An Integrated View on Plasma Power Exhaust and In-vessel Components

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20% of the energy produced by the thermonuclear DT reaction will be involved in interaction inside the plasma resulting in ion and electron kinetic energy and electromagnetic waves. In addition control processes are expected to be instigated in the plasma core contributing to its heating. For stationary conditions this power is transferred to the surrounding plasma facing components (PFCs) contributing to their thermal loading. A reliable prevision of the distribution of this energy (average and local peak) is background for the engineering design of the PFCs (i.e. blankets and divertors). In particular, for an ITER-like plasma exhaust configuration, the heat flux \( \dot{q}_\text{wall} \) to the divertor target plates constitutes a major challenge for the technology (design and materials) with ion fluxes exceeding 10 MWm\(^{-2}\). The loads on the Blanket FW constitute a not minor challenge, due to the complexity of this component that has to achieve multiple functions (i.e. tritium breeding, heat removal, high temperature coolant heating for electricity generation, shielding) already peak fluxes exceeding \( 0.5 \text{ MWm}^{-2} \) can constitute issue for the large affected FW surface (>1000 m\(^2\)). The situation can be worst during transients. During plasma “start up” and “shut down” the blanket system has to act as plasma limiter and therefore its plasma-facing part should be designed to withstand the plasma heat loads generated during these transition phases. In addition, instabilities of plasma will produce off-normal transients in which local values of heat flux will abundantly exceed the stationary values.

Furthermore, the design of the PFCs has to cope with a damage of the surface materials caused by the direct impact of ions and neutrons coming from the SOL. To optimize the consequences of erosion in normal operation and of off-normal events, moderate shaping of the first wall is envisaged with the protruding structures (Enhanced Heat Flux, EHF, elements) acting as sacrificial layers which are replaceable with moderate effort. Of course aggravating conditions will impact the design of EHF PFCs. In particular the materials will be exposed at high neutron fluence of high energetic neutrons. The damages induced in the materials (including He production) will concur and act in synergic effect to the “erosion” to limit the component lifetime. Furthermore, electromagnetic transients that will produce thermal quench, mechanical impulses and runway electron interaction will contribute to the loading of these components.

Radiative loss power \( L_z \) as a function of electron temperature \( T_e \) for different species. Data are calculated from the ADAS database for an electron density of \( 10^{20} \text{ m}^{-3} \) assuming Corno equilibrium. The radiated power density is obtained by multiplication with the electron and the impurity densities.

Effective sputtering yields for tungsten as a function of temperature for different species, assuming \( T_e=1 \). The black curve gives the total yield for the species as indicated in the figure. The right picture shows the corresponding tungsten influx for the condition of the left graph. The assumed DEMO divertor erosion limit is indicated, showing that for plasma temperatures below 4 eV are acceptable in front of the divertor target.

DEMO, but likely also a first generation of FPP will cope with a plasma exhaust power configuration ITER-like. Hence, the increase of the power exhaust density in a true reactor will require efficient dissipation method to reduce the flux to the divertor target at levels compatible to a near-term technology. For a 3 GW fusion power reactor with relative moderate neutron wall load (\( \leq 5.5 \text{ MWm}^{-2} \)) a core dissipation of circa 60% and a divertor power dissipation of more than 30% could reduce the heat flux to the solid target to about 10 MWm\(^{-2}\). This configuration seems viable with reasonable extrapolation from the today physics. This configurations seams also compatible to ensure temperature at the edge lower than 3 eV, with modest expected “erosion” of the W divertor armour. The same configuration would load the blanket first wall with a radiative power of 0.5 MWm\(^{-2}\) average (1 peak) heat flux with weak SOL “erosion”. In any case it will be required also limiter-like surface (EHF) elements to cope with SOL interaction during transients of at least 5 MWm\(^{-2}\).

These operational conditions are at the limit of the present technology. In particular divertor plates able to remove 10 MWm\(^{-2}\) have been produced (ITER water cooled) or are under development (KIT helium cooled divertor) but their lifetime expectation due to the behaviour of the materials under beyond-ITER neutron conditions (1-3 MWm\(^{-2}\)) are not encouraging. A further reduction of the target flux (under 5 MWm\(^{-2}\)) could open other design possibility besides the use of more resistant materials, but at this point the required enhancement of the power dissipations seems to arrive at physical limits. Blanket concepts have been developed in EU for modest neutron wall load and mostly for only 0.5 MWm\(^{-2}\) surface heating. This can be insufficient for large part of the FW where higher heat fluxes are likely to be expected. The extension of the heat removal performances up to the more comfortable 1 MWm\(^{-2}\) would be a serious challenge for several of these blanket concepts. Concepts of EHF elements should be also addressed in the reactor design and integration.