1. DIVERTOR APPROACHES

A DEMO-class tokamak must produce significant fusion power, while at the same time justifying economic viability. To optimize the return on investment, it is important for a DEMO reactor to operate in regimes favorable to “high performance,” particularly with regard to optimizing the plasma energy confinement time (τ_E) and the plasma beta (β_T). Higher values of τ_E and β_T are more readily obtained as the plasma shape becomes increasingly “triangular,” and this is achieved more naturally in a balanced double-null (DN) configuration rather than in a single-null (SN).

While the SN shape has only one divertor and the DN two distinct divertors, it does not necessarily follow that a design based on the DN shape is significantly more costly or complicated to build than a design based on the SN. For example, because pumping on the two inner divertor legs of a DN plasma has been shown to be largely superfluous in maintaining density control, there is no need to install pumping hardware on the difficult to engineer high-field side of the core plasma. Similarly, the DN requires less heat protective armor at the inner divertor targets. In normal operation more than 80% of the power from the core flows into scrape-off layer on the outer, low-field side, which is magnetically disconnected from the inner divertor targets. But of course provision must be made for fault conditions and some magnetic imbalance owing to control system precision that might result in temporary SN operation, in which case the inner divertors have to be able to take about half the power that would flow to the outer SN divertor. When the physics and engineering considerations are taken together, the DN offers a greater promise of high performance (and economic viability) for DEMO, as evidenced by the implementation of DN for the ARIES series. A double null design has also been chosen for JT-60SA, EAST, KSTAR, MAST, and NSTX, all the most recent tokamaks. Hence the baseline design for FDF is double null.

The ability to operate the DN depends crucially on how well the shaping/control coils can maintain divertor balance. In both FDF and DEMO, protecting these coils from neutron damage is crucial. Even with shielding, these coils would not be as close to the X-points as present day DN tokamaks, meaning that maintaining control over the magnetic balance would be a more difficult task for DEMO. FDF would take on these problems and the solutions found for FDF would be relevant to DEMO.

There are two overarching principles of effective divertor design.

1. The divertor must present a perfectly toroidally flat surface to the divertor field lines.
2. No edges can be allowed to protrude up into the huge parallel heat flux flowing along the divertor field lines.

The reason for these two principles is that the parallel heat flux is so enormous, over 0.5 GW/m^2, that no material can stand up to that. Any edge protruding up into such a heat flow will be very rapidly eroded away, producing damage and an large impurity source to the core plasma and a source of material to codeposit Tritium.

In the DIII-D tokamak, axisymmetric divertor rings were installed with a flatness of 2 mm over 5 m distance. Tiles on these rings have had their edges aligned to 0.1 mm accuracy. Tiles installed on the less even floor of the vacuum vessel were ground to achieve the same 0.1 mm tile to tile edge alignment. The result of this precision alignment is that infrared thermography pictures of the divertor in DIII-D show total axisymmetry; one can hardly see the tile edges.

This quality of alignment amounts to an angle of about 0.2 deg in the toroidal direction. The baseline design of FDF’s maintenance scheme will facilitate such precision alignment The divertor can be built in a factory setting as a single precision toroidal ring with precision placement of tiles on it. Slight fish-scaling of the tiles may be done to hide edges; FDF does not plan to reverse the directions of the magnetic fields in its operation. With such precision alignment, there is no practical limit to the use of flux expansion and the problem of heat flux handling becomes axisymmetric and can be reduced to looking at the poloidal projection of the heat flux. Hence the baseline divertor design for FDF is a conventional tilted plate type with the intention to make maximal use of flux expansion.

The lower half of the poloidal cross-section of the FDF double-null plasma and vacuum vessel is shown in Fig. 1. The upper half of the vessel is the mirror image of the lower half. A divertor plate, shown in red in Fig. 1, is oriented to the incoming poloidal flux surfaces at a shallow angle in order to minimize the peak in the heat flux profile by increasing the wetted area of the divertor surface. Pumping is done from both near the separatrix strike point and from farther out into the scrape-off layer. Deuterium gas puffing can be added near the neck of the divertor to enhance the flow of the fuel ions into the divertor in order to increase the trapping of impurities in the divertor. Some significant degree of closure is designed to impede the mobility of neutrals away from the target areas and facilitate pumping. In addition, this divertor design must be compact enough to allow for adequate shielding of the toroidal field and shaping coils beneath and above the two divertors. Simple projections and modeling calculations indicate that for this baseline divertor the peak poloidally projected heat flux will stay below 10 MW/m^2, which we are certain can be engineered. However we are also going to look at segmenting the divertor coil to enable larger flux expansion, as has been advocated by the University of Texas – the so-called X-divertor. With the precision alignment discussed above, there is no practical limit to the use of increased flux expansion.

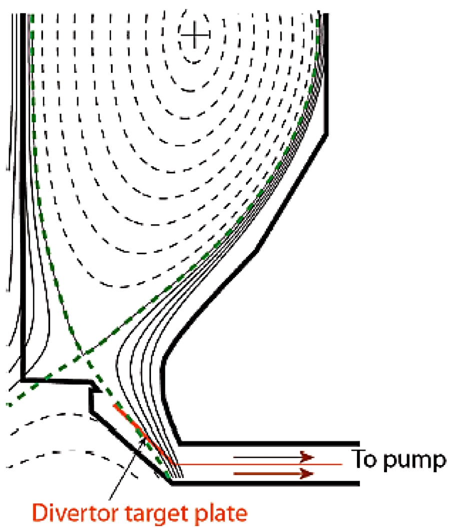

Fig. 1. The lower half of the double-null plasma equilibrium and FDF vessel is shown. The upper half of the plasma and vessel are mirror images of the lower half.

Advanced divertor concepts, such as the Super-X divertor (U. of Texas) and the Snow Flake divertor (Lawrence Livermore National Laboratory), can be viewed as forms of the flux expansion approach to control divertor power exhaust and would be relatively straightforward to implement in DN devices, such as FDF or DEMO, because high field side pumping of the inner divertor legs is unnecessary and the power flow to the inner divertor targets is much smaller than that to the outer divertor targets. Each concept uses a much larger wetted area at the outer divertor targets to spread out the heat loads. FDF’s maintenance scheme of demounting the TF coil and allowing crane lift of all components inside the TF coil allows PF coils to be placed where they need to be for the SX divertor. FDF may be able to mount a rather unique and compelling test of either the Super-X and Snow Flake concepts; hence these ideas are being pursued as secondary options for FDF. Although FDF does not require the SX divertor for heat flux handling, these is the possibility that the long divertor field lines afforded by the SX divertor can more greatly decouple the core plasma from the plasma at the divertor plates, thus offering increased possibilities for radiative divertor operation, and/or keeping neutrals away from the core plasma to increase confinement, and/or containing eroded materials in the divertor chamber. Tests of these physics aspects of long divertor field lines may be doable in the DIII-D tokamak in the near future.

2. HEAT FLUX HANDLING

We assume that the radial profile of the heat flux in the scrape-off layer (SOL) has an exponential form. Assuming attached plasmas with zero SOL and divertor radiation, an expression for the peak heat flux at the outer divertor targets can be written:

$$Q_{div,S} = \frac{\left[P_{heat} \times \left(1 - f_{rad}\right) \times f_{out/total} \times f_{gradB/total} \times \left(1 - f_{pfr}\right) \times \sin\alpha\right]}{2\pi \times R_S \times f_{\exp} \times \lambda p}$$

where

$Q_{\mathrm{div,S}}$:	The peak heat flux at the divertor strike point
P_{heat}:	The total heating power = 108 MW for the FDF nominal case
R_{S}:	The major radius of the divertor strike point = 2.3 m
λ_{q}:	The *midplane* exponential heat flux scrape-off width = 0.007 m
f_{exp}:	The flux expansion at the divertor target = 4
α:	The angle between the divertor incline and the divertor separatrix = 10°
f_{rad}:	The ratio of total radiated power to total input power = 0.54. The total radiated power is calculated as the sum of 42% impurity line radiation from the core plus the explicitly calculated Bremstrahlung power of 13.2 MW.
$f_{\mathrm{out/total}}$:	The ratio of power flowing into the outboard SOL to the total power flowing into the SOL = 0.86
$f_{\mathrm{gradB/total}}$:	The ratio of power striking the outboard divertor in the grad-B direction to the total power striking both the upper and the lower outboard divertors = 0.5
f_{pfr} :	The fraction of power flowing into the private flux region = 0.1

The peak heat flux prediction of this simple formulation has been compared with DIII-D data in strongly attached plasmas and generally found to overestimate the measured peak heat flux at the outboard divertor strike point by about 20%. Taking the base case parameters given above, we find that $Q_{\mathrm{div,S}} \approx 8.3$ MW/m^2. A similar calculation for the inner divertor targets indicate that the peak heat flux would be roughly half this value.

Generally, by pursuing radiative divertor operation with deliberately introduced impurities to radiate in the SOL and divertor, we can lower this peak heat flux in DIII-D by up to a factor of 3 while still remaining attached and a factor of 4-5 if we go to the detached divertor state [1].

We relied on extrapolations of available experimental data in choosing “reasonable” values for the required input. For example, $f_{\mathrm{out/total}}$ comes from heat flux measurements in a series of experiments in DIII-D for which H-mode DN plasmas were determined to have a ratio of outboard-to-inboard power flow ≥ 6:1, as discussed in Ref. [1]. About 10% of the power flow to the divertors spills into the private flux region (i.e., $f_{pfr} \approx 0.1$), again based on analyses of single-null (SN) and double-null infrared camera data.

A major source of uncertainty is what the exponential heat flux scrape-off width λ_q at the midplane for conditions expected for high triangularity, highly powered tokamaks. For this study we use Loarte's formulation for H-mode plasmas [2] and estimate $\lambda_q \approx 0.7$ cm.

There are a number of factors which can increase or decrease the peak heat flux from the simple field line following estimate in the equation above. We collect these in Table 1. We see in the table the potential for a cumulative potential reduction of order a factor 18, considering the snowflake and SX divertor options as forms of flux expansion. We see also the potential to increase the peak heat flux a factor 6. Overall, we conclude the peak heat flux in FDF will probably stay below what can be reliably engineered, 10 MW/m^2.

Table 1
Factors Which Can Alter Peak Heat Flux from Simple Field Line Following Estimates

Factor	Increase/Decrease
Increased Core Plasma Radiation	Down 1.3x
SOL and Divertor Radiation	Down 4-5x
Increasing Flux Expansion at the Target Plate, "X" Divertor	Down 2x
Snowflake Type Divertor	Down 1.6x
Super X Divertor	Down at least 2x
Slow Rotation of the Distorted Heat Flux Pattern Produced by the RMP Coils to Suppress ELMs	Down 1.5x
Less Plate Tilt Angle, perhaps 20 deg	Up 2x
Shorter Midplane Heat Flux Falloff Length	Up 2x
Excursions in Control from DN to SN	Up 1.6x

2.1. Preliminary 2-D Modeling of the FDF Divertor

In the previous section we introduced a plausible design for an FDF divertor and estimated the peak heat flux based largely on geometrical considerations. In this section we use 2-D modeling with the SOLPS and UEDGE [3] fluid transport codes to do more detailed analyses of the divertor and scrape-off layer (SOL) plasmas of FDF. As we describe below, each code was used to examine different questions related to FDF divertor and SOL plasmas. For a more expansive discussion of this modeling [4].

Estimates of the peak heat flux at the divertor targets by SOLPS are somewhat more favorable than the estimate made in Section 2 above. In this analysis, the perpendicular heat and particle diffusion coefficients in the scrape-off layer were taken as $\chi_{\perp e} = \chi_{\perp i} = 1$ m^2/s and $D_\perp = 0.3$ m^2/s. In order to find a low-radiation, conduction-limited case to compare the heat flux width to scalings, the gas-puff was raised just until the point where there is some parallel gradient in the electron temperature. These assumptions produced a midplane heat flux falloff length of ~3.5 mm, about 1/2 of the value used in the simple analysis above; hence we expect peak heat fluxes in these simulations about twice the simple results above. SOLPS used a Monte Carlo neutrals code (EIRENE) to simulate neutrals behavior. In these cases, carbon sputtered from the graphite protective tiles is the principal radiator in both the core and scrape-off layer/divertor plasmas. Core radiation is roughly 40% to 45% of the total core heat power of 108.1 MW. Plasma particle drifts in the scrape-off layer and divertor were not considered in this analysis.

Using the divertor geometry of Fig. 1, SOLPS showed that the profiles of electron temperature, electron temperature, and heat flux at the outer target(s) showed significant variation as the deuterium gas puff was varied from zero to levels that yielded a partially detached state. The highest peak heat flux was found to be ≈9 MW/m^2, below the expected value of ~16 MW/m^2 (i.e. twice the estimate made in Section 2). In the detached state, the SOLPS analysis predicted a peak heat flux was ≤2 MW/m^2.

UEDGE was used to assess what effect that injected seed impurities might play in moderating the peak heat flux and produced a rather interesting result with respect to the value of a tilted divertor plate when recycling impurities were injected. Unlike SOLPS, which used a Monte Carlo neutrals code, UEDGE used a fluid neutral model, which included parallel momentum exchange between deuterium ions and deuterium neutrals, to determine the spatial variation of deuterium neutrals. But like SOLPS, the electron thermal, the ion thermal, and the momentum cross-field transport was assumed diffusive with spatially constant diffusivity. Plasma particle drifts were not considered.

UEDGE analysis showed that a 1% argon concentration in the divertor led to a ≈30% drop in the peak heat flux at the (tilted) divertor targets. This is less than what was found when the same calculations were performed on divertor plates that were orthogonal to the divertor flux surfaces. Unlike the SOLPS cases, the divertors in the UEDGE were much more “open” than that described in Fig. 1, and the present hypothesis to explain why a tilted divertor makes a radiating divertor less is that there is increased mobility of the neutrals from the divertor targets out into the main chamber.

The preliminary results from both the geometry-based analysis and the more sophisticated 2-D plasma modeling analysis predict that the peak heat flux at the outer divertor targets to be less than 10 MW/m^2. However, confidence in this result must depend heavily on how well the SOL and divertor plasma transport in an FDF regime is understood, and at present this understanding is somewhat lacking. Still, our result for the peak heat flux, based on analyses using “reasonable” values of $\chi_{\perp e,i}$, $D_\perp$, and/or λ_q , must be considered as a hopeful sign for the viability of the FDF divertors. In future work, we will refine our methodologies, including the addition of particle drifts to the UEDGE and SOLPS analyses. Refinements to the divertor designs, perhaps closing the divertor further in order to better trap neutrals and ions in the divertor, would also be under consideration.

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