Karlsruhe Institute of Technology

Karlsruhe Institute for Technology, Campus Nord
Institute for Neutron Physics and Reactor Technology
Hermann-von-Helmholtz-Platz 1
D-76344 Eggenstein-Leopoldshafen
Germany

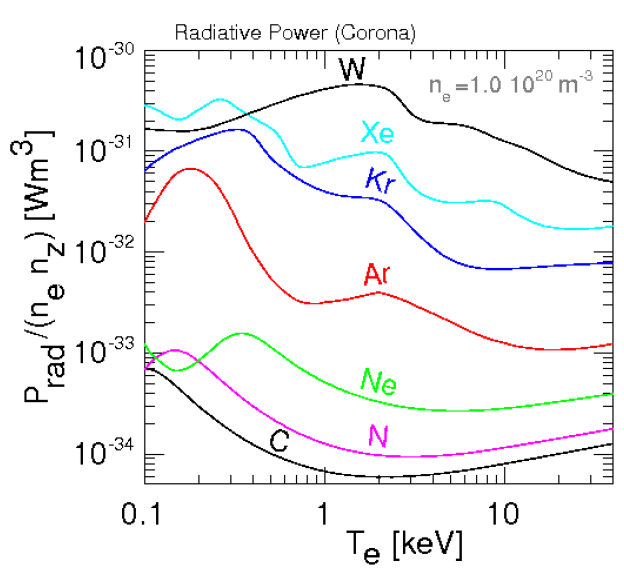
An Integrated View on Plasma Power Exhaust and In-vessel Components

Lorenzo V. Boccaccini[a], Arne Kallenbach[b]

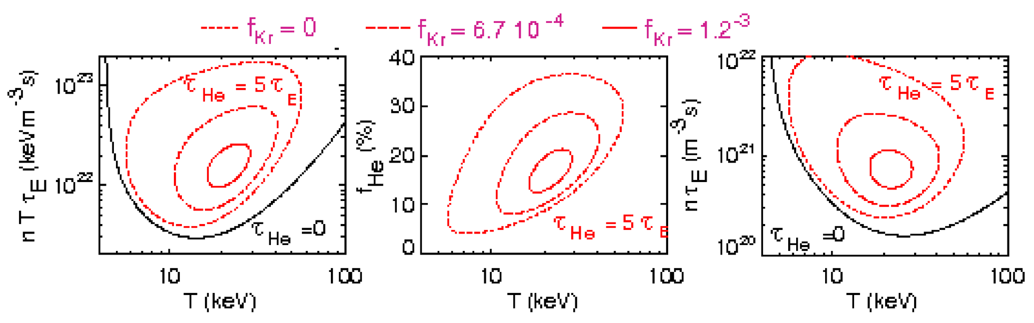
[a] Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany
[b] Max-Planck-Institut für Plasmaphysik, Garching, Germany

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 power will be injected 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 "diverted" to the divertor target plates constitutes a major challenge for the technology (design and materials) with ion fluxes exceeding 10 MW/m². 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 MW/m² can constitute issue for the large affected FW surface (>1000 m²). 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

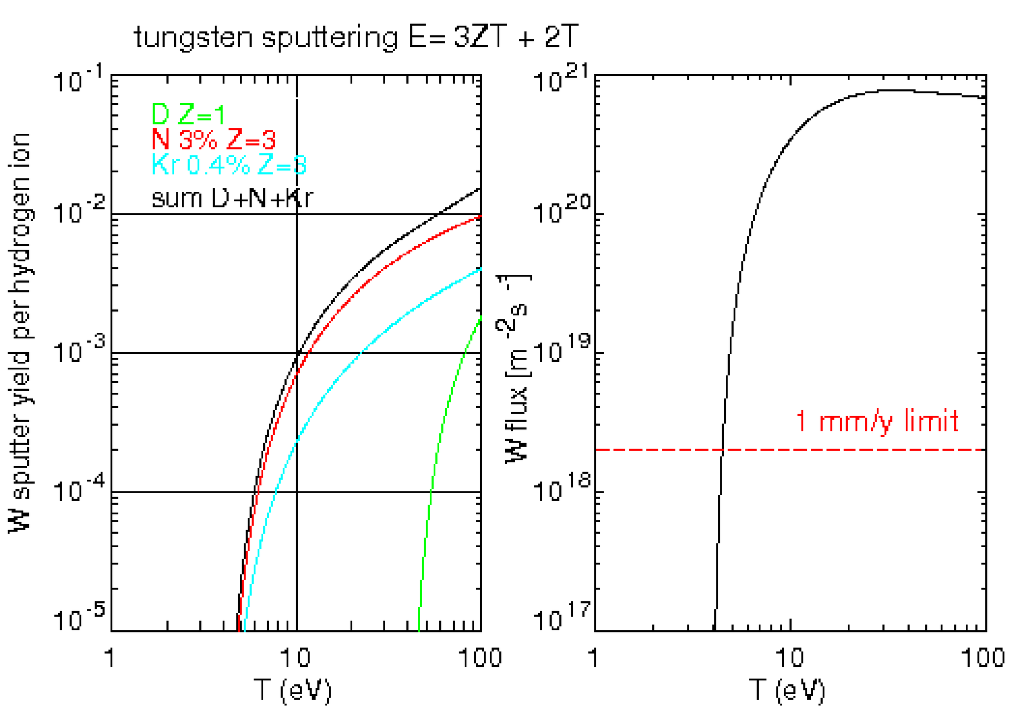
Furthermore, the design of PFCs has to cope with a damage of the surface materials caused by the direct impact of ions and neutrals coming from the SOL. To optimize the consequences of erosion in normal operation and of off-normal events, moderate shaping of the first wall is envisaged with the protruding structures (Enhanced Heat Flux, EHF, elements in ITER) 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.



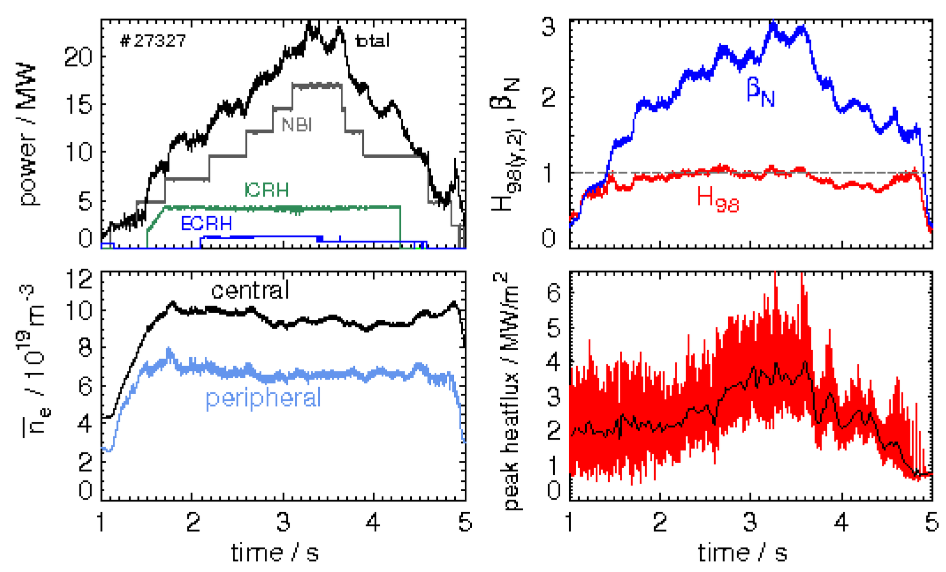
Radiative loss power L_z as a function of electron temperature for different species [Kallenbach]. Data are calculated from the ADAS database for an electron density of 10^{20} m^{-3} assuming Corona equilibrium. The radiated power density is obtained by multiplication with the electron and the impurity



0D-burn diagram for a DEMO case with strong Kr seeding. The He concentration is self-consistently calculated from the fusion rate assuming the He confinement time τ_{He} = 0 (black) or 5 τ_E (red). With a Kr concentration of 1.2 10^{-3}, the minimum $n\tau T$ of 9.5 10^{21} m^{-3} s keV is reached at T=18.9 keV. Assuming a burning volume of 500 m3, and an electron density of 1.5 10^{20} m^{-3}, an α-heating power of 320 MW is obtained, with τ_E = 3.4 s and a radiated power of 130 MW. The curves show conditions with alpha heating power equal to loss power.



Effective sputtering yields for tungsten as a function of temperature for different species, assuming Te=Ti. The black curve gives the total yield for the species mix 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 only plasma temperatures below 4 eV are acceptable in front of the divertor target.



ASDEX Upgrade discharge with maximum heating power and divertor power dissipation by nitrogen seeding to stay below the critical value of 5 MW/m².

Model:	A [Sardain]	AB* [LiPuma06]	B [Hermsmeyer]	C [Norajitra03]	D [Giancarli03]
• Fusion Power	5.0 GW	4.24 GW	3.6 GW	3.45 GW	2.5 GW
Blanket	"WCLL"	"HCLL"	"HCPB"	"DCLL"	"SCLL"
• Struc. Material	EUROFER	EUROFER	EUROFER	EUROFER	SiC$_f$/SiC
• Breeder	PbLi$_{eu}$	PbLi$_{eu}$	Li_4SiO_4	PbLi$_{eu}$	PbLi$_{eu}$
• Multiplier	PbLi$_{eu}$	PbLi$_{eu}$	Be	PbLi$_{eu}$	PbLi$_{eu}$
• Coolant (inlet-outlet)	Water (285-325°C)	Helium (300-500°C)	Helium (300-500°C)	Helium (300-460°C) PbLi$_{eu}$ (460-700°C)	PbLi$_{eu}$ (600-1100°C)
• Neutron wall load, average (peak)	2.2 MW/m² (2.5 MW/m²)	1.84 MW/m²	2.0 MW/m² (2.4 MW/m²)	2.2 MW/m²	2.6 W/m² (3.4 MW/m²)
• Surface heating (max)	0.57 MW/m²	0.5 MW/m²	0.5 MW/m²	0.59 MW/m²	0.5 MW/m²
Divertor					
• Material (armour – structure)	W – CuCrZr	W – W/Steel	W – W/Steel	W – W/Steel	W – SiC$_f$/SiC
• Coolant (inlet-outlet)	Water (140-200°C)	Helium (540-710°C)	Helium (540-710°C)	Helium (540-710°C)	PbLi$_{eu}$ (600-1000°C)
• Heat flux at the strike point	15 MW/m²	10 MW/m²	10 MW/m²	10 MW/m²	5 MW/m²

*The Model AB was added after the official conclusion of the EU PPCS

Acronym:	WC: Water Cooled HC: Helium Cooled DC: Dual Cooled SC: Self Cooled	PB: Pebble Bed LL: Lithium Lead eu: eutectic

Blanket and divertor selection for the EU PPCS Models.

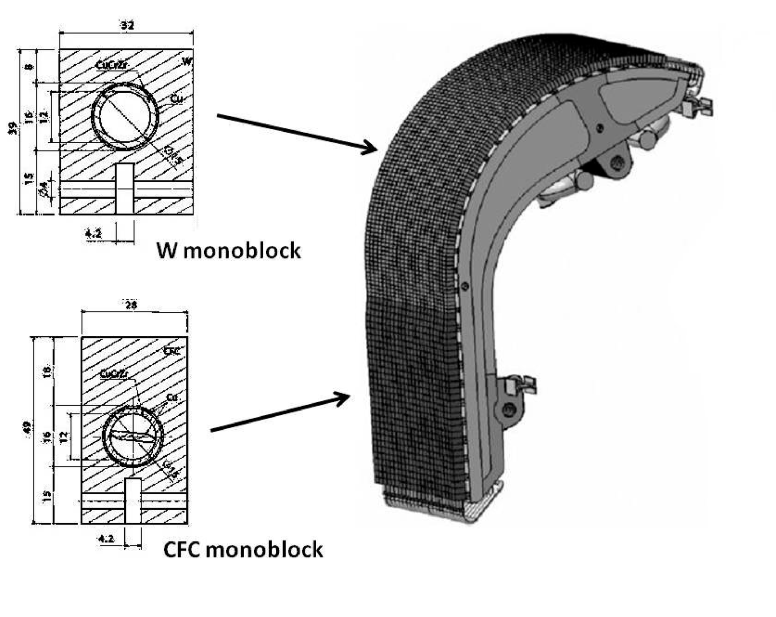


Illustration of outer vertical target of the ITER divertor. In ITER the divertor plates are designed to withstand stationary heat fluxes at the strike point of 10 MW/m² with possible flux increase up to 20 MW/m² during transient events of 10 s. The use of low temperature water cooling and excellent heat sink materials like the CuCrZr-IG allow to achieving the required thermo-mechanic performances .

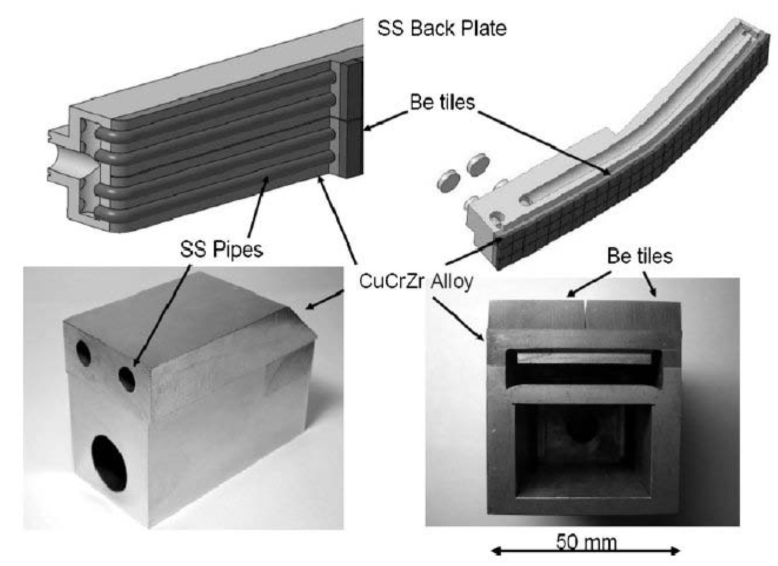
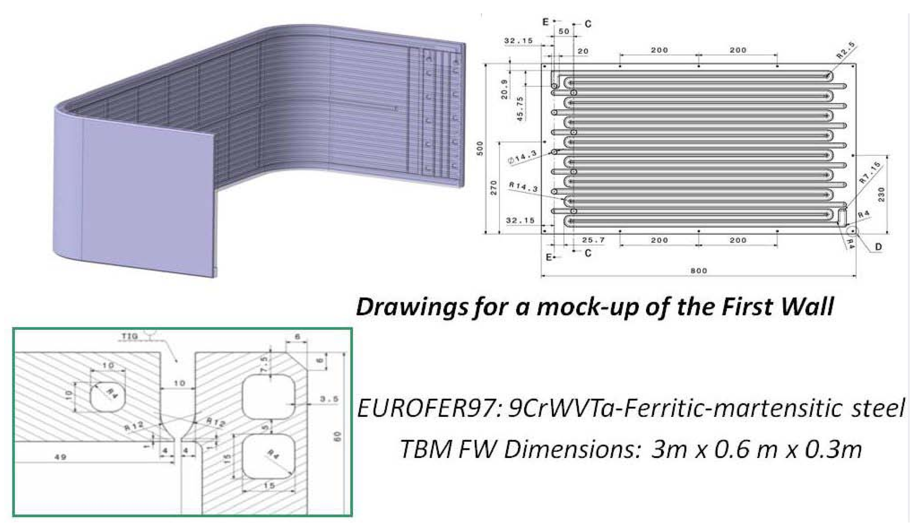


Illustration of the basic FW panel structure and fingers. Left: normal heat flux fingers (concept with steel cooling pipes). Right: enhanced heat flux fingers. The current design assumes a heat flux value of 1-2 MW/m² on almost all the blanket surface with limited regions (EHF elements) in which the value can reach 5 MW/m².

Manufacturing of the 9-finger module for the Helium cooled divertor developed in KIT. In 2010 a new series of tests conducted at the Electron Beam facility in Efremov achieved a first breakthrough in the qualification programme for such as divertor target: one finger was able to survive 1000 cycles at 10 MW/m² under high temperature helium cooling.



Drawing for the manufacturing of a FW to be tested in the TBM Programme for the EU TBMs (KIT Design). Theconcepts are designed for a relatively modest neutron wall load (max 2.4 MW/m²). Assuming a target withstand of the structural material (EUROFER) up to 150 dpa neutron damage, a blanket lifetime of about 5 FPY was considered for the component. About the interaction to the power exhaust, the design of the FW of these components was specified for only 0.5 MW/m².

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 vertical target at levels compatible to a near term technology. For a 3 GW fusion power reactor with relative moderate neutron wall load (<2.5MW/m²) 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 MW/m². 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 MW/m² 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 MW/m².

These operational conditions are at the limit of the present technology. In particular divertor plates able to remove 10 MW/m² 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 to 3 MWa/m²) are not encouraging. A further reduction of the target flux (under 5 MW/m²) could open other design possibility with the use of more neutron 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 MW/m² 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 MW/m² 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.