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

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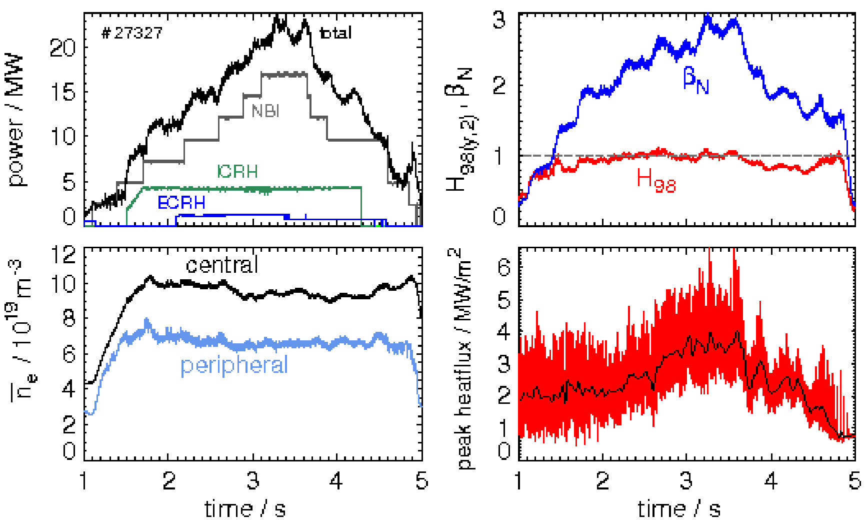

1. INTRODUCTION

20% of the energy produced by the thermonuclear DT reaction, the part not related to the neutron source, will be involved in interaction inside the plasma resulting in ion and electron kinetic energy and electromagnetic waves. In addition control power (burn control or current profile control) will be injected in the plasma core contributing to its heating. For stationary conditions, the sum of α-power and control power is transferred to the surrounding plasma facing components (PFCs) either via radiation or particle impact and contributes 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/m^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 MW/m^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 PFCs has to cope with a damage of the surface materials caused by the direct impact of ions and neutrals coming from the SOL. This "erosion" consists of sputtering during normal operation and sputtering or melting during of-normal events like unmitigated ELMs, plasma position excursions or the impact of runaway electrons during disruptions. 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. If EHF elements e.g. based on tungsten are used, the remote wall areas will experience small particle loads and weak erosion. Therefore, steel could be an acceptable plasma facing material, removing just radiative heat flux.

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.

This paper will address mainly the PFC heat load to be expected in a future reactor (DEMO or Fusion Power Plant, FPP) and the erosion caused by the direct interaction with the SOL. It will discuss present previsions from the physic and possible improvements that can be expected in the time scale of a beyond ITER fusion reactor. The paper will also present a review of the present technological development for blanket and divertor, focusing the assumptions considered in the design and the present achievements. The view of the paper is mostly referred to an EU point of view if not otherwise mentioned; some works in US and JA have been revised also, and explicitly mentioned when referred to for comparison.

2. PHYSICS BACKGROUND

Albeit its size is expected to be not much larger compared to ITER (up to 8 m in some proposed lay-outs), the plasma power flux in DEMO is expected to be much higher (~650 MW for a ~3 GW fusion power machine with 50 MW additional control power, corresponding to about a factor 5). This means that the fluxes that would load the PFC will be at least 2-3.5 times higher than in ITER. On the other hand, due to much stronger neutron induced material degradation and the corresponding material temperature restrictions, the engineering design of a solid target DEMO divertor could not be specified to higher heat load capabilities compared to the ITER divertor.

As a consequence, a major part of the DEMO power flux has to be radiated in the confined plasma region. This appears feasible since the power flux by far exceeds the H-L power limit in DEMO, different to ITER. Medium-Z noble gases are considered as radiators due to their high recycling coefficient and chemically inert behaviour. Confined plasma radiation has to be optimized and a mix of species may be used. Here, the lower-Z species generally has the more favourable spatial radiation distribution, with its maximum in the outer plasma region, but causes higher fuel dilution. The remaining heat flux crossing the separatrix has to be dissipated by the divertor, using a combination of low-Z seed radiation losses and plasma detachment.

2.1 Core plasma radiative cooling

Seed impurities will be injected by gas valves, using a feedback on the radiated power, and removed by the pumping system. Figure 1 shows core radiative loss parameters for different species of interest, calculated from the ADAS database in Corona equilibrium for an electron density of 10^{20} m^{-3}, which is applicable for the plasma core. For krypton, we read a value Lz= 10^{-32} Wm^3 for temperatures at the pedestal and further inward. The radiation loss function Lz predicts a total radiated power of 100 MW for 1000 m^3 radiating volume, electron density ne= 10^{20} m^{-3} and a Kr concentration cz= 10^{-3}. The corresponding increase of the plasma Z_{eff}, dZ_{eff}= c_z Z(Z-1) is 1.2 if almost full ionisation of the Kr atom is assumed throughout the core plasma. If the power dissipation capability of the DEMO divertor cannot be improved above the current standard vertical target design, about 400 MW core radiation due to seed impurities will be required. Figure 2 shows zero-dimensional calculations of the plasma power balance [Reiter] under such conditions as an example. He production is taken into account according to the α birth rate, He loss is calculated assuming an effective He particle confinement time of 5 τ_E. A realistic solution with a minimum $n\tau_E T$ of 9.5 10^{21} m^{-3} s keV is obtained at T=18.9 keV. Assuming an electron density of 1.5 10^{20} m^{-3}, τ_E= 3.4 s follows and with a plasma volume of 500 m^3, 320 MW α-heating power would result. The (maximum) total radiated power here is 130 MW, which is not sufficient for power exhaust alone. However, this 0-D model is very simple, and a surrounding radiating plasma volume without significant fusion processes should be feasible, corresponding to the region inside the pedestal in reality. The numbers for parameters here should just sketch the global operational range. Self-consistent core and divertor simulations for DEMO parameters with impurity seeding have been presented in [Pacher], demonstrating acceptable peak power loads.

The most critical issue with the proposed strong core radiative cooling is whether a sufficient energy confinement time can be maintained. To answer this question, further work is needed on pedestal parameter scaling and first principle transport calculations inside the pedestal including the radiative loss channel. The pedestal parameter scaling has to take into account the fuelling method as well as the applied ELM mitigation technique. In case that such strong radiative losses are not compliant with the required energy confinement, more power will reach the divertor zone requiring to develop improved divertor concepts able to dissipate higher heat flux.

For the 3 GW fusion power machine anticipated in the beginning, about 450-500 MW main chamber plasma radiation (seed radiation plus D+T+He bremsstrahlung and cyclotron losses) have to be distributed over about 1000 m^2 surface area, resulting in average heat loads of the main chamber wall of up to 0.5 MW/m^2, and 1 MW/m^2 has to be assumed due to spatial inhomogeneities and peaking. There is no room for substantial plasma energy fluxes to wall components. If the power handling of these elements is already at their limits, sufficient space between wall and separatrix has to be given to allow for a radial decay of the scrape-off layer power flux due to parallel conduction into the divertor.

2.2 Divertor power dissipation

Assuming a current standard vertical target divertor geometry operating under partially detached conditions, dissipation of 150-200 MW in the divertor should be feasible, assuming moderate conceptual improvements against the ITER divertor like, e.g., a longer outer divertor leg. A critical element in DEMO divertor operation is the limited allowed material erosion rate of about 1 mm per operation year. This turns out to be a very challenging issue if compared to the campaign-averaged net tungsten erosion in the outer divertor of ASDEX Upgrade in 2007, where on average a factor 5 higher erosion rate was measured [Dux]. Therefore, only scenarios with very small erosion rates need to be considered, and tungsten or tungsten-based composites or alloys are the candidates as plasma facing materials.

Due to the technological difficulties in the design of the divertor target plates, there is a tendency to try to have lower heat flux (time average, but peak). Coming under 5 MW/m^2 will offer new perspectives for feasible design. The plasma wall interaction in the divertor has to take place under low temperature conditions. The strike points need to be detached, attached conditions are allowed further up along the target where the impinging ion flux density is low. Indeed, dedicated discharges with strong nitrogen seeding in ASDEX Upgrade remained well below the limit of 1 mm / year (or 2 10^{18} W/ m^2s).

Figure 3a shows the effective tungsten sputtering yields w.r.t. the hydrogen flux (that means the yields shown are the true sputtering yields multiplied by the assumed impurity concentration/flux fraction). Shown are the yield for deuterium (which is negligible at low temperatures), nitrogen and krypton, as well as the total effective yield per impinging hydrogen ion.

Figure 3b shows the tungsten sputtering rate as a function of plasma temperature in front of the target. The impinging ion flux has been calculated from Te under the assumption of a heat flux of Γ_{heat} = 5 MW/m^2 using the relation Γ_{heat}= $(7T_e + E_{rec})$ e Γ_i with the surface recombination energy per electron-ion pair E_{rec} = 14 eV. Prompt re-deposition has not been taken into account here.

The tungsten erosion is no problem if Te can be kept below 3 eV, but is completely unacceptable for Te > 8 eV. The requirement to limit the surface heat flux below about 5 MW/m^2 should go along with low values of Te since momentum loss processes are inevitable, and finally low values of Te are required to achieve momentum loss. The current chain of processes is strong radiation at the divertor entrance leading to higher density and lower temperature, hydrogen charge exchange redcuing the ion flux and temperature and finally radiative and 3-body recombination, which become effective around T_e= 1 eV. But extrapolations from current devices are unclear, and 2D edge modelling is not ready for precise predictions.

Improvements of the divertor concept are in particular welcome to reduce the required high core radiation. The improvements should in particular aim at reduced plasma temperature in front of the target in order to limit surface erosion. Geometrical power spreading alone is not sufficient; improvements should rather aim at promoting detachment of the strike point area.

3. ENGINEERING ACHIEVEMENTS

The ITER configuration for the exhaust system that will be tested in the first thermonuclear experiment can be considered the reference concept also for the first generations of FPPs, starting from DEMO.

In ITER the divertor plates are designed to withstand stationary heat fluxes at the strike point of 10 MW/m^2 with possible flux increase up to 20 MW/m^2 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 [Merola]. Large uncertainties are in the estimation of withstand of the selected materials under the ITER D-T environment that can limit the predicted lifetime of this component; namely neutron damages and SOL erosions. Copper and copper alloy are very sensitive to neutron irradiation that at the divertor location can reach a cumulative neutron fluence of 0.3 MWa/m^2 at EOL of the plant. To cope with the SOL interaction, armour materials shall protect the target plates that have to withstand sputtering and thermal transients maintaining acceptable core plasma impurities. W and CFC are used in the ITER design (see Figure 4) ensuring a sacrificial layer of about 4 mm W (or 11 mm CFC). It is difficult to predict the lifetime of this component in ITER; it will probably limited by off-normal events or insufficient detachment.

Also the FW of the ITER shielding blanket interacts with the plasma power exhaust. The current design [Merola] assumes a heat flux value of 1-2 MW/m^2 on almost all the blanket surface with limited regions (EHF elements) in which the value can reach 5 MW/m^2. This is due to the limiter function that the blanket assumes during start up and shut-down. The design proposed in ITER (again water cooling and Cu-alloy tubes) can provide a suitable thermo-mechanical design for this component. The issue of FW protection from the SOL particles is addressed with a Be armour of max 5 mm thickness [Hunt]. Figure 5 taken from [Merola] illustrates the ITER concept. Again a lifetime prevision of this component according to today's plasma physics expectations is quite difficult.

The extrapolation of the ITER PFC design to true reactor conditions have to cope with most severe requirements, e.g. with neutron fluences that can be of two orders of magnitude higher than in ITER. In fact, also assuming a relative conservative design of a first generation of FPP (a possible EU DEMO lay-out), fluence at the first wall at EOL will reach at least 10-15 MWa/m^2; a mature reactor generation would reach fluences beyond 70 MWa/m^2. Hence, the questions of the lifetime of divertor and blanket, and of a viable replacement strategy are key elements for providing attractive FPP concepts.

In EU a comprehensive study on FPPs was conducted in 1999-2004 in the frame work of the Power Plant Conceptual Study (PPCS). In this study [Maisonnier], five configurations (models) for FPPs were analysed (see Table I) under different assumption of component/system selection. Underlying assumption for these models was a plant lifetime of at least 40 FPY with an assumed availability of ~75%. The plasma loading was relatively moderate (in comparison to analogous US and JA studies) with neutron wall load around 2.5 MW/m^2. The divertor heat flux was selected around 10 MW/m^2.

3.1 Divertor development for a FPP

One of the conclusions of the PPCS study was the necessity to explore solutions for a divertor able to withstand a heat flux of at least 10 MW/m^2 for a lifetime corresponding to of 3 MWa/m^2 of neutron fluence on the targets. In addition, the divertor plate should withstand the "erosion" of the particle flux; a 3-mm-of sacrificial layer on the W tiles was added to the specifications in the opinion that could be able to ensure at least 2.5 FPY of lifetime expectation that is congruent with the assumed neutron fluence at EOL of the component.

Only the water cooled divertor with copper tubes as coolant pressure containment (a version derived from the ITER W-monoblock design) associated with model A was considered capable to operate beyond 10 MW/m^2 of heat flux on the target plates [LiPuma-02]. The W-monoblock concept is the reference PFC concept for the upper part of the vertical divertor targets in ITER. Recent results of test of this concept demonstrated the capability to withstand fluxes of 10 MW/m^2 for 3000 cycles and even 20 MW/m^2 for 1000 cycle in out-of-pile conditions [Visca]. For this divertor concept the main concern for the use in a FPP was the expected lifetime under neutron irradiation; copper presents several issues under irradiation including loss of thermal conductivity, hardening vs. thermal softening, loss of ductility, transmutation and swelling. A possible range of application should be limited only at lower temperatures (under 250°C); in this condition the relatively low operational temperature of water contributes to make the water cooled concept not particularly attractive for a FPP in which the possibility to achieve high efficiency of the power conversion system is considered one of the keys to reach low electricity costs [Ward].

The use of water cooling in in-vessel components was also criticised for issues of chemical compatibility with almost all the breeding materials (in particular with Be) and insufficient thermal windows in comparison to the Ferritic Martensitic (FM) steel (e.g. EUROFER or F82H) that should be used as structural materials (too low operational temperatures to ensure sufficient margin for DBTT increase under irradiation). Few ideas for alternative design have been explored since the conclusion of PPCS on the field of water cooled concepts. Attempt was done to substitute copper with a FM steel (that could ensure a better withstand to neutron damage); the drawback is a strong decrease of heat removal performances as steel has a thermal conductivity of only ~1/10 of copper. The limit of 10 MW/m^2 seems hard to be reached with these concepts. Some ideas to improve the heat removal capability in this configuration have been presented in [Giancarli05] but the proposed thermal barrier is remained up to now at a initial conceptual stage. The issue if the irradiation temperature is adequate for a sufficient DBTT margin is also open. Practically no EU R&D has been done in the last 10 years on a water concept for the FPP, apart the divertor design of ITER.

In Japan water cooled divertors (W-monoblock) are reference concept for a DEMO reactor [Tobita]; copper tubes are, however, substituted by the FM steel F82H to avoid issues under irradiations; R&D is ongoing to test heat transfer capability on mock-ups. [Suzuki] refers of mock-ups able to withstand heat fluxes of 5 MW/m^2 for 650 cycles whereas the mock-ups survived more than 10000 cycles at 3 MW/m^2.

Considering the above mentioned issues connected to the water cooled divertor and the fact that 3 of the 5 PPCS models could show attractive performances only with the use of an Helium cooled divertor, interest arise at the end of the PPCS to explore if the conceptual design proposed in these studies could be manufactured with a relatively modest extrapolation of the present technology. An ambitious R&D programme for a 10 MW/m^2 helium cooled divertor has been conducted (under EFDA support) in KIT; the progress are documented in regularly issued publications during the last 10 years [Norajitra11]. The concept use finger elements that are cooled almost in parallel by helium jet flow; W-alloy is use in small caps as structural material to contain the high pressure coolant. 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/m^2 under high temperature helium cooling. From this result further manufacturing improvements are planned up to the production and testing in the next years of a 9-finger module that is the lower unit for the assembly of a divertor plate (see Figure 6). Also for this concept the behavior under irradiation is unknown; the helium cooling in the range of 600-700°C seems to low to ensure sufficient margin for DBTT increase in W under irradiation.

Helium cooled concepts are followed also in the ARIES-CS programme [Raffray]; the divertor concept proposed in this study aims to improve the weak points of the EU design, e.g. the range of temperatures associates to brittle behaviour of W under irradiation and manufacturing issues connected to the reliability of the concept that cope on a large amount of small structure (1.5 cm is the typical diameter of a finger) that have to be assembled together to form the surface of the divertor target.

3.2 Blanket FW development

The most promising candidates for a EU DEMO/FPP are the blanket concepts included in the ITER Test Blanket Programme, namely the Helium Cooled Lithium Lead (HCLL) and Pebble Bed (HCPB), that were considered in Model AB [LiPuma06] and B [Hermsmeyer] of the PPCS, respectively. Both concepts are designed for a relatively modest neutron wall load (max 2.4 MW/m2). This allows a design that fulfil almost all the requirements in term of tritium production, mechanical design and temperature range of materials, and acceptable efficiency of the power generation system. 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/m^2. It was not excluded that limited portion of the reactor FW could have been loaded by peak of flux (up to 1-2 MW/m^2) in limiter-like function (see Figure 5). However, the technologic realisation of this local high flux components (> 0.5 MW/m^2) has been never addressed in detail (at least in EU) and neither their integration in a reactor. Temperature limits in EUROFER (practically not higher than 550°C) and modest cooling performance of the Helium coolant make the enhancement of heat removal capability of their FW questionable. Maybe the introduction of an ODS (Oxide Dispersion Strengthened) version of EUROFER steel plating the FW and the use of turbulence promoters for the helium cooling could help to improve the thermal design.

The issue of a first wall protection (in form of sacrificial layer) was addressed in the EU PPCS only partially. In the requirements of all the models it was prescribed the addition of a W sacrificial layer of 2 mm (average) on the overall blanket FW. This assumption was used in almost all the neutronic calculations for the evaluation of Tritium Breeding Ratio of the reactor, but not completely addressed and integrated in the design and in the manufacturing technology of the blankets. The repartition of this 2mm not excluded again that the protection could be mostly concentrated in limited FW surfaces and that a large part of the blanket wall could survive with lower (or none) W layer. A concept of EHF elements like in-ITER has been never considered in these designs.

In US some innovative ideas for a FW plasma facing material have been presented [Wang]. Here the loading conditions assumed in the design were 1 MW/m^2 in static conditions and up to 2 MW/m^2 during fast transients (of few seconds). The combination of W and steel (in a brush configuration) should enhance the thermal conductivity of the FW and at the same time improve the resistance to "erosion" of this layer.

4. CONCLUSIONS

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/m^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 MW/m^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 MW/m^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 MW/m^2.

These operational conditions are at the limit of the present technology. In particular divertor plates able to remove 10 MW/m^2 have been produced (ITER water cooled [Merola]) or are under development (KIT helium cooled divertor [Norajitra11]) but their lifetime expectation due to the behaviour of the materials under beyond-ITER neutron conditions (1 to 3 MWa/m^2) are not encouraging. A further reduction of the target flux (under 5 MW/m^2) 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. Current investigations at ASDEX Upgrade try to maximise the heating power while keeping the peak power load in the divertor acceptable. Figure 8 shows the current best results, where up to 23 MW were injected during divertor heat flux control by nitrogen seeding, keeping the peak heat flux below 5 MW/m^2. Modelling of such discharge conditions with the SOLPS code suite will set the basis for more reliable extrapolations to DEMO divertor conditions in the future.

Blanket concepts have been developed in EU for modest neutron wall load and mostly for only 0.5 MW/m^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 MW/m^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.

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Table 1: Blanket and divertor selection for the EU PPCS Models.

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/m^2 (2.5 MW/m^2)	1.84 MW/m^2	2.0 MW/m^2 (2.4 MW/m^2)	2.2 MW/m^2	2.6 W/m^2 (3.4 MW/m^2)
• Surface heating (max)	0.57 MW/m^2	0.5 MW/m^2	0.5 MW/m^2	0.59 MW/m^2	0.5 MW/m^2
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/m^2	10 MW/m^2	10 MW/m^2	10 MW/m^2	5 MW/m^2

* 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

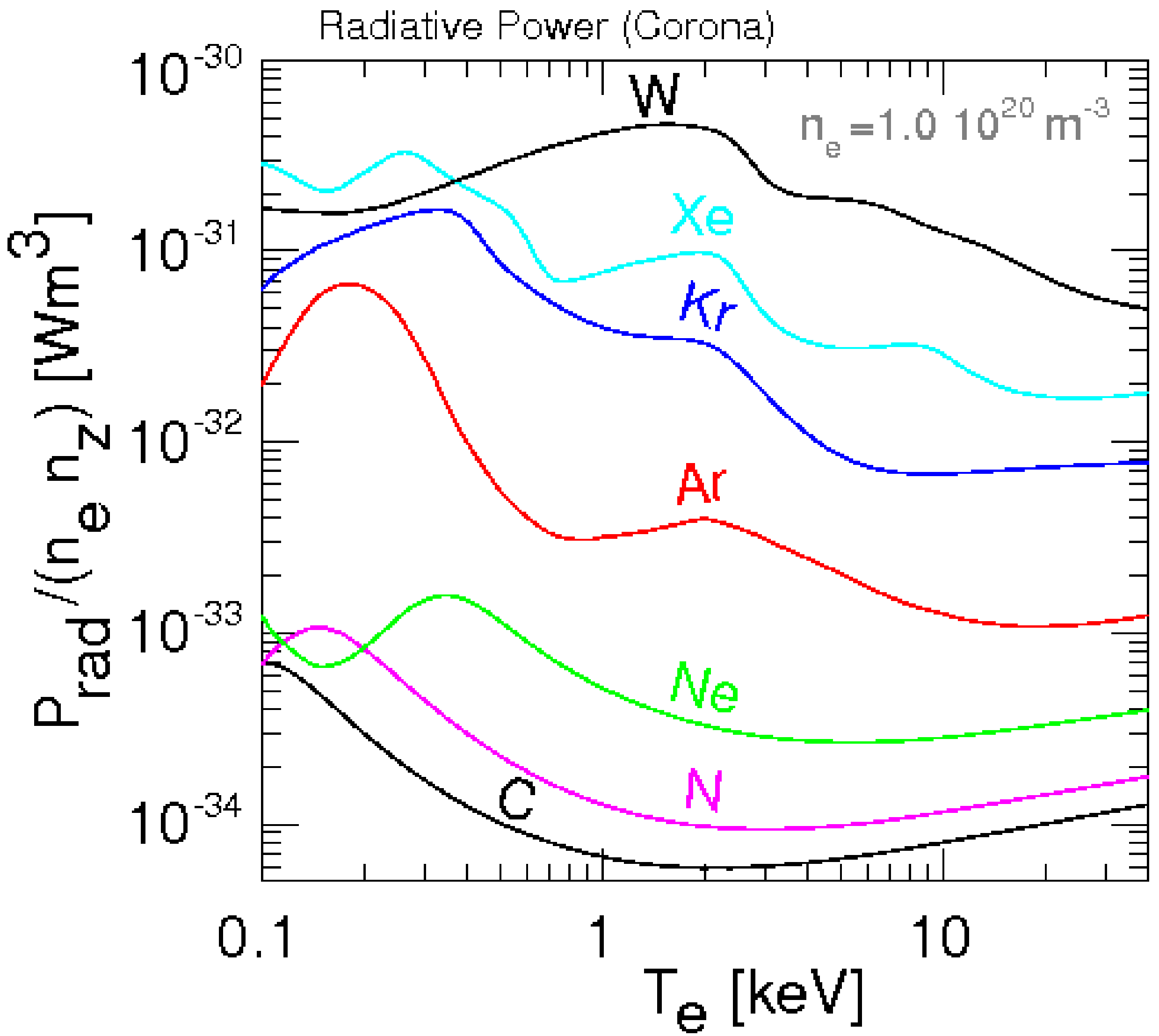

Figure 1: Radiative loss power L_z as a function of electron temperature for different species [Kallenbach]. Data are calculated from the ADAS database for an electron density of 10^{20} m^{-3} assuming Corona equilibrium. The radiated power density is obtained by multiplication with the electron and the impurity densities.

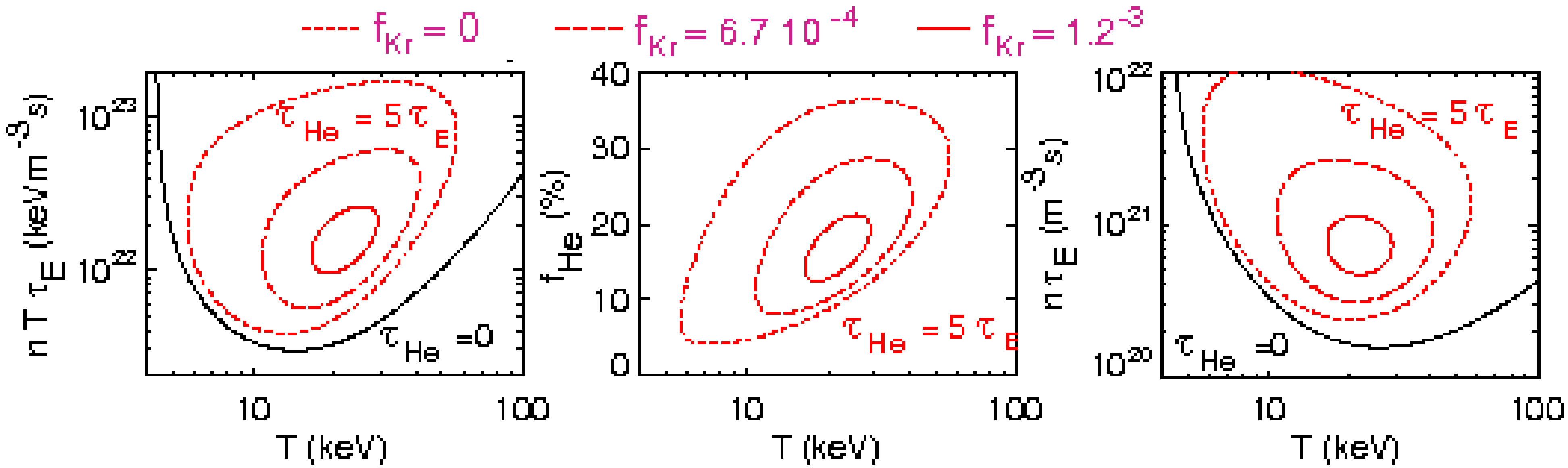

Figure 2: 0D-burn diagram [Reiter] for a DEMO case with strong Kr seeding. The He concentration is self-consistently calculated from the fusion rate assuming the He confinement time τ_{He} = 0 (black) or 5 τ_E (red). With a Kr concentration of 1.2 10^{-3}, the minimum $n\tau T$ of 9.5 10^{21} m^{-3} s keV is reached at T=18.9 keV. Assuming a burning volume of 500 m3, and an electron density of 1.5 1020 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.

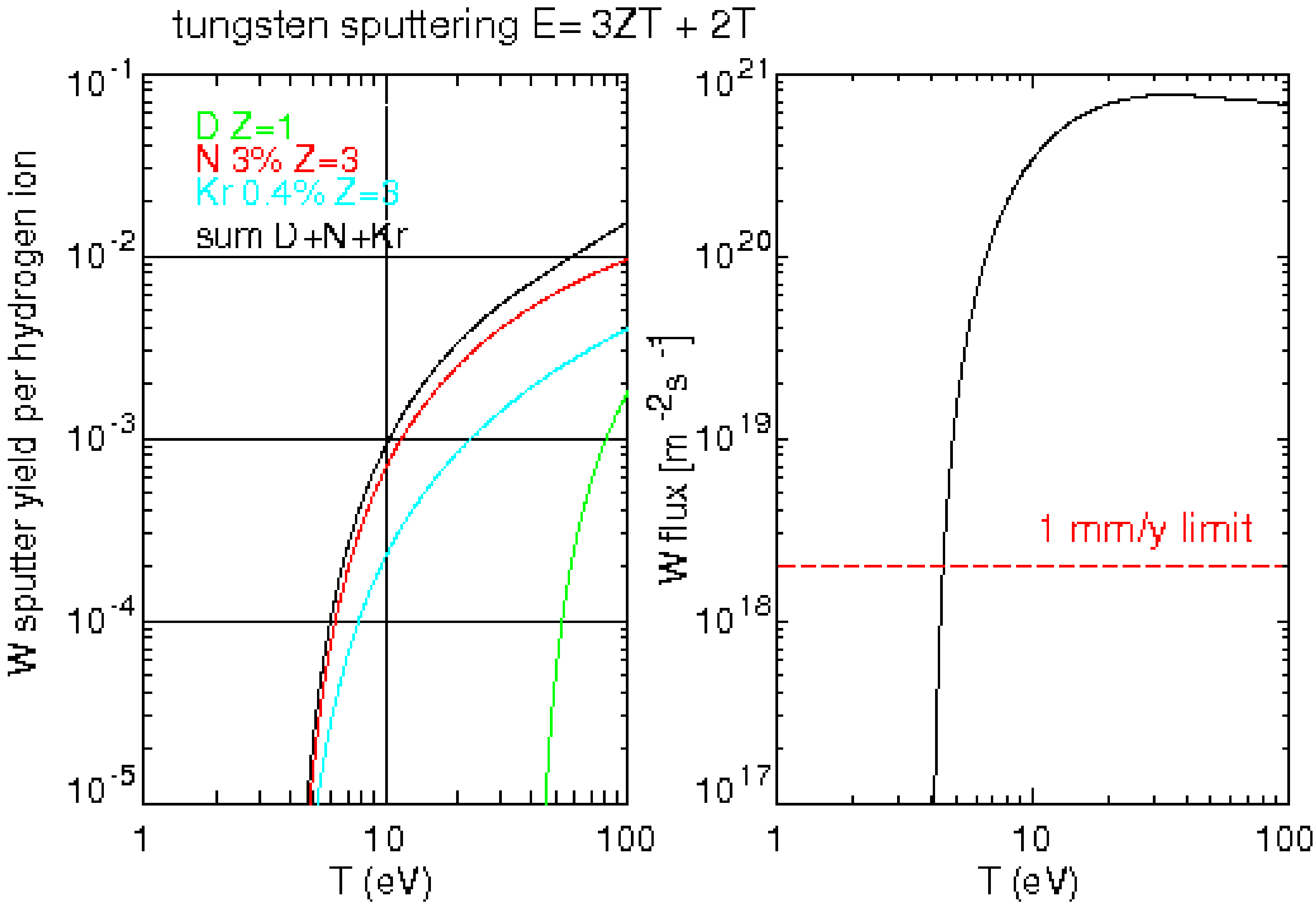

Figure 3: Effective sputtering yields for tungsten as a function of temperature for different species, assuming T_e=T_i. The black curve gives the total yield for the species mix indicated in the figure. The right picture shows the corresponding tungsten influx for the condition of the left graph. The assumed DEMO divertor erosion limit is indicated, showing that only plasma temperatures below 4 eV are acceptable in front of the divertor target.

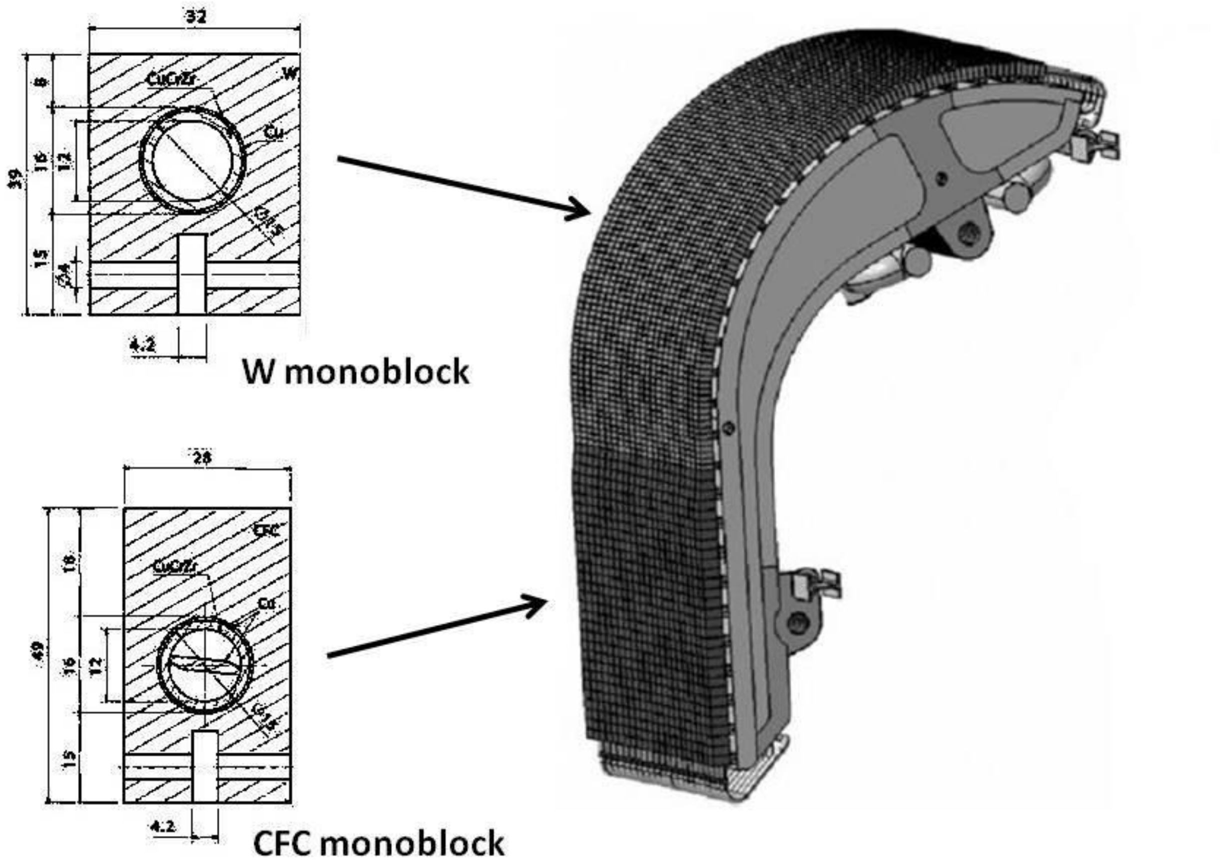

Figure 4: Illustration of outer vertical target of the ITER divertor as described in [Merola].

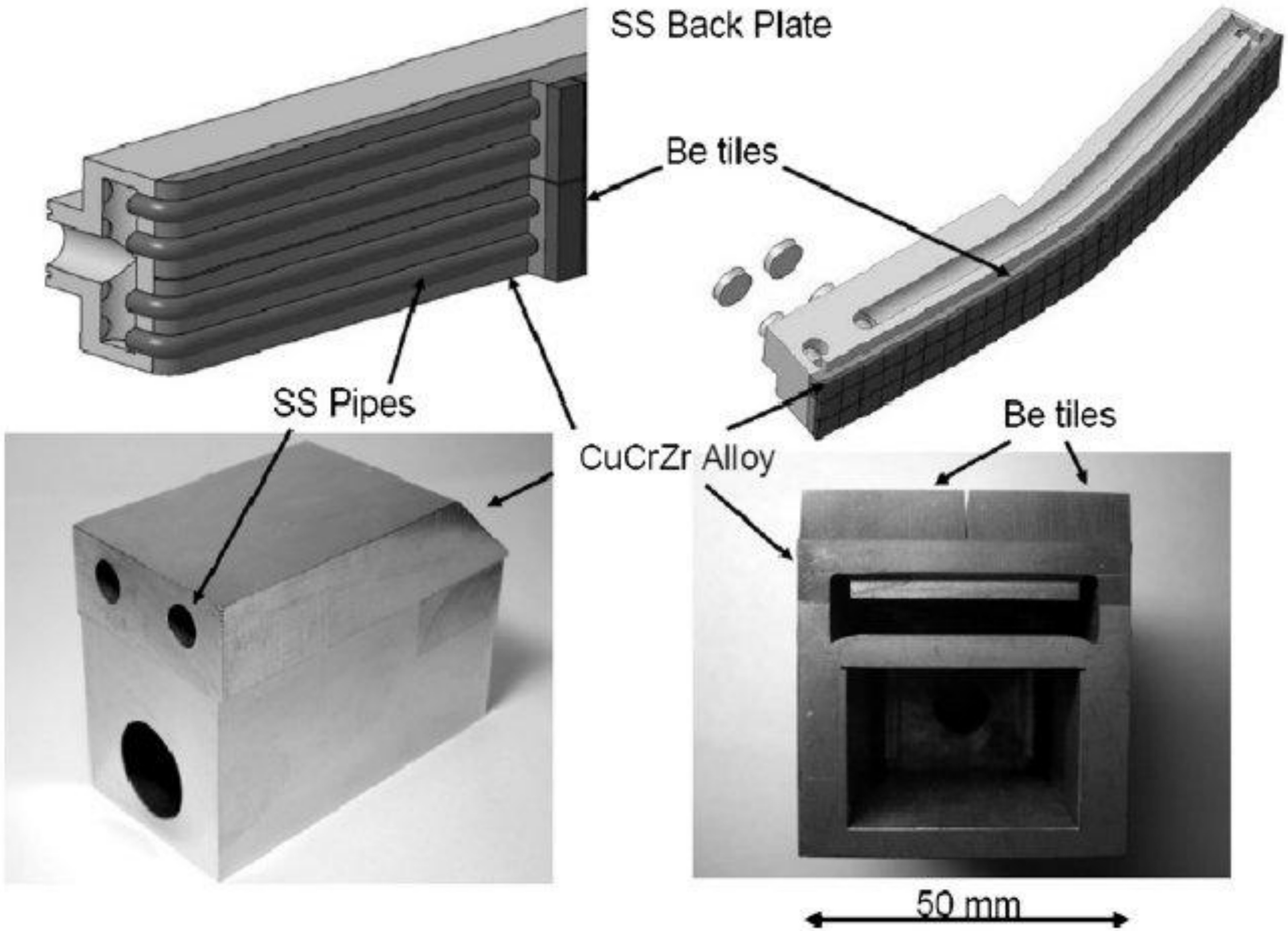

Figure 5: Illustration of the basic FW panel structure and fingers taken by [Merola]. Left: normal heat flux fingers (concept with steel cooling pipes). Right: enhanced heat flux fingers.

Figure 6: Manufacturing of the 9-finger module for the Helium cooled divertor developed in KIT [Norajitra].

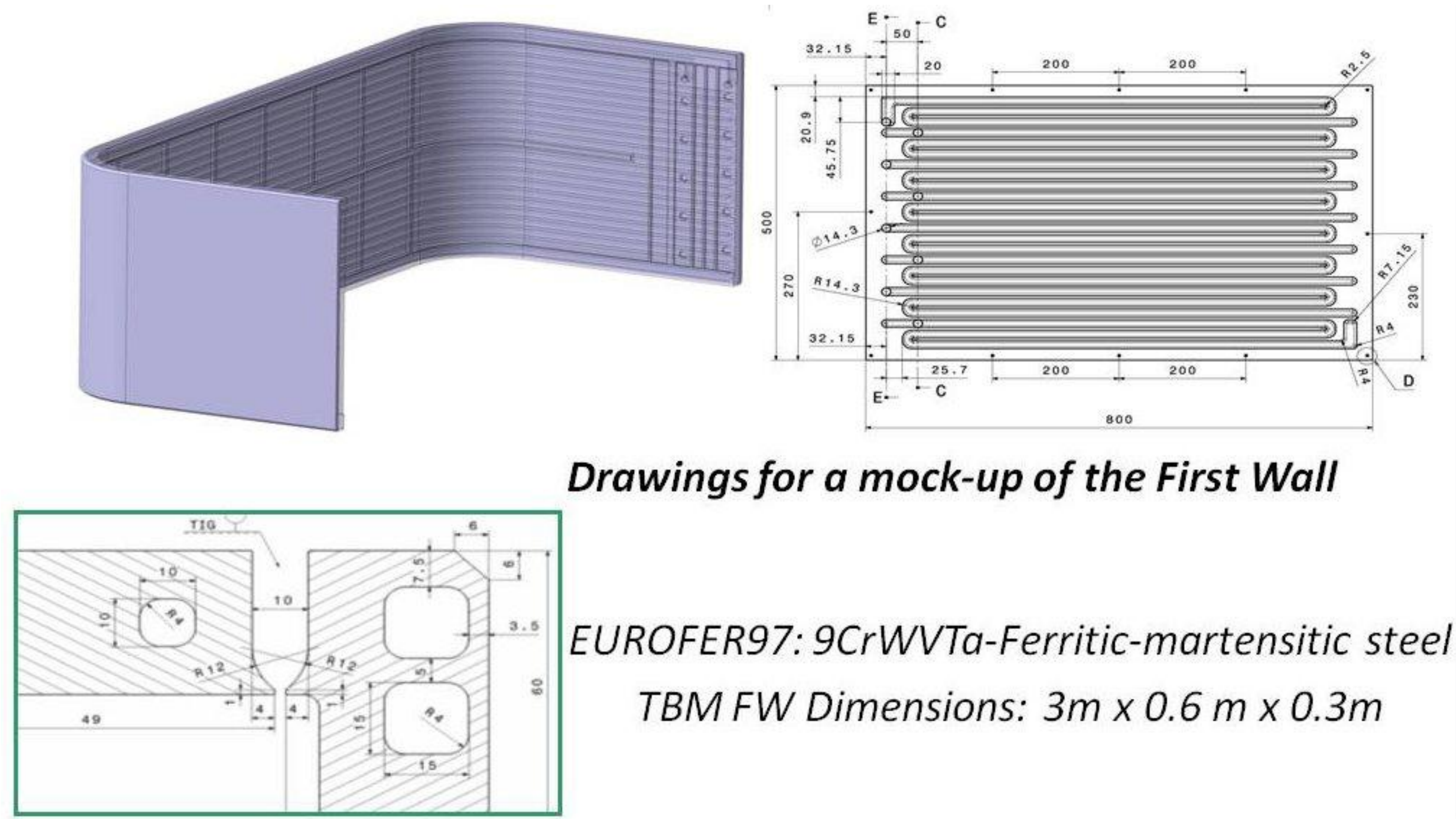

Figure 7: Drawing for the manufacturing of a FW to be tested in the TBM Programme for the EU TBMs (KIT Design).

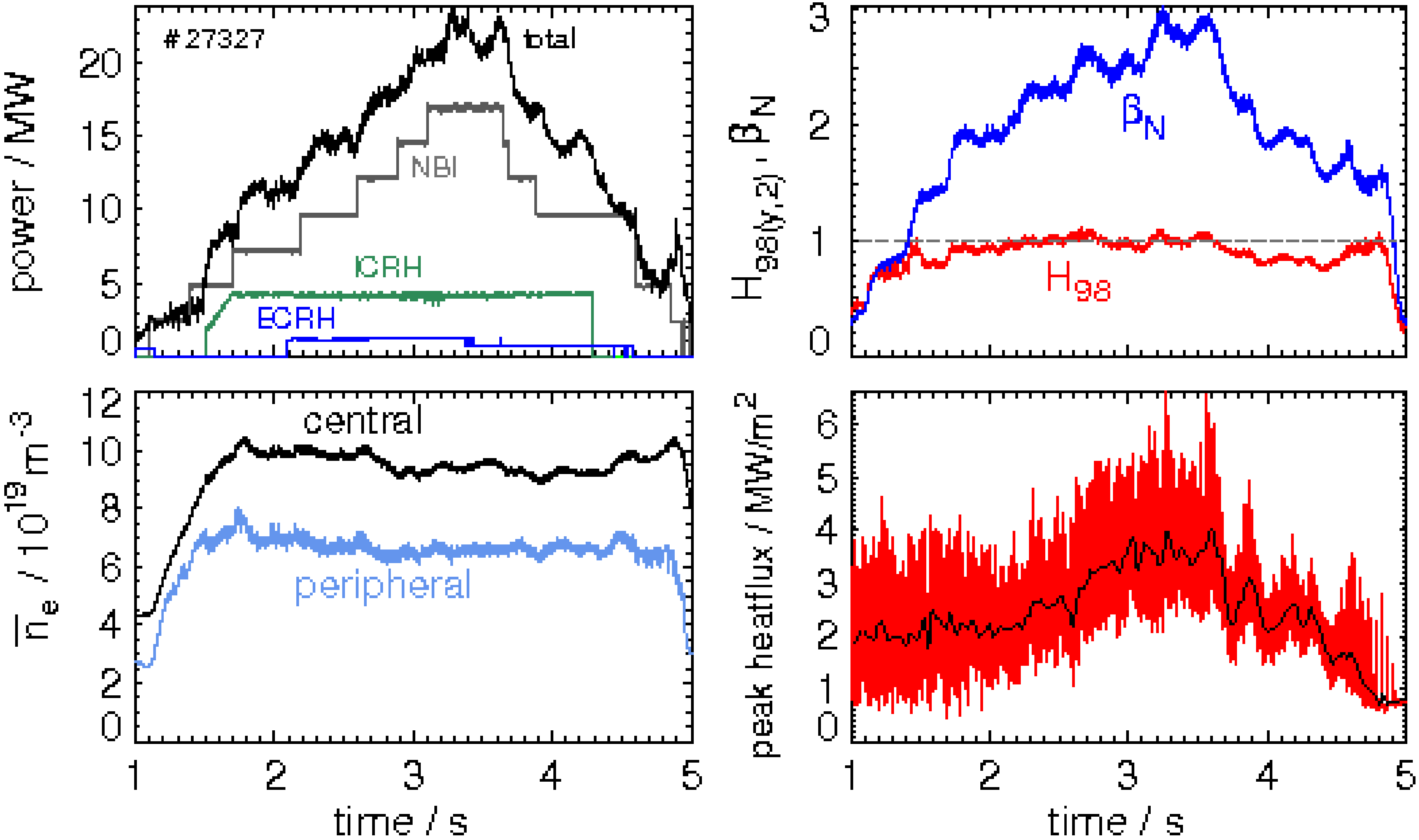

Figure 8: ASDEX Upgrade discharge with maximum heating power and divertor power dissipation by nitrogen seeding to stay below the critical value of 5 MW/m^2.