed equal to $\langle T_e \rangle$; the edge $Z_{eff}$ is assumed to be the same as the core; the separatrix density ($n_{e,sep}$) for a H-mode is assumed $\langle n_e \rangle/3$ (Kukushkin 2001); and the separatrix temperature ($T_{e,sep}$) is calculated iteratively assuming classical Spitzer-Harm conduction (Stangeby 2000). Despite the significant uncertainty associated with the scalings and the assumptions needed, it is reassuring that they all predict the heat flux width in SPARC to within a factor of 2.5 of each other, which increases our confidence in the extrapolation. However, to be conservative, the narrowest heat flux width of 0.18 mm is assumed for the SPARC design. The 2D modelling studies discussed in section 4 adjust cross-field transport coefficients so as to match this target value of $\lambda_q$.

2.3 Divertor radiation fraction

As mentioned in the introduction, access to dissipative divertor regimes in SPARC is difficult to predict. One method is to make use of the simple 1D models developed by Reinke (2017) and Goldston (2017) to examine the scaling for the impurity fraction needed to achieve detached divertor conditions with tokamak operating parameters (e.g. magnetic field, size). Recent experiment results from JET and ASDEX Upgrade have found that the nitrogen fractions needed to maintain detached divertor conditions in a power scan roughly agree with these scaling law predictions (Henderson 2019), although the absolute magnitudes are still uncertain. Reinke (2017) estimates that to reach divertor target ion temperatures of 5 eV, a proxy for divertor detachment, a nitrogen impurity fraction of 11.4% is needed in a 1.65 m major radius, 12.0 T device that is running at a Greenwald fraction of 0.8. Scaling to SPARC V2 with $B_0$, $R_0$, and $f_{GW}$ takes this up to a 33.5% nitrogen impurity fraction and 11.5% for neon. Such high impurity fractions in the divertor will likely result in significant core contamination and impact fusion performance. As the design is currently focused on the primary reference discharge and it is unclear if such a high impurity fraction will be tolerable for the core plasma, rather than aim for ~100% radiation fraction leading to detached or near detached conditions, a moderate 50% radiation fraction is assumed for baseline design scenario of SPARC V2. As 2D SOL simulations become more mature, the specification for the radiation fraction will be revisited.
It is important to note that a 50% divertor radiation fraction is not necessarily conservative, however, there are some mitigating factors that can be used during SPARC operations to ensure divertor target survivability while still meeting the primary mission of $Q>2$. The high impurity fraction calculated for SPARC V2 based on the scaling by Reinke (2017) is largely due to the low target $f_{GW}$, and the assumed ratio of $n_{e,sep}$ to the pedestal density ($n_{e,ped}$). The former is a result of an administrative fusion power limit while the latter is driven by a desire to maximize pedestal performance. The current primary operating point is chosen to maximize fusion gain while limiting the total fusion power (Creely Submitted) but there exists a relatively large operational window that can meet the primary mission of SPARC (Q>2) at higher $f_{GW}$. Note, however, that the plasmas at higher $f_{GW}$ would have to be run ‘lean’ with a higher deuterium to tritium ratio (or larger impurity dilution) in order to keep within the fusion power limits. Similarly, the assumed ratio of $n_{e,sep}$ to $n_{e,ped}$ can be increased, but at fixed $f_{GW}$, would imply reducing the pedestal pressure, likely reducing Q from the primary reference discharge. Since the primary reference discharge projects to a Q of 11, some reduction in core plasma performance is tolerable while still allowing SPARC to achieve its primary mission target of $Q>2$.

2.4 Divertor target geometry

The divertor target geometry must be designed to fulfill a number of roles: the target plates have to be shaped to maximize the surface area swept out by the strike point sweep (figure 1a) while at the same time ensuring that sufficient divertor volume is allocated for the XPT configuration (figure 1b). The divertor outline as shown in figure 1 is an approximate outline designed to meet these requirements. The features shown in the outer divertor are a result of combining the swept heat flux mission and the XPT divertor mission. Only during XPT operation is the outer strike point intended to intersect with the vertical and horizontal faces at larger major radius. The sweeping, outside of XPT operation, will be controlled to ensure that the strike point is kept on the angled, high heat flux surfaces highlighted in pink. The development of this divertor outline is in progress as considerations have yet to be made for disruption forces, installation, assembly, and maintenance. Furthermore, it has yet to be assessed how much space is required for the secondary X-point of the XPT divertor, which will be determined from an assessment of the diagnostic needs as well as simulations of the XPT divertor. Therefore, the divertor geometry will continue to evolve as the design and analysis of the SPARC divertor progresses.

The current divertor geometry shown in figure 1, allows for the inner divertor strike point to be swept over a target surface poloidal arc length of 0.3 m, corresponding to an area of $\sim$2.6 m$^2$. During the same sweep process the outer divertor strike point sweeps out a poloidal arc length of 0.4 m, corresponding to an area of $\sim$4.5 m$^2$. In addition, the magnetic field line surface incident angle varies across the high heat flux surfaces as the strike point is swept but stays below 1°, assuming a perfectly axisymmetric target. However, machining and installation tolerances along with the needed chamfering of tile edges will increase the incident field line angle. Because of this, ‘fish-scaling’ of the divertor tiles may be necessary to protect adjacent tiles, as was done for ITER (Pitts 2017), where each tile surface is tilted by 1°. Therefore, to be conservative, an incident field line angle of 2° is assumed for the SPARC V2 divertor thermal analysis; this will be reassessed as a more detailed design is produced.
2.5 Divertor target heat flux profile

To estimate the divertor target heat flux profile, the SPARC V2 primary reference discharge magnetic equilibrium is used to calculate the flux expansion to the divertor target. The divertor target surface plasma heat flux profile is approximated using the Eich (2011) profile of an exponential function convoluted with a Gaussian function of width $S$, where the multi-machine H-mode database suggest that $S \sim \lambda_q/2$ (Eich 2013). Due to divertor and SOL radiation, the total integrated power due to incident plasma heat flux profiles is less than $P_{\text{SOL}}$, reducing the peak heat fluxes. Experimentally, as the radiation fraction increases and divertor plasma temperatures decrease, a broadening of the plasma heat flux profile is also observed (Kallenbach 2015) which should further reduce the peak; however, as the exact scaling is unclear, these additional effects are not included.

Due to the enclosed divertor geometry, the majority of the divertor radiation will be deposited on the divertor target surfaces. The radiation heat flux profile across the divertor target is dependent on the source locations and is difficult to predict for SPARC at this time. Instead, for simplicity, the divertor radiated power is simply added to the plasma heat flux profile as a constant uniform background heat flux term, similar to that used in the multi-machine H-mode database study (Eich 2013), to get the total surface heat flux profile. While this uniform background is much smaller, it is being applied continuously compared to the transient sweeping of the high heat flux feat and still plays a role in assessing the divertor survivability. Thus, total integrated power from the plasma heat flux profiles and the uniform radiation term across the entire divertor target surface is taken to equal $P_{\text{SOL}}$. Assuming a 50% radiation fraction, this gives the heat flux profiles shown in figure 2 with peak heat fluxes at the inner and outer divertor of 257 and 357 MW/m$^2$ respectively.

![FIGURE 2. SPARC V2 divertor surface heat fluxes shown as a function of distance along the divertor target surface. The profiles assume a 50% divertor radiation fraction that is ‘returned’ to the profile as a uniform background heat flux term across the entire target surface resulting in the small non-zero values of $q_{\text{Surf}}$ at the ends of the profiles shown. The rest of the profile follows an Eich (2011) profile with $\lambda_q=0.18$ mm and $S = \lambda_q/2$. Note that the total integrated power over both divertors is 31.9 MW and includes an additional 10% power due to the conservative power sharing assumptions used.](image)
2.6 Divertor ELM loading

Through a study of the SPARC pedestal included in a companion paper (Hughes Submitted), it is anticipated that Type 1 ELMs size ($\Delta W_{\text{ped}}$) in SPARC V2 will be greater than 1 MJ per ELM (14-22% of the pedestal stored energy). To translate that ELM energy into divertor power loading we utilize a multi-machine database by Eich (2017) which provides an approximate model for the peak parallel ELM energy fluence ($\varepsilon_{\|,\text{peak}}$) on the divertor target that has a linear dependence on the pedestal pressure. In that study the surface heat flux to the divertor is integrated in time across an ELM pulse and projected parallel to the magnetic field using the steady state magnetic field line incident angle. The $2^\circ$ incident field line angle assumed in section 2.5 will be used here to project $\varepsilon_{\|,\text{peak}}$ into a target surface thermal load. It has been experimentally observed that the ELM power deposition at the divertor target comes in two phases (Loarte 2007, Eich 2011): an initial fast deposition phase where a significant fraction of the ELM energy is deposited on a short timescale ($\tau_{\text{ELM}}$) leading to a rise in the divertor target surface temperatures; and a second slower phase where power continues to be deposited on the divertor targets at a lower level and the surface temperature begins to decrease. The initial fast phase is the most damaging on material surfaces and $\tau_{\text{ELM}}$ has been experimentally observed to correspond to twice the pedestal transit time for ions in the SOL (Loarte 2007). For SPARC V2, assuming a pedestal pressure of 325 kPa and pedestal density of $n_{\text{p,ped}} = 2.8 \times 10^{20} \text{m}^{-3}$, we find that $\tau_{\text{ELM}} \sim 0.120 \text{ ms}$ and $\varepsilon_{\|,\text{peak}} \approx 10.7 - 32 \text{ MJ/m}^2$. Note the factor of three range in $\varepsilon_{\|,\text{peak}}$, which roughly corresponds to the range in ELM loading observed in experiments between small to large ELMs. This is remarkably close to the energy fluence scaling projection for ITER at $\varepsilon_{\|,\text{peak}} \approx 10 - 30 \text{ MJ/m}^2$ (Eich 2017).

A common figure of merit for transient thermal loading is the surface heat flux factor (Pintsuk 2007). For a given material, the heat flux factor is a proxy for the resulting surface temperature rise based on an infinite semi-plane approximation. Approximately 15-40% of the ELM energy fluence is deposited during the first fast phase of an ELM (Loarte 2007). Accounting for this and assuming an incident field line angle of $2^\circ$, $\varepsilon_{\|,\text{peak}}$ and $\tau_{\text{ELM}}$ can be combined to calculate the surface heat flux factor for SPARC V2: $HHF_{\text{ELM}} \approx 5.1 - 41 \text{ MJ/m}^2\text{s}^{1/2}$. There is significant uncertainty in this prediction captured by the wide range in the expected transient thermal loading of the divertor targets, the upper end of which is close to the cited surface melt limit of tungsten $\sim 50 \text{ MJ/m}^2\text{s}^{1/2}$ (Pintsuk 2007) and where cracks have been generated $\sim 30 \text{ MJ/m}^2\text{s}^{1/2}$ (Hirai 2009). These ELM loads on a carbon divertor would lead to localized ‘blooms’ (Ulrickson 1990). Current efforts within the team are focused on reducing the uncertainty to more tightly bound predictions.

ELM mitigation techniques are being assessed for SPARC, as is the potential to access intrinsically ELM suppressed regimes. Any ELM mitigation or suppression technique is likely to result in reduced operating pedestal. Nonetheless, core transport modelling indicates that even if the predicted pedestal pressure drops by half due to ELM suppression, core plasma performance will still meet the primary mission of SPARC with $Q>2$ (Rodriguez-Fernandez submitted, Hughes submitted). Figure 3, estimates the ELM size and $q_{\text{ELM,peak}}$ on the divertor as a function of pedestal pressure. A scheme for ELM mitigation in SPARC has yet to be developed but this presents a potential pathway by which divertor survivability can be ensured while still meeting the primary mission of SPARC.
FIGURE 3. ELM size and the divertor peak surface heat flux predictions for SPARC V2 as a function of pedestal pressure. The two lines in each subplot bracket the range in the anticipated values. The critical surface melt limit for tungsten is shown given an expected $\tau_{\text{ELM}} \sim 0.12$ ms.

2.7 Divertor disruption loading

Thermal and structural loading during disruptions is a significant concern for SPARC and is the subject of a companion article (Sweeney Submitted) where the timescales and energy release are predicted. An unmitigated disruption can deposit up to 90% (Riccardo 2005) of the plasma stored energy (24.3 MJ) onto the divertor over the thermal quench timescale (0.1 ms). There is significant uncertainty as to the heat flux footprint during the thermal quench. Experimental measurements suggest that the footprint might grow by a factor of 3-10 (Riccardo 2005), or even up to a factor of 20 (Hender 2007). Assuming, that the steady state power sharing between the inner and outer divertor as well as the divertor target surface incident field line angle, an unmitigated disruption results in a transient thermal loading to the divertor surface with a heat flux factor of approximately $5.3 - 36$ GJ/m$^2$s$^{1/2}$. Note that this approximation only considers the stored thermal energy of the plasma and not the poloidal magnetic energy which could also contribute to surface heating of the PFCs (Lipschultz 2011), but on a timescale that is one to two orders of magnitude longer. Even on the low end of this range, the heat flux pulse will result in flash melting for tungsten or a ‘bloom’ for carbon. Simulations are underway to determine if such high transient thermal loads will result in bulk melting of the tungsten PFCs, which can lead to permanent deformation of the divertor target tile surfaces and negatively affect operations. Even if no bulk melting is expected, the flash melting of tungsten or the erosion of carbon PFCs would result in significant dust production, which can also negatively affect operations once a significant amount has accumulated – affecting core impurity content, as a fire hazard, and for tritium retention. An emphasis is being placed on disruption mitigation in SPARC making use of the ITER ‘disruption budget’ approach to determine what would be acceptable operational protocols.

In addition to thermal loads, disruptions result in significant structural loads on PFCs. Eddy and halo currents that are driven by the disrupting plasma lead to significant mechanical stresses in the material and, more importantly, in structural mounting joints. These loads have
been quantified by Sweeney (Submitted) and efforts are underway to design in-vessel components that will be able to withstand these forces.

Both the steady state heat fluxes (figure 2) as well as the transient loading from ELMs and disruptions can severely stress PFCs. The design of the SPARC divertor to safely accept the steady state heat fluxes with strike point sweeping is discussed in the next section along with the plan of incorporating transient loads into the design process.

3. Mitigation of steady-state heat load through strike point sweeping

In order to mitigate the high steady-state surface heat fluxes shown in figure 2, the baseline design scenario in SPARC is to sweep the strike point for the duration of the 10 second flattop. Due to the relatively short pulse length in SPARC and the strike point sweeping, the divertor targets can be designed to operate without active cooling during the pulse. As such, they can withstand much higher heat fluxes than a long-pulse, actively cooled divertor since steady state thermal equilibrium is never reached and the design is not limited by the coolant temperature and flowrates. Instead, the maximum heat flux is limited by how quickly and how often the strike point is moving past a point along the surface relative to the material’s thermo-mechanical properties. With each pass of the strike point, the surface temperature of the PFC spikes locally but cools dominantly via conduction into the bulk material. Repeated passes cause the surface temperature of the PFC to ratchet up slightly, but small in comparison to the initial temperature rise when the strike point passes a given location. In order to examine this behaviour, and understand its limitations, a simple 1D heat equation solver was built to simulate a 2 cm thick PFC with an insulated boundary condition at one edge (vessel or mounting plate) and the input surface heat flux as a function of time on the other edge. The 1D heat equation was solved numerically using MATLAB’s partial differential equation package. The initial temperature was set to 700 K for all the scans to mimic the average temperature rise of the divertor target surface close to the end of the 10 second flattop. The temperature dependent thermal properties of pyrolytic graphite were used (Incropera 2007) in the carbon-based PFC analysis, to take advantage of its anisotropic thermal conductivity properties, the high in-plane $\kappa_{||}$ was used. The same qualitative results of these scans were observed when the temperature dependent properties of tungsten were used and are thus not shown. The temperature dependent material properties are shown in figure 4 for completeness. Blackbody radiation was not included, which can aid in suppressing the peak temperature achieved per sweep.
FIGURE 4: Temperature dependent thermal properties of pyrolytic graphite and tungsten (Incropera 2007). Top: the in-plane ($\kappa_\parallel$) and cross-plane ($\kappa_\perp$) thermal conductivity of pyrolytic graphite note the different y-axis units. Middle: the thermal conductivity of tungsten. Bottom: the specific heat capacity of tungsten (blue) and pyrolytic graphite (orange).

To start, the sensitivity of the strike point sweep to the assumptions discussed in section 2 are assessed. First, the sensitivity of the peak temperature rise due to a passing strike point is examined as a function of the surface heat flux profile shape ($\lambda_q, S$) and total power. The divertor heat flux assumptions used are the same as discussed in section 2 but the radiation fraction is set to zero. Figure 5a shows two heat flux profiles at the two extremes of the $\lambda_q$ scaling predictions. Figure 5b shows the resulting peak change in surface temperature ($\Delta T$) resulting from the heat flux profile being scanned past a point at 0.5 m/s. As may be expected, the peak surface $\Delta T$ decreases with increasing values of $\lambda_q$. Increasing $\lambda_q$ from 0.18 mm to 0.4 mm (an increase by a factor of 2.2), results in the peak surface $\Delta T$ decreasing from 991 K to 629 K (a decrease of 37%). Figure 6 shows a similar profile shape scan but now as a function of $S/\lambda_q$, assuming $\lambda_q = 0.18$ mm. Note that in figure 6b, a second case with a different the total power to the outer divertor ($P_{odiv}$) is given. As $S$ decreases, the peak surface heat flux increases but the peak surface $\Delta T$ starts to plateau. Conservation of energy would suggest that the saturation $\Delta T$ is linear with total power, which is generally consistent with the simulations, though there is some discrepancy because of the temperature dependent thermal properties used. Figure 5 and 6 suggest that there is a clear benefit if the $\lambda_q$ is found to be larger than the
most conservative empirical predictions, although the effect is less than linear. If $S$ turns out to be narrower due to suppressed cross-field transport in the divertor leg, the saturation in $\Delta T$ indicates that it is unlikely to cause a significant increase in divertor target surface temperatures.

FIGURE 5. (a) The surface heat flux profile for two values of $\lambda_q$ as a function of the distance along the target surface. The profiles are consistent with the assumptions made in Section 2 with the exception of the radiated power fraction being set to zero. (b) The peak temperature of the passing strike point as a function of $\lambda_q$.

FIGURE 6. (a) The surface heat flux profile for $S = [0.2,0.5,1] \times \lambda_q$ ($\lambda_q = 0.18 \text{ mm}$) as a function of the distance along the target surface. The profiles are consistent with the assumptions made in this report with the exception of the radiated power fraction being set to zero. (b) The peak temperature due to the passing strike point for two values of total power. $\Delta T$ plateaus as the peak surface heat flux increases.

We have also varied strike point sweep velocity and the fraction of $P_{SOL}$ that is radiated in the divertor legs between the X-point and the target ($f_{rad}$); see figure 7. As discussed in section 2.5, the radiated power is applied back on the target surface as a uniform heat flux so as to conserve the same total power across the scan in $f_{rad}$. However, as the same profile is being swept at different velocities, the total ‘exposure’ time is a function of the strike point sweep velocity – the slower strike point sweep requires more time to pass a point and vice versa. As
a result, for the $f_{rad} = 1$ case – a flat uniform heat flux profile, $\Delta T$ is not constant with respect to sweep velocity. The power is constant but the total integrated energy being simulated decreases as the strike point velocity increases. In general, sweeping faster with a higher radiation fraction reduces the peak temperature on the divertor target. As the sweep velocities increase, the $\Delta T$ begins to asymptote indicating that there are diminishing returns as expected from consideration of the thermal time constants. The current design strike point sweep frequency is ~0.8 Hz, corresponding to a strike point sweep velocity of 0.5 m/s on the inner divertor and 0.7 m/s on the outer divertor. Although it would be good to be able to sweep the strike points faster, the power needed for the poloidal field coils for a strike point sweep frequency of 0.8 Hz exceed their power needs of the initial current ramp up, which are driving the power supply requirements for SPARC. Means to improve the sweep velocity by way of modifying the poloidal field coil designs and their power supplies are being assessed.

![Graph](image)

**FIGURE 7.** Scan of the peak change in temperature as a function of sweep velocity and radiation fraction. At the assumed 50% radiation fraction the peak surface $\Delta T$ starts to asymptote past 0.8 m/s.

To obtain a quantitative prediction of the divertor target temperature response to the strike point sweep, a 2-dimensional thermal model of the divertor target plate is simulated in COMSOL. A slab geometry was used, which will be updated when a more realistic divertor target model is available. The slab is 2 cm thick and 0.43 m long in the poloidal plane to simulate the outer divertor high heat flux target surface extent and the strike point is swept across at 0.7 m/s. The temperature dependent thermal properties shown in figure 4 of both pyrolytic graphite and tungsten (Incropera 2007) were used, with the simulation initial temperature set to 298 K. Note that the high in-plane thermal conductivity was only applied into the material, in the direction normal to the surface where the heat flux is applied. The lower cross-plane thermal conductivity is used in the other direction of the slab. This is a conservative approach since for the tile design, the planes of the material are aligned with the poloidal plane of the tokamak (the 2D plane being simulated) with the ‘poor’ cross-plane direction of the material oriented in the toroidal direction. Figure 8, shows the simulation results of sweeping the outer divertor strike point for the 10 second duration of the flattop. We can see that given the current surface heat flux assumptions (figure 2) on the outer divertor target, the surface of a tungsten PFC would quickly cross the recrystallization temperature (~1700 K) after a few passes. Thermal cycling of the material across the recrystallization temperature is likely to result in embrittlement and the formation of micro-cracks similar to that observed in laboratory
ELM testing. Significant testing of tungsten PFCs will be performed to ensure that they can survive the design lifetime of the device. In contrast, the surface temperature excursions of the pyrolytic graphite simulation have some margin up to the carbon bloom temperature of 1900-2100 K (Ulrickson 1990). Current efforts are focused on including the disruption structural loads in the PFC design and the ELM heat loads on top of the strike point sweep.

FIGURE 8. Simplified COMSOL 2D thermal model of the outer divertor strike point sweep over a 10 sec flattop and the subsequent cool-off for both a tungsten and pyrolytic graphite PFC material. The surface temperature of the PFC is plotted as a function of time at (a) the middle of the sweep and (b) near the turn around point of the sweep. Note that the variation in $\Delta T$ for each peak is due to the coarseness of the computational grid.

4. Modelling of the SPARC divertor and access to dissipative divertor regimes

Even though the baseline design scenario for operations is to take a conservative approach and assume moderate radiation fractions with an attached strike point, it is also of interest to understand what can be done to access dissipative divertor regimes. Standard boundary plasma-neutral simulation tools have been developed by the fusion boundary community and have been applied to simulate SPARC in UEDGE (Rognlien 1992) and SOLPS-ITER (Weisen 2015, Bonnin 2016). The use of both codes will allow for cross-code comparisons to increase the confidence in their results and projections for SPARC. The SPARC operating parameters are significantly beyond the limits of validation of these simulation tools and at present, there are difficulties in verifying solutions and assessing convergence for such high density and short gradient scale-length plasmas. The modelling studies, although ongoing, are presented herein, along with a description of the proposed future work to assist in refining the baseline design guidelines for the SPARC divertor, and are designed to play to the strengths of the different packages. Furthermore, the Lengyel model as used by Reinke (2017) is applied to the SPARC V2 primary reference discharge to assess the impurity fraction needed to access moderate to high SOL radiation fractions and highlight the concern that the attainment and control of a dissipative divertor regime may be challenging, at high power density and relatively low upstream plasma densities currently desired for optimizing the core plasma performance.

4.1 Preliminary UEDGE simulations with 1% Carbon

In this section, we present results from UEDGE (Rognlien 1992) modelling of the SPARC V2 double-null equilibrium in an up-down symmetric simulation. UEDGE solves the Braginskii fluid equations for the plasma with specified anomalous cross-field transport
coefficients. A fluid model is utilized for the neutral transport which is appropriate for a short neutral mean free path regime as is anticipated in the SPARC divertor. For impurity radiation, the fixed-fraction impurity model is used and is tied to a fraction of the ion/electron density. Impurity radiation rates are obtained from the ADPAK code (Hulse 1983). The equations are discretized by finite volume method on a 2D spatial grid, and solved by fully implicit methods, finding exact steady states to machine precision.

For setting up the magnetic geometry, UEDGE uses the MHD equilibrium of the SPARC primary reference discharge and creates a grid conformal to the magnetic flux surfaces in the region of \( \psi_N \approx 0.98 - 1.02 \), where \( \psi_N \) is poloidal magnetic flux normalized to the value at the separatrix. The simulations are performed in the lower half-domain, below the midplane, assuming a balanced double null configuration, with corresponding upstream profile and power sharing assumptions. For the simulation results shown here, the target plate is chosen to be normal to the flux surfaces. Boundary conditions are specified to match SPARC upstream operation parameters and to determine as accurately as possible the variables of interest at the outer boundary, approaching the limiter and walls. At the core interface, the boundary conditions use fixed plasma density, fixed ion and electron power, and zero radial derivative of the parallel ion velocity. In the simulations, the total power into the edge domain is generally set to 20 MW, although this will be pushed as high as 29 MW to investigate SPARC operation at full power. The common flux region (edge of grid towards PFC surfaces) boundary conditions are extrapolated ion/electron density, ion temperature, and electron temperature. The private flux region has boundary conditions for density and temperature specifying a gradient scale length of 1 cm (value divided by its derivative in the radial direction). Assuming that material surfaces come to saturation, the neutral recycling coefficient is set to 1 for all material wall boundaries. This means that each ion crossing the boundary reappears locally as a neutral, and none are injected into the system or pumped. The standard sheath boundary conditions are used at the divertor targets.

The cross-field diffusive transport coefficients relate density and energy fluxes to their gradients, i.e. \( \Gamma_{n,\perp} = -D \nabla n \). The radial transport coefficients are user-specified to match predicted midplane plasma profiles, heat flux profiles on the target plates, and inner/outer divertor power sharing, from empirical scalings and experimental data. The simulation domain was split into three regions: from the midplane to the X-point of the high field side (imp); from the midplane to the X-point on the low field side (omp); and the two divertor leg regions. The transport profiles shown in figure 9 were constant along flux surfaces in their respective

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**FIGURE 9.** Particle \((D)\) and energy \((\chi)\) cross-field diffusion transport coefficients profiles at the inner (imp) and outer midplanes (omp).
domain. Across the entire simulation domain, the radial ion momentum diffusion coefficient is set to 0.5 m²/s. While the radial energy diffusion coefficient (χ) for both ions and electrons is set to the values shown in figure 9, with low transport on the inboard side and enhanced transport with a barrier to simulation the pedestal on the outboard side. A flat χ of 0.1 m²/s was used across the divertor legs consistent with outboard side value but without the transport barrier. The depth of the energy transport barrier is used to control the exponential decay length of the parallel heat flux profile at the entrance to the outer divertor. Decreasing χ on the inboard side results in decreased power to the inner divertor target, and this was set such that the divertor power sharing was 1:4 near the X-point at full power with no impurities consistent with experimental observations of double null power sharing. The radial density diffusion coefficient, D, is set to the values shown in figure 9. D, along with the core density, is set to obtain the desired upstream density profile. The inboard and outboard D profiles shown in figure 9 were constant along flux surfaces all the way from the midplane to the divertor target. The D profile which rises in the radially outward direction on the outboard side is a stand-in for convective transport (Umansky 1998). UEDGE runs including the convective transport terms are being developed. Some of the presented simulation parameters are also adjusted in individual cells to ensure converging solutions; the solver sometimes has difficulty converging due to cells with very low density, especially near the end-points of the divertor targets. Experimental data informing the value of the diffusion coefficients everywhere in the plasma are scarce, and no data exist on long-legged divertor transport, so the values used here are mostly ad-hoc and tuned to match experimental profile predictions where available for fully attached divertor conditions with no impurities.

A case with 1% carbon fraction, P_{\text{Sol}} = 20 MW, and a separatrix density of $1 \times 10^{20}$ m⁻³ is shown in figures 10-12. In figure 10, the density and temperature at the outer divertor target are mapped to the outer midplane. Target plate plasma temperatures are low while densities are high, indicating high recycling conditions. The inner target profiles are not shown but and the target plasma temperature ($\sim 3.5$ eV) is significantly reduced from midplane values and the target surface heat flux has also dropped to manageable levels of <1 MW/m². Figure 11 shows the convected and conducted power from ions, electrons, and neutrals entering the outer divertor, in the direction parallel to the magnetic field near the X-point. A single exponential fit to the data indicates a heat flux width of 0.215 mm, close to the empirical prediction of 0.18 mm by Eich (2013). Finally, a breakdown of the power to the outer divertor target is shown in figure 12. Convection, conduction, and surface recombination are the dominant contributors; radiation has relatively little impact. This picture is different for the inner divertor target, where heat flux to the surface is dominated by radiation. The broadening of the heat flux profile relative to the x-point is being investigated. There are likely some effects due to neutrals and side wall interactions in the long leg divertor geometry (Umansky 2017). These power loss mechanisms might explain the low target temperatures despite the relatively low impurity fraction.

These results suggest that a combination of the long tightly baffled divertor leg geometry used in the simulations and a low level of impurities may be sufficient to significantly reduce the peak heat fluxes to the divertor target as compared to the profiles shown in figure 2. However, care should be taken in the interpretation of quantitative results from a single UEDGE simulation due to the significant uncertainty in the assumed cross-field transport coefficients and other inputs. Sensitivity scans are planned to increase the overall confidence. Furthermore, UEDGE simulations are being developed to provide a wide 2D scan in both power and impurity fraction with carbon and neon. This will help provide a map of the parameter space at which detachment is possible and the overall qualitative trends which can be used to inform divertor design decisions. Additional simulations are also planned for lower
single null discharges. These full domain simulations will allow for drift terms to be included and will provide high fidelity predictions of SOL and divertor behavior in SPARC.

FIGURE 10. Density and temperature profiles at the outer divertor target plate mapped along flux surfaces to the outer midplane.

FIGURE 11. Convected and conducted power including electron, ion, and neutral energy flux parallel to the magnetic field at the outboard side X-point, mapped along flux surfaces to the outer midplane. The total power crossing into the outer divertor is shown in parentheses in the legend. The dashed line shows a single exponential fit to the data from which the heat flux width is obtained.

FIGURE 12. Total power flux to outer divertor target and its components mapped along flux surfaces to the outer midplane. The total power of each component is shown in parentheses in the legend.
4.2 Preliminary SOLPS simulation results

The SOLPS-ITER code (Bonnin 2016) is applied to SPARC to study particle balance and impurity seeding properties of SPARC. Specifically, the simulations will be used to inform requirements on the fuel ion puffing location and flow rate, estimate the neutral pressure in the sub divertor as needed to design pumping systems, and estimate impurity flow rate requirements and optimal puffing location to reduce the divertor heat flux via dissipation. Simulations are performed for SPARC V2 but in an unbalanced double null topology biased towards the lower divertor. The distance between the two X-points when mapped along flux surfaces to the outer midplane is 2 mm, which is large compared to the baseline design SOL heat flux width discussed in section 2.2. Figure 13 shows the SOLPS-ITER grid, with the colored regions used for plasma and neutral transport, and the black triangulated region treated as vacuum for neutral transport only. The initial simulations shown here are unseeded. The code inputs are core boundary conditions of 29 MW of power into the SOL, equally split between electrons and ions, and a density of $1.5 \times 10^{20}$ m$^{-3}$, which corresponds to a separatrix density at the outboard midplane of $1.0 \times 10^{20}$ m$^{-3}$. The indicated surface is used to pump neutral particles, with a recycling coefficient of 0.9. Cross-field transport coefficients of $\chi_\perp = 0.01$ m$^2$/s and $D_\perp = 0.1$ m$^2$/s are chosen to reproduce the extrapolated heat flux width (figure 14), an exponential fit near the separatrix gives a decay length of 0.14 mm.

Figure 13. SOLPS-ITER grid used for SPARC simulations. Colored regions indicate mesh for plasma transport, while the colored mesh and black triangles are available for kinetic neutral transport.
Figure 14. Parallel heat flux density at the outboard side X-point, mapped along flux surfaces to the outer midplane. Note that the parallel heat flux shown is higher than in figure 11 of the previous UEDGE simulations which used a lower $P_{SOL}$ and assumed perfectly balanced double null power sharing.

Figure 15. Radial profiles as a function of distance along the outer divertor target of a) electron density, b) electron temperature, c) deposited heat flux carried by the plasma, and d) deposited ion particle flux.
Figure 15 shows profiles of the electron density and temperature, as well as the particle and heat fluxes carried by the plasma at the lower outer divertor. Without seeding the divertor is strongly attached, with high electron temperature \( T_e^{\text{peak}} \approx 300 \text{ eV} \) and modest density \( n_e^{\text{peak}} \approx 1.7 \times 10^{21} \text{ m}^{-3} \). The peak heat flux density \( q_{\text{surf}}^{\text{peak}} \) is \( \approx 180 \text{ MW/m}^2 \), with \( \sim 22 \text{ MW} \) of power deposited at the outer target. Compared to Figure 2, the difference in peak surface heat flux is largely due to the lower incident angle of the magnetic field at the target used in the SOLPS simulation (~0.8 degree at the OSP, 2 degree in figure 2). Accounting for this difference, there is reasonably good agreement between the SOLPS modelling and simple estimates with the Eich model which increases confidence in the simulation output. The peak particle flux density \( f_{\text{peak}}^{\perp} \) is \( \sim 10^{24} \text{ m}^{-2}\text{s}^{-1} \), occurs further into the SOL as compared to \( q_{\text{surf}}^{\text{peak}} \), and is similar in magnitude to typical values for partially detached conditions in ITER (Pitts 2019). The values are similar despite the different transport regimes, because of the much higher divertor T\( _e \) in SPARC.

### 4.3 Lengyel model estimates of impurity fraction

As a parallel scoping of the operating space of the divertor versus impurity fraction we have utilized the simple Lengyel model applied in (Reinke 2017). The model derives the required impurity fraction that is consistent with a target temperature that both satisfies the desired reduction of \( q_{\parallel} \) due to radiation as well as the sheath heat flux boundary condition, assuming a sheath heat flux coefficient of \( \gamma = 7 \). This is shown in figure 16 for connection length, \( L_{\parallel} = 24 \text{ m} \), upstream density, \( n_u = 1.0 \times 10^{20} \text{ m}^{-3} \) and using a non-coronal radiation model for neon and argon assuming a finite impurity residence time, \( n_e \tau = 0.5 \times 10^{20} \text{ m}^{-3}\text{ms} \).

SPARC highlights an interesting behaviour found in its parameter space that is a result of its access to high parallel heat fluxes and low \( f_{GW} \) operational point. At low upstream \( q_{\parallel} \approx 0.5 \text{ GW/m}^2 \) there is the expected behaviour that as the impurity fraction is increased, the SOL radiation fraction increases monotonically. An arbitrary SOL radiation fraction is possible, although the non-linearity suggests that control may be a challenge. At higher \( q_{\parallel} \), the behaviour fundamentally shifts, and unless the impurity fraction reaches a higher level there is no solution, i.e. there is no target temperature where the radiation fraction reaches 10’s of percent and the sheath heat flux boundary condition is satisfied. The lack of a solution may in part be due to the Lengyel model analysis assuming a conduction-limited SOL which is not realized at high \( q_{\parallel} \) and low \( n_u \) where a sheath-limited SOL is expected. This is interpreted as due to the high attached target temperature, the SOL can progressively fill with impurities, in this case neon or argon, without causing any substantive radiation until a tipping point is reached, the target temperature collapses, and the radiation fraction jumps to a high level. This ‘minimum’ radiation fraction progressively increases as \( q_{\parallel} \) is increased as shown in Figure 16. Furthermore, there exists a hot and cold branch where the sheath heat flux boundary condition can be satisfied, due to the non-linear nature of the impurity cooling curves. Increasing the upstream density moves the transition to this bifurcated behaviour to higher \( q_{\parallel} \), and it is also sensitive to the selection of impurity and assumption of its \( n_e \tau \). The Lengyel model suggest that SPARC may not have arbitrary control over its divertor radiated power. When target plate temperatures start to decreases from the \( \sim 300 \text{ eV} \) predictions of the SOLPS simulations (figure 15), divertor radiation fractions may already be significantly higher than the 50% assumed in section 2. This is in contrast with the results from UEDGE and efforts are underway to understand the discrepancies. Further analysis is required to understand the behaviour of the model and comparisons made to the other simulations to increase the confidence in this result.
5. SPARC X-point target divertor

Divertor tokamaks rely on ‘conventional’ configurations known as the ‘vertical target’ (C-Mod, JET, AUG, ITER) and ‘horizontal target’ (DIII-D, JET) that have been in use for many years and are fairly well understood. In order for these ‘conventional’ divertors to be viable in a fusion pilot plant, efforts have grown to address whether large core radiation fractions (e.g., 95%) can be used to reduce $P_{\text{SOL}}$ in a reactor to levels that a conventional vertical target will accept safely, <10 MW/m$^2$ (Kukushkin 2013, Wischmeier 2015, Bernert 2017). ‘Advanced’ divertor configurations have been pushed forward as an alternative solution that may not require such high core radiation fractions, and experiments are currently being pursued or are planned on a number of different devices: TCV (Reimerdes 2017), MAST-Upgrade (Katramados 2011), ASDEX-Upgrade upper divertor (Lunt 2017), and Italian-DTT (Ambrosino 2019). The configurations that have been actively discussed within the boundary plasma community to-date include: a) ‘snowflake’; b) ‘X-divertor’; c) ‘Super-X’ (SXD); and ‘X-point target’ (XPT). Simulations have suggested that these advanced divertors have significant advantages over ‘conventional’ configurations in terms of their access to dissipative divertor regimes and their ability to provide a wide detachment power window (Umansky 2017). Of the ‘advanced’ divertors, the XPT divertor geometry was found to have the widest detachment power window which is advantageous from a power plant operations perspective (Kuang 2018). Therefore, in order to assess the performance of the XPT divertor at reactor level power and plasma densities, it is being incorporated into the SPARC V2 design (figure 1b).

Due to the high coil currents needed to pull the secondary X-point, the XPT divertor geometry is only accessible at somewhat reduced plasma current ($B_0 = 12.2$ T, $I_p \leq 5.7$ MA). Even then, preliminary assessment of the core performance using the POPCON analysis discussed in Creely (Submitted) indicate that $Q \sim 1.9$ with 37 MW of fusion power may be achievable. For the divertor this corresponds to $P_{\text{SOL}} = 22$ MW and a heat flux width of $\sim$0.25 mm (based on Eich 2013), which maps to a parallel heat flux entering the divertor that is higher than in present day devices. Some of the key issues that SPARC will attempt to address with the XPT divertor geometry include the sensitivity of power dissipation to the secondary null location in flux space relative to the primary separatrix, how sensitive the divertor detachment location and dissipation is to external controls, and its response to transients. Although it has

![Figure 16. The radiation fraction ($f_{\text{rad}}$) as a function of the impurity fraction ($f_Z$) for (a) neon and (b) argon, calculated using the Lengyel model.](image_url)
yet to be started, modelling of the XPT with UEDGE and SOLPS-ITER is planned in order to help optimize the design and determine the optimal feedback scheme.

6. Discussion

Table 3 summarizes how SPARC V2 compares against present day devices, ITER and ARC for the normalized maximum conducted power entering the SOL. Note that the maximum conducted power used in the table is defined as the maximum available auxiliary power in present day devices, while for SPARC V2, ITER and ARC, the projections for $P_{SOL}$ assuming no core radiation are used. Though a simplification, table 3 nonetheless provides an understanding of the extrapolation involved, and the significant uncertainty, in projecting experimental results from present day devices to SPARC V2, or worst, to an ARC-class pilot plant. As a result of the large uncertainty, the conservative approach had to be taken when outlining the baseline design scenario for the SPARC V2 divertor or potentially risk SPARC failing to reach its primary mission of $Q>2$. Even without the added impact of transient loading from ELMs and disruptions, simulations of the strike point sweeping in section 3 shows that the conservative baseline design scenario is pushing the limits of materials to withstand these heat fluxes. Using this same conservative approach, it will likely be impossible to design a divertor for an ARC-class device. This highlights the importance of intermediary devices like SPARC V2 and ITER for reducing the uncertainty in developing design guidelines for a future fusion pilot plant. SPARC will be a valuable platform for providing crucial data for designing the divertor in a fusion pilot plant. How valuable though will depend on the available diagnostic set which is currently being evaluated.

The focus of diagnostics on SPARC will be to inform future divertor design efforts. Potential areas for investigation would include but are not limited to: divertor target power sharing and the ability to maintain double null; the divertor target heat flux profile; and access to dissipative divertor regimes. Conservative assumptions had to be made in section 2 with regards to these topics when developing the design guidelines for SPARC V2 due to the uncertainty in the extrapolations. Operations on SPARC will help to inform how well perfect double-null has to be maintained in order to share power between the upper and lower divertors as well as reduce power to the inner divertors. If it has to be maintained to within $\sim \lambda_q$, then it could potentially be a big impact on power plant divertor designs. Understanding how power-sharing measurements may scale and impact next step device design though, depends on the ability to measure $\lambda_q$ on SPARC. Combined, this will inform the magnitude of the heat exhaust challenge for a fusion pilot plant. Between the various scalings shown in section 2.2, there is a factor of 2.5 range in the projections to SPARC V2. In addition, results from codes like XGC predict a broadening of the heat flux width for ITER (Chang 2017). A prediction of the $\lambda_q$ in SPARC by XGC has yet to be made. Although not as low as in ITER, the SPARC V2 reference H-mode has a $\rho'=0.0027$ (Creely Submitted). It is possible, that the increase in the turbulent electron effects proposed as the driving mechanism for the broadening in the predicted $\lambda_q$ of ITER, may be observed in SPARC (Chang 2018). Definitely, measurements of SPARC V2 $\lambda_q$ will be an important data point for benchmarking empirical scalings and validating first principles computational models. Lastly, the inclusion of the XPT divertor geometry in SPARC is driven by the desire to assess its viability for an ARC-class device. Key issues of interest for this topic have already been mentioned in section 5.
TABLE 3: A comparison of the normalized maximum SOL conducted power for ASDEX-Upgrade (Streibl 2003), JET (Garzotti 2019), Alcator C-Mod (Marmar 2006), SPARC V2 (Creely Submitted), ITER (Casper 2013), and ARC (Kuang 2018). Note that the maximum conducted power ($P_{SOL,max}$) corresponds to the maximum available auxiliary power for the present-day devices while for SPARC V2, ITER, and ARC, it refers to the projections for $P_{SOL}$ assuming no core radiation.

7. Conclusions

In conclusion, heat flux mitigation will be a key challenge for SPARC operations and an opportunity to extend our experimental knowledge into the fusion power regime, with the baseline divertor design scenario projecting to peak perpendicular surface heat fluxes on the inner and outer divertor target of ~250 and ~350 MW/m² respectively. Initial simplified calculations of the divertor target surface heat loads suggest that melt temperatures for tungsten and ‘bloom’ thresholds for carbon will not be reached if the strike point is swept at 0.8 Hz across the target surfaces for the duration of the 10 second flattop. Transient loading from ELMs and disruptions have yet to be incorporated into the design of the PFCs, however estimates of the thermal and structural loads have been made with wide uncertainties remaining due to the significant extrapolations made to conditions perceived for SPARC. As in ITER, these appear to challenge the ability of materials to withstand unmitigated transient fluxes without damage. Dissipative divertor operation may not be needed for achieving the SPARC primary mission of $Q>2$ but SPARC represents a unique opportunity to study divertor operations under reactor conditions. Simulation efforts are currently being made to develop predictions of SOL and divertor conditions to inform design decisions so as to ensure that dissipative divertor regimes are not precluded. Furthermore, SPARC is being designed to allow for XPT divertor operations. Even at the reduced plasma current needed to pull the secondary X-point into the divertor volume, with a $P_{SOL} = 22$ MW and a heat flux width of ~0.25 mm, it will still present a significant heat exhaust challenge for the divertor and will be useful in assessing whether the XPT divertor geometry is a viable option for a future fusion pilot plant. Ideally measurements made on SPARC will be able to reduce significantly the uncertainty in projections of divertor performance and to increase the confidence in the design of a fusion pilot plant.

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References


Riccardo, V., & Loarte, A. 2005 Timescale and magnitude of plasma thermal energy loss before and during disruptions in JET. *Nucl. Fusion* **45** (11), 1427.


