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European roadmap to the realization of fusion energy: Mission for solution on heat-exhaust systems

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HIGHLIGHTS

- A summary of the main aims of the Mission 2 for a solution on heat-exhaust systems.
- A description of the EUROfusion consortium strategy to address Mission 2.
- A definition of main unresolved issues and challenges in Mission 2.
- Work Breakdown Structure to set up the collaborative efforts to address these challenges.

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ABSTRACT

Horizon 2020 is the largest EU Research and Innovation programme to date. The European fusion research programme for Horizon 2020 is outlined in the "Roadmap to the realization of fusion energy" and published in 2012 [1]. As part of it, the European Fusion Consortium (EUROfusion) has been established and will be responsible for implementing this roadmap through its members. The European fusion roadmap sets out a strategy for a collaboration to achieve the goal of generating fusion electricity by 2050. It is based on a goal-oriented approach with eight different missions including the development of heat-exhaust systems which must be capable of withstanding the large heat and particle fluxes of a fusion power plant (FPP). A summary of the main aims of the mission for a solution on heat-exhaust systems and the EUROfusion consortium strategy to set up an efficient Work Breakdown Structure and the collaborative efforts to address these challenges will be presented.

1. Introduction

The main challenge in realizing a fusion power plant (FPP) is the adequate control of heat exhaust which is the main aim of Mission 2 of the European Fusion Roadmap [1]. Already demanding for ITER, the problem is amplified for a FPP where the assumed linear dimensions are ≈50% larger and the fusion power output at least 3 times higher (see for example [2]). The power crossing the magnetic separatrix is channelled along the magnetic field lines to the divertor where it is exhausted on actively cooled divertor targets. The heat flows in a narrow radial layer (SOL) of width λ_q (~1 mm at the midplane for ITER) which scales only weakly with machine size [3]. This means that the loaded divertor area scales approximately with the major radius R of a device making P/R (P being the exhaust power) a crucial parameter when extrapolating to larger devices. On the current ITER exhaust assumptions ≈40% of the ≈150 MW ITER heating power (fusion alpha particle power and auxiliary heating) is radiated inside the magnetic separatrix and 60% (≈90 MW) will flow into the SOL with ≈60 MW (i.e. 2/3) towards the divertor outer target. In order to minimize the heat load, the divertor targets are inclined at a shallow angle to the magnetic field lines and are located in a region near the

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separatrix X-point with significant magnetic flux expansion, increasing the divertor target area in ITER to $\approx 2\,m^2$. In the attached divertor regime, when almost all the heat entering the SOL ultimately ends up on the divertor target, this area increase lowers the power load to $\approx 30\,MW/m^2$. However, despite significant progress during the last two decades, such heat loads exceed present technological capabilities–prototypes of water-cooled copper alloys with either carbon or tungsten armour tested under cyclic power loads, have only demonstrated $\leq 20\,MW/m^2$. These values are close to the intrinsic limits of the thermo-mechanical properties of the small number of materials suited for application in the fusion environment. In realistic fusion plant conditions these properties will be significantly degraded by neutron irradiation at the level of a few displacements per atom (dpas). Transients, tile misalignments and considerations of other realistic design tolerances will further reduce the power handling limits for reliable divertor target to $\approx 10\,MW/m^2$, in the case of water-cooled, and to even lower values in the case of *He*-cooled components.

Solutions for the heat exhaust in DEMO/FPP are presently being explored along three main lines:

- Baseline divertor solution – a combination of radiative cooling and detachment. In such conditions a significant temperature gradient can be established and volume recombination of the plasma can take place, hence reducing the ion fluxes to the target;
- Innovative magnetic divertor configurations to achieve higher flux expansion, spreading the heat over a larger area or to achieve longer divertor connection lengths and larger divertor radiated power;
- Advanced plasma-facing components (PFCs) (e.g. liquid metals) that could exhaust higher heat loads.

Low SOL temperature, associated with detached divertor conditions, also reduces the erosion of the divertor armour, the main factor defining its working life. The baseline divertor approach will be tested by ITER, thus providing an assessment of its adequacy for DEMO where limits are likely to be even more demanding on both SOL temperature and radial extend of detachment mainly due to much stricter erosion requirements.

Nevertheless, the risk of non-applicability of this solution for the high-confinement operation in the DEMO fusion reactor remains significant, potentially delaying the realization of fusion energy and prompting the search for alternative, risk mitigation solutions. Besides intensified research on divertor detachment in existing devices, the European fusion community discusses the plans for a Divertor Test Tokamak (DTT) which potentially could yield answers to the above mentioned questions in parallel to ITER operation. The strategy also includes a technological study of the feasibility and performance of water-cooled divertor targets concepts. It extends the ITER design and technology to DEMO relevant conditions (e.g., higher coolant temperatures and pressures and higher n-dose) and must be applicable to any potential divertor solid target concept. Finally, divertor pumping must be sufficient to exhaust the neutralized gas, most notably He ash, as well as limit the eroded impurities entering the main plasma. Any of the divertor acceptable solutions must satisfy these requirements together with those of heat exhaust.

2. Baseline strategy

2.1. Detachment and radiating

Detached divertor conditions have been obtained in several tokamaks and will be pursued by studies based on existing, especially all metal PFC, divertor devices (see for example [4]). The plasma detachment is normally characterized by a strong pressure gradient along magnetic field lines in front of the target and a reduced ion flux to the target in such a way that the plasma temperature close to the target decreases below several eV allowing volume recombination of the hydrogen isotopes. The required high collisionality can be achieved by: reducing the power flowing to the SOL (P_{SOL}); increasing the SOL density; and producing magnetic configurations with a large connection length between the midplane and the divertor target. Decreased P_{SOL} can be achieved by radiating a large amount of power from the plasma edge (using extrinsic impurities). However, H-mode operation requires a minimum power to be conducted through the pedestal (P_{thr}) which will limit main chamber radiation specifically for ITER. In addition, the tungsten sputtering limit, which is largely determined by the impurity concentration in the divertor plasma, must also be evaluated [5]. Furthermore, detached conditions will have to be carefully controlled to ensure safe operation, requiring robust sensors, algorithms and actuators. ITER will play the ultimate role in proving the applicability of the “conventional” power exhaust scenario for DEMO [2] (Pulsed DEMO1: $P_{fus} \sim 1.8\text{–}2\,GW$, $f_{rad,core} \sim 65\%$) but it can provide this information only after the successful achievement of long pulse high fusion gain ($Q \approx 10$) operation around 2030. In preparation of a safe ITER start-up and to provide further input for a decision on a DTT, the behaviour of detachment at high levels of heating power and radiation must be investigated during the first half of Horizon 2020. Specifically, the control of detachment, its compatibility with ELM mitigation and the behaviour close to the H–L threshold must be documented. Although divertor detachment has been achieved on present day tokamaks, its behaviour cannot be described by the existing numerical codes in a predictive fashion and must be supported by a strong model validation and code development efforts.

2.2. Challenges of erosion of the PFC

The PFC in the main chamber wall will receive power from radiation and particles and undergo erosion. For ITER, *Be* melting and excessive erosion can hamper operation whereas for DEMO the choice of the PFC material and the cooling technology depends critically on the particle spectrum and the total absorbed power. Therefore all solutions envisaged for the power exhaust in the divertor must also treat the main chamber issues in a consistent way. In addition to the power handling requirements of steady state and transient power loads (although not as demanding as in divertor targets), the erosion of the PFC has to be minimized in order to maximize the availability of the device and to reduce the deleterious effects of tritium co-deposition and dust production. To optimize the material choice specifically in the main chamber, the temperature and flux of plasma filaments must be quantified (including impurities). In parallel, improved PFC materials consistent with the engineering requirements must be developed. The specific plasma wall interaction (PWI) of seeding impurities with the respective armour material as well as the effect of material mixing will have to be determined. Since all conventional solutions foresee metallic PFCs, the effect of accidental melting by the plasma and on the performance of the component must be clarified

3. Risk mitigation strategy

A significant risk remains that high-confinement regimes of operation are incompatible with the larger core radiation fraction required in DEMO. Therefore, an investment in assessment of the adequacy for DEMO and proof-of-principle tests of innovative geometries as well as the use of liquid metals (LM) is required.