ReNeW White Paper:

Innovative divertor development to solve the plasma heat-flux problem

T. Rognlien, D. Ryutov, M. Makowski, V. Soukhanovski, M. Umansky,
R. Cohen, D. Hill, and I. Joseph. LLNL

Motivation and significance: Large, localized plasma heat exhaust continues to be one of the critical problems for the development of tokamak fusion reactors. Excessive heat flux erodes and possibly melts plasma-facing materials, thereby dramatically shortening their lifetime and increasing the impurity contamination of the core plasma. A detailed assessment by the ITER team for their divertor has revealed substantial limitations on the operational space imposed by the divertor performance. For a fusion reactor, the problem becomes worse in that the divertor must accommodate 20% of the total fusion power (less any broadly radiated loss), while not allowing excess buildup of tritium in the walls nor excessive impurity production. This is an extremely challenging set of problems that must be solved for fusion to succeed as a power source; it deserves a substantial research investment.

Material heat-flux constraints: Results from present-day tokamaks show that there are two major limitations of peak plasma heat exhaust. The first is the continuous flow of power to the divertor plates and nearby surfaces that, for present technology, is limited to 10-20 MW/m^2. The second is the transient peak heat-flux that can be tolerated in a short time, τ_m, before substantial ablation and melting of the surface occurs; such common large transient events are Edge Localized Mode (ELMs) and disruptions. The material limits imposed by these events give a peak energy/$\tau_m^{1/2}$ parameter of ~40 MJ/m^{2}s$^{1/2}$ [1]. Both the continuous and transient limits can be approached by input powers in the largest present-day devices, and future devices are expected to substantially exceed the limits unless a solution can be found.

Wisdom of broad-base approach: Since the early 90's LLNL has developed the analytic and computational foundation for analyzing divertor plasmas, and also suggested and studied a number of solid and liquid material concepts for improving divertor/wall performance, with the most recent being the Snowflake divertor concept [2] and generating Resonant Magnetic Perturbations by the SOL currents [3]. However, the specific approaches discussed here are part of a wider class of innovative divertor ideas that have come from the community in the last several years, and we certainly advocate the need to consider a range of options. Indeed, the most effective solution to the heat-flux problem may well contain features of various ideas. For example, there are the X-divertor (Kotschenreuther et al. [4]) that expands the magnetic flux surface in the vicinity of the near-X-point divertor plate, and the super X-divertor (Valanju et al. [5]) that guides the near-separatrix SOL flux tubes to a larger major radius to increase the surface area available for power deposition. These approaches have the common feature of manipulation of the edge magnetic geometry. Another approach is the use of liquid divertor surfaces that can increase the heat-flux capability by flowing the heated material to a cooling region and eventually out of the machine, and/or by being able to withstand a higher peak heat flux [6]. All of these areas are only emerging concepts that require substantially more analysis and definitive experimental tests, and given the need for a large improvement in this area, we advocate a substantial program to systematically assess the approaches. Because of space limitation here, we present some details of one of the concepts, namely the Snowflake divertor configuration.

Snowflake divertor concept: The Snowflake (SF) divertor [2] exploits a tokamak geometry in which the poloidal magnetic field varies quadratically with distance from the X-point null, Δr. The name stems from the characteristic hexagonal, snowflake-like, shape of the multi-branched separatrix for this exact second-order null. In contrast, the standard X-point configuration has a poloidal field varying linearly with Δr. The different variations mean that a flux expansion is much larger in the vicinity of a null of a snowflake divertor, and one can try to exploit this fact for reducing the divertor heat load. A unique feature here is also that the shear in the magnetic field near the X-point is substantially larger for the SF configuration, which may favorably affect microinstabilities and ELMs. Practical realization appears straightforward; the SF can be obtained using existing poloidal field coils in various present-day devices, and in general can be produced with coils located well outside the vacuum vessel.

The SF configuration increases the flux expansion near the X-point that can be exploited by some increase in the plate wetted area, longer field-line length, and larger volume for impurity radiation. Initial UEDGE simulations comparing the SF with the standard divertor for the same conditions show a reduction in the peak heat flux for the SF of ~1.2-1.6 when comparing cases for the same angle of the total magnetic field to the divertor plate [7]. The increased magnetic shear does influence ELM stability and preliminary analysis indicates that the ELM threshold pressure gradient increases.

Short-term experimental tests: An efficient way to have a first glimpse of the behavior of the SF geometry is by adjusting poloidal-field-coil currents in present-day tokamaks. The snowflake configuration has already been transiently achieved on the DIII-D tokamak. Here, two poloidal field coils are used to generate a pair of X-points in the lower divertor region. The distance between the X-points can be varied using a third coil located approximately midway between the other two. This allows the geometry to be continuous varied from a well-isolated first-order null to a SF-plus [2] configuration to a pure SF. Reversal of the current in the third coil allows access to the SF-minus configuration. While generation of the desired geometry has been demonstrated, deterministic control of the location of the X-points is still needed. Recent intensive work on this has yielded a promising solution. Upon validation of the control algorithm, a dedicated experiment can be performed which would quantitatively explore the physics of the SF configuration relative to the standard first-order X-point configuration and explore the predicted modifications to edge stability and associated effects on pedestal physics and ELMs.

Initial modeling of the “snowflake” divertor configurations for NSTX indicated that a variety of plasma equilibria with the “snowflake-plus” and “snowflake-minus” divertors can be obtained using the existing PF1A, PF1B, and PF2L divertor coils. In order to obtain and assess these configurations, an experiment has been proposed and tentatively scheduled for the FY 2009 run. The SOL and divertor diagnostic set on NSTX is well suited to study the “snowflake” divertor, as the divertor heat and particle flux profiles, ELM stability, and divertor and SOL turbulence characteristics can be assessed simultaneously. It is also worthwhile noting that the planned NSTX upgrades include a new center stack and an additional neutral beam injector. A new lower divertor coil is considered as part of the upgrade, thus expanding the flexibility of NSTX divertor shaping even further.

The SF configuration has also been generated in experiments on TCV tokamak in Lausanne, where the three topological states SF, SF-plus, and SF-minus [2] have been obtained [8].

Theory/simulation short-term studies and needs: Detailed analytic analysis of the SF magnetic topology including flux expansion and magnetic shear is reported in Ref. [2]. Full MHD equilibrium analysis using the Corsica code is presented in Ref. [7], together with UEDGE transport simulations of the peak heat flux as compared to the standard divertor. MHD stability analysis has also begun.

As with the standard divertor configuration, much remains to be done to appropriately analyze the SF configuration. Tasks with existing tools: a broader set of transport simulations is needed that includes classical drifts, multi-charge-state impurities, Monte Carlo neutrals, and near-wall sputtering code (e.g., WBC); turbulence simulations, e.g. BOUT; extended ELM stability analysis with ELITE; and comparison with experiment as available. Tasks for tool development: analytic analysis of the SF plasma impact, e.g., neoclassical drift orbits; gyrokinetic transport simulations; gyrokinetic turbulence simulations.

Strawman Innovative Divertor Thrust outline:

0-5 years: Implement/evaluate performance in present-day experiments, theory/simulations with validation, diagnostic upgrade, model upgrade, international collaboration

3-10 years (overlap): Design/test optimized innovative divertors in tokamaks such as NSTX, DIII-D, C-Mod; continue supporting activity and international collaboration

10+ years: Implement on a next-step experiment, continue supporting activity and international collaboration

Requirements for divertor characterization/qualification:

- Sufficient power to achieve edge pressure gradients to examine ELM stability limits
- Diagnostics to measure PWI and SOL properties, turbulent transport including 2D heat flux; radiative loss, impurity content
- Sufficient parallel heat flux to test limits on radiative dissipation and detachment
- Equilibrium reconstruction to verify snowflake or other geometries
- Particle exhaust control of global particle balance and diagnostics to quantify results
- 2D/4D transport, 3D/5D turbulence, & near-surface transport simulation to validate with data

Requirements for core/edge integration:

- Capability for obtaining high-confinement, high-beta, and steady-state operating modes
- Maintains high power capability with acceptable impurity levels for candidate PFC materials
- Sufficient core diagnostics to quantify: confinement, plasma purity, stability, and etc.

References:

[1] A. Loarte et al., “Chapter 4, Power and Particle Control,” Nucl. Fusion **47** (2007) S203.

[2] D.D. Ryutov et al., Phys. Plasmas **15** (2008) 092501, and earlier SF references therein.

[3] I. Joseph, R.H. Cohen, D.D. Ryutov, submitted to Phys. Plasmas.

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[6] R. Kaita, R. Majeski et al., J. Nucl. Mater. **363-365** (2007) 1231.

[7] M. Umansky et al., submitted to Nucl. Fusion.

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