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Wall protection strategies in DEMO for plasma transients

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The present DEMO breeding blanked design heat load capability is limited to ≈1 MW/m^2 for steady state plasma, due to the specific requirements on breeding self-sufficiency, high neutron irradiation capable materials, and high coolant temperature for efficient energy conversion. While this limit is achievable in present DEMO design, by allowing enough plasma-wall distance in nominal conditions, the greatest challenges arise from the plasma transients. In this paper are presented the simulations on a preliminary list of plasma transients. 3D field-line tracing codes have been employed to estimate the maximum heat flux and energy densities for a specific first wall shape design. A scoping study thermal analysis has been performed with the code RACLETTE, using a broad range of transient input heat fluxes, on a series high heat flux (HF) components concepts, using tungsten as armor, Eurofer or copper alloy as structural materials, and helium or water as a coolant. The results allow identifying a narrow operational space of the maximum HF density that can be tolerated by each high HF concept, for the different transient models.

Keywords: DEMO, Heat wall load, Plasma Facing Components, Transient loads.

1. Introduction

The specific DEMO requirements bear a strong impact in the design and technology selection process of the components surrounding the plasma. Among the DEMO unique requirements there are: the tritium breeding cycle self-sufficiency, the high coolant temperatures needed for efficient power conversion, and the need of structural materials which can operate in high neutron fluence environment, to maximize the machine availability. The steady state heat flux density that can be sustained by the present first wall (FW) technology in DEMO is of the order of 1 MW/ m^2*[CITATION FAr \m JAu \l 1033]*. In comparison ITER fist wall (FW), which does not have to fulfil the above requirements, is designed to sustain heat flux densities up to 4.6MW/m^2*[CITATION ARR \l 1033]*. In this paper the analysis of steady state and transient heat loads is presented. The study was carried out on a preliminary 3D FW, including gaps and chamfers, and roof shaped tiles. A preliminary list of plasma transients has been analyzed. Dynamic nonlinear simulations were performed to evaluate the different time duration of each event and the location of the plasma-wall contact points, or the larger plasma shape variation to the nominal case. Conservative assumptions were made regarding the plasma energy content during these phases, based on present European DEMO scenarios. 3D field line tracing codes*[CITATION MFi13 \m WAr15 \l 1033]* where run on the resulting plasma scenarios, to evaluate the resulting heat flux density. The results of these simulations were finally compared with a broad simplified transient thermo-hydraulic analysis, performed using the code RACLETTE*[CITATION RRa97 \l 1033]*, on a series high heat flux (HHF) components concepts, using tungsten as armor, Eurofer or copper alloy as structural material, and helium or water as a coolant. The results allow an initial assessment of the heat flux capabilities to transients of a standard FW like solution, and a comparison with different possible HHF concepts.

2. Preliminary studies on wall design and static and controllable transient heat loads

An initial evaluation of the wall load specification is being performed, based on the 2015 EU DEMO baseline configuration described in*[CITATION RWe1 \l 1033]*. The DEMO heat wall loads are highly dependent on the details of the plasma-wall system. For this reason an initial engineering design of the DEMO FW was developed in*[CITATION FMa17 \l 1033]*, to evaluate the plasma steady state and controllable transient phases.

The present assumptions of the heat load due to the charged plasma particles are based on the latest DEMO predictions, and some conservative assumptions that can be found in*[CITATION RWe \l 1033]*. The total heating power P_{tot} = 450MW (alpha heating power and auxiliary heating power), of this P_{rad_core} = 300MW is radiated in the core, while the power across the separatrix is P_{sep} = 150MW. A margin is assumed, as the maximum value of $P_{sep,max}$ is considered 1.5 times the nominal value of P_{sep}. A simple model is assumed for the distribution of P_{sep}, with the power distributed in 3 main channels:

- $P_{rad,edge}$= 0.4*$P_{sep,max}$ = 92MW
- $P_{\lambda q=1mm}$ = 0.3*$P_{sep,max}$ = 69MW
- $P_{\lambda q=50mm}$ = 0.3*$P_{sep,max}$ = 69MW

With $P_{\lambda q=1mm}$ representing the part associated with the narrow scrape off layer (SOL) heat transport via charged particles and loading mainly the divertor, $P_{\lambda q=50mm}$ representing the part associated with the charged particle blobby transport, and $P_{rad,edge}$ representing the part radiated in the SOL and divertor, the latter two loading both divertor and FW. The analysis of the heat flux density calculation is carried out using the 3D field-line tracing codes*[CITATION MFi13 \m WAr15 \l 1033]*, using as input $P_{\lambda q=50mm}$ = 69MW, on an initial fist wall engineering design optimized for a e-folding length λ_q =50mm, with 15 poloidally divided sectors (Figure 1).

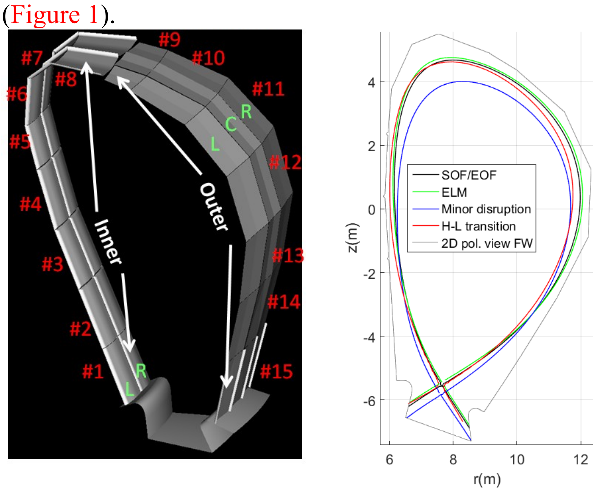

Figure 1. Preliminary 3D engineering design of DEMO FW, on the left, and maximum plasma shape perturbations due to transient events, compared with the nominal case (SOF/EOF), on the right.

The equilibria flux map used for the calculations are the nominal Start and End Of Flat top (SOF/EOF), and the maximum displaced controllable equilibria due to a series of plasma perturbations at full plasma current (*i.e.* 20MA), such as ELMs, minor disruptions, H-L transitions. The calculations on the plasma perturbations were performed using the 2D dynamic nonlinear simulation code CREATE-NL*[CITATION RAl15 \l 1033]*, with simplified assumptions on the controller. In particular a best achievable performance controller was used to control back the vertical position in the case of the ELM and minor disruption (*i.e.* a voltage was given as input to the system equal to two times the ideal value that would stop the plasma vertical movement at infinity), while an ad-hoc optimized ideal shape controller and VS controller were used to counteract the radial movement of the plasma towards the inner fist wall, for the H-L transition. Additionally the heat flux density due to the radiation were considered, as described in *[CITATION RWe \l 1033]*, where the source distribution consists 1) in a core radiation $P_{rad,core}$, and 2) a radiation source clustered around the x-point. For the analysis of the radiation wall load in DEMO these two components are superimposed, where the x-point radiation is equivalent to 1/3 to 1/5 of the total radiation. A 3D Monte-Carlo tool has been developed to calculate on the basis of this radiation source distribution the load on the wall. The tool currently does not account for absorption in the plasma or reflections at the first wall. This 3D tool was applied on a 2D wall, rotated axisymmetrically, to obtain an equivalent 2D calculation, as in *[CITATION RWe \l 1033]*. In this preliminary phase it was decided to consider the highest heat flux density per module due to:

- the different flat top and perturbed equilibria;
- the combination of the worst of the two x-point clustered power cases.

The total integrated power per module was also evaluated. Both results are conservative and do not respect the conservation of power, which will be higher than the injected, because they add up a series of worst cases, and do it by module, when not necessary the peak of heat flux for charged particles and radiation will be at the same location. The results, reported in Table 1, on the power density shows that it is possible to approach the limit on the DEMO FW technology total power density, as the maximum value is a little above 1MW/m^2 for the H-L transition on the inner central modules (Mod #4). Nevertheless this is already at the technological limit of 1MW/m^2, and further studies have to be made in order to:

- Improve this technological limit, in term of design options (*e.g.* materials, coolant, geometry)
- Further optimize the the FW contour, for instance increasing the plasma wall clearance in the zones of interest of the FW.

	Peak heat flux (MW/m²)			Total power (MW) on 360°		
	max charg. part.	max rad.	Tot.	max charg. part.	max rad.	Tot.
Mod #1	0.23	0.23	0.46	0.4	15.2	15.6
Mod #2	0.18	0.19	0.37	0.04	11.6	11.6
Mod #3	0.10	0.17	0.27	0.47	11.3	11.8
Mod #4	0.92	0.16	**1.08**	1.53	10.9	12.4
Mod #5	0.49	0.16	0.65	0.87	7.86	8.73
Mod #6	0.72	0.18	**0.9**	1.65	10.5	12.2
Mod #7	0.68	0.22	**0.9**	2.36	5.46	7.82
Mod #8	0.36	0.27	0.63	2.4	22.2	24.6
Mod #9	0.10	0.29	0.39	0.35	14.6	14.9
Mod #10	0.21	0.30	0.51	1.13	39.5	40.6
Mod #11	0.06	0.30	0.36	0.37	40.9	41.3
Mod #12	0.03	0.31	0.34	0.1	49.3	49.4
Mod #13	0.06	0.31	0.37	0.7	47.1	47.8
Mod #14	0.01	0.32	0.33	0.05	44	44
Mod #15	0.19	0.47	0.66	0.98	45.7	46.7

Table 1. Peak HF and total power per blanket module, obtained by adding the contribution of the worst cases per module for the charged particles, and the radiation.

The theoretical lower limit of the ideally homogeneous spreading of the radiation part of the power, is equal to ≈ 0.28MW/m^2, calculated as 392MW (300MWcore + 92MW SOL radiation) /1400m^2 (the approximate FW surface). A further detrimental effect will be evaluated on the misalignment of the FW, which affect mainly the charged particles contribution, due to installation, manufacturing and operational tolerances, once this data will be available. From the ITER experience this effect could be larger than a further factor 2.4*[CITATION RMi15 \l 1033]*. For DEMO this is optimistic as in ITER some misalignment adjustments can be made on the FW, which are not foreseeable at the moment for DEMO. Finally further decrease of margin will likely result once realistic controller schemes and diagnostic systems concepts will be available for DEMO. This may results in a different treatment of the perturbations that will eventually bring to excessive heat flux, such as the development of termination strategies, depending on the detection and actuators response time for the specific event. The charged particles give the highest contribution, in term of peak heat flux density, although they are not predominant in term of integrated power, as only 69 MW of $P_{\lambda q=50mm}$ are injected in the SOL, most of which ends up in the divertor. The localization of the critical values can be used by the designers to foresee a specific protection in the areas of interest, *i.e.* baffle and area close to the secondary inactive null, which will likely change with the magnetic flux map evolution from the start to the end of flat top. The total integrated power is instead mainly due to the 392MW of the core plus SOL radiation*[CITATION RWe \l 1033]*. The latter initial indications will be used to dimension the system needed to evacuate the total integrated heat on the FW.

3. Ramp-up limiter design and HF calculations

All divertor tokamaks operate in a limiter configuration for some part of the plasma discharge, during the phase of the plasma current ramp-up, before an x-point is formed, and possibly, depending on the scenario, during the current ramp-down. ITER has been designed with a wall-limiter, *i.e.* the wall panels perform the function of limiter with very small gaps between panels. Covering large areas with non-breeding FW does not seems to be viable for DEMO, because of the requirement on the tritium breeding self-sufficiency*[CITATION PPe16 \l 1033]*, and the accurate positioning of wall-limiter panels would be prohibitively costly in the present wall design, with FW and breeding blanket modules mounted on a common back supporting structure. Instead, discrete limiters are presently evaluated for DEMO. A port limiter solution, similar to the initially proposed ITER design*[CITATION ACa \l 1033]*, is being investigated, as it could be accessed via remote handling more readily than the blanket segments, and could be more accurately positioned or frequently replaced, compared to the blanket system. The latter advantage could allow the use of lower neutron damage tolerant materials. A Eurofer-based plasma-facing component would be preferable in terms of lifetime duration, but the use of a copper alloy would give far superior power handling capability, *e.g.* in case of a DEMO divertor like technological solutions*[CITATION JHY16 \l 1033]*. An initial design of outer mid-plane limiter has been proposed and analyzed. The toroidal and poloidal profiles have been optimized considering an e-folding length of 6mm, based on the DEMO prediction in *[CITATION RWe \l 1033]*. The limiters are located at the outer port of the central "banana", out of the 3 per toroidal outer sector, of the blanket. A list of limited configurations was produced for the outboard plasma current ramp-up scenario. These configurations include an increasing plasma volume and current from 3MA to 6 MA, the latter being the point in time when the plasma transit from a limited configuration to a diverted one. Although a full plasma ramp-up scenario has not yet been developed for DEMO, these values are similar with respect to the ITER predictions *[CITATION RAP11 \l 1033]*, where a transition as early as 3.5MA has been predicted. The present hypothesis of the maximum plasma current ramp up is dIp/dt = + 0.2 MA/s during ramp up, similar to the one used in the ITER studies in*[CITATION FIm11 \l 1033]*, which leads to a time for which the plasma transit from a limited to a diverted configuration from 18s up to 30s, for a transition current of 3.5MA and 6MA respectively. For the total power in the SOL during the limited ($P_{sol,lim}$) phase the same hypothesis of ITER was used, *i.e.* $P_{sol,lim}$(MW)=Ip(MA).

3D field-line tracing runs were performed for:

- 18, 9 and 3 discrete limiters, using the last limiter plasma scenario at 5MA;
- [3, 4, 5 and 6] MA, for a number of 3 limiters.

The results are shown in Table 2, and in Figure 2. The maximum heat flux (HF) density for the last limited plasma is 1.16MW/m^2 for the case with 5MA equilibrium ($P_{sol,lim}$ = 5MW) and with 3 limiters.

	Equil. Ip 5MA ($P_{sol,lim}$=5MA), # Limiters:			3 Discrete limiters, equil. Ip(MA)= $P_{sol,lim}$(MW):		
	#18	#9	#3	3MA	4MA	5MA
Limiter peak HF (MW/m^2)	0.76	0.9	1.16	1.09	0.97	1.16

Table 2. Maximum HF density on the limiter, optimized for e-folding length λ_q=6mm, for different toroidal limiter numbers, and different plasma currents/shapes.

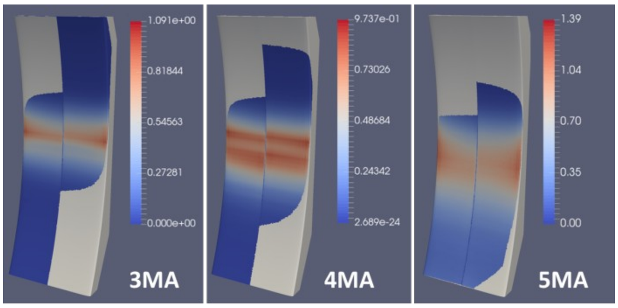

Figure 2. HF density map (color bars in MW/m^2) on 3 toroidally equally spaced discrete limiters, evaluated with 3D field-line tracing codes, using as input plasma currents/shapes from 3MA to 5MA.

The option of 3 limiters was chosen to be evaluated on the whole range of evolving limited plasmas, as the minimization of the occupied outer ports would be desirable for DEMO. A large number of ports are likely to be needed for additional heating, diagnostic and safety systems. The results for the series of equilibria [3, 4, 5 and 6] MA resulted in a non-monotonic trend of peak HF on the limiter, up to 1.16MW/m^2, while negligible values were found in the rest of the FW. This is due to the geometrical effect of the increasing poloidal matching of the 2D plasma profile to the limiter, which is concurrent with the also increase of injected power in the SOL. Detrimental effects are expected in the case of misaligned limiters, due to tolerances. A calculation of the peak HF on the limiters during plasma flat top phase was performed. The results, using the standard input parameters used for the steady state case (*i.e.*

$P_{\lambda q=50mm}$=69MW), are of a maximum peak HF of 0.15MW/m^2. These initial results do not exclude the possibility to use the described limiters as solution for the DEMO ramp-up scenario, as even a factor 10 increase on the present value would still be in reach of a technological solution as the one used for the divertor.
It has to be noted that, as the number of limiters decreased, the total integrated power in the camber did not match any more the injected power. This problem will be further investigated to understand where the missing power would be deposited, and which would be the resulting HF density in such zones.

4. Uncontrolled perturbations leading to disruptive events

An initial assessment has been carried out for the transients that bring the plasma with high energy in contact with the FW, following a perturbation. The analysis was focused in one of the most severe events represented by the upward unmitigated vertical displacement events (VDE). The assumptions used follow the one described in *[CITATION RWe17 \l 1033]*, and are hereafter briefly summarized. Two simulations are considered:

1. the plasma is assumed to move upwards at full plasma energy of 1.3GJ, as predicted in *[CITATION RWe \l 1033]*, until it is limited and q_a=2, when the thermal quench (TQ) occurs,
2. the plasma moves upwards at full energy until it becomes limited and then the energy decreases linearly, until the TQ (at q_a=2), to 0.65GJ.

The simulations are performed using the code CarMa0NL*[CITATION FVi13 \l 1033]*, which includes 3D conductive structures as blanket, vacuum vessel and ports, and the upward movement is initiated by an artificial upward kick in the vertical stabilization system. At the TQ the assumptions is that energy is conducted in the SOL with a broadening factor equal to 7 (in ITER *[CITATION RPi \l 1033]* it is assumes the range 3-10), hence bringing the input e-folding length λ_q≈7mm.

A 3D field line tracing calculation has been performed at the time of the TQ assuming 4ms deposition time of the energy of the cases 1) and 2), respectively of 1.3GJ and 0.65GJ, as defined in *[CITATION RWe \l 1033]*, with 1ms rise and 3ms decay of the power. The initial simulations were performed on a standard FW, optimized for λ_q=5cm, much different from the incident HF considered which has λ_q=7mm. The resulting power densities, reported in Table 3, are of the order of several tens of GW/m^2 for the 4ms of deposition time considered. In these range of extremely high HF some beneficial effects may be expected, in term of incident HF, from the tungsten vapor shielding, as reported in simulations and experimental validations performed for the ITER divertor, using similar power levels *[CITATION Ser16 \m SPe17 \l 1033]*.

	E(GJ)	time (ms)	$P_{\lambda q=7mm}$ (GW)	module	Max HF (GW/m²)
Case 1)	1.3	4	325	M10	**44.5**
Case 2)	0.65	4	162.5	M9	**19.2**
				M10	**25.6**

Table 3. Maximum HF density evaluated on the standard FW, for two different VDE disruptive events. The power is concentrated in the modules 9 and 10 (see Figure 1).

Nevertheless severe damages have to be expected in such extreme cases, and a failure of the present standard FW*[CITATION FAr \m JAu \l 1033]*, which is made of 2mm thick tungsten armor, 3mm Eurofer structural material, and then a high pressure coolant, seems unavoidable, with a possible loss of coolant accident (LOCA) in the machine. For this reason adequate high HF components, possibly sacrificial, have to be designed at protection of the interested contact regions of. By using these initial results, some initial studies will be performed of the design of a discrete protruding high HF component, which may use a thicker armor layer, to take advantage of the thermal inertia, and the expected vapor shielding. A careful geometrical shaping of the 2D profile of the FW may limit the poloidal range of poloidal interaction locations of the plasma, as shown in Table 3, where only module 9 and 10 where interested, limiting the coverage of such high HF components. An advantage would also be the possibility to limit the interaction plasma-wall in regions close to a port, which could make the high HF components relatively easily maintainable. Some preliminary considerations on the heat flux capability of different geometrical designs are given in the next paragraph.

5. Parametric thermo-hydraulic calculations on high heat flux component designs

The code RACELTTE*[CITATION RRa97 \l 1033]* was used to simulate a set of transient events presently foreseeable for DEMO. A broad range of constant heat flux densities, in MW/m^2, was used as an input such as to have:

- Time duration: from 0.1ms to 10s;
- Energy density: from 1MJ/m^2 to 100MJ/m^2.

For the combination of each of the above cases, a number of desired outputs were selected, such as the maximum temperature reached at the interfaces of all elements, the melting/evaporation depth of the armor, and the critical heat flux of the cooling pipe. The evaluation of these maps was repeated for a series of possible heat wall geometries:

a. Eurofer as structural material, with thickness of 2mm, and Helium as a coolant, at temperature of 400°C and velocity of 80m/s.
b. Copper alloy as structural material, with thickness of 2mm, and water as a coolant at temperature of 300°C and velocity of 8m/s.

For the two cases a sensitivity scan was performed on the armor thickness in the range [2, 10, 20 and 30] mm. The outcome of this investigation will be used to evaluate the performances of each concept on some of the characteristic time and energy content expected for each transient event. For the purpose of showing the large amount of results, a number of criteria are defined and shown in Figure 3, such as: the maximum tungsten armor melting of 100µm, the maximum Eurofer temperature of 550°C[CITATION FAr \l 1033], and the maximum copper alloy pipe temperature of 350°C[CITATION MMi08 \l 1033]. For the very fast events of the duration of ≤ 2ms, the phenomenon interest only the armor surface, as the heat flux does not have the time to diffuse in the below materials, and is almost independent of structural material and coolant choices. Above this duration the criterion on the structural material of the Eurofer concepts reaches quickly the considered limit. For the copper alloy concept the limit is reached at higher deposition time, for the same armor thickness, up to the point when the limit is not reached any more, with a thickness of at least 2cm tungsten. In the latter case, although severe damages may occur at the armor, the integrity of the cooling pipes would be verified. These results are conservative and strong mitigating effects coming are expected from the vapor shielding, not included in RACLETTE, for very high power densities. Further studies on vapor shielding are planned in this direction, using the similar tools used by ITER in [CITATION Ser16 \l 1033]. Detailed 2D/3D FEM concepts are being developed based on these very preliminary results, the specific Energy density/deposition time ranges expected for the different transients described in this paper.

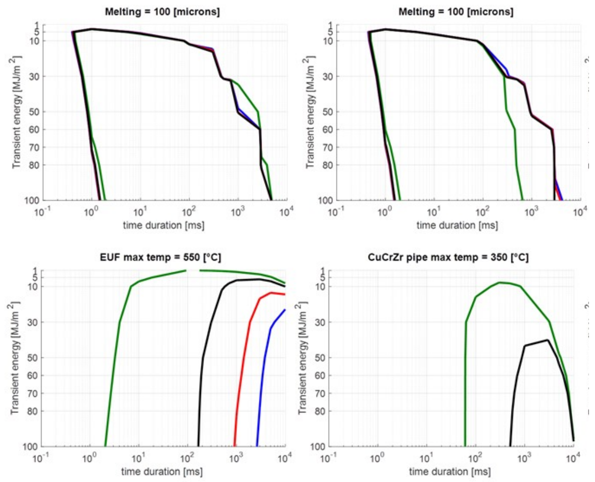

Figure 3. Maps of Energy density over deposition time, evaluated with the code RACLETTE, of: 100µm tungsten armor melting (top figures), and structural material temperature (bottom figures) equal to 550°C for Eurofer (EUF), and 350 for Copper alloy (CuCrZr). Simulation run on the geometry a) (left figure), and b) (right figure) described in paragraph 5, with armour thickness 2mm (green), 10mm (black), 20mm (red) and 30mm (blue).

6. Conclusions

In this paper it is reported an update on the status of the transient heat wall loads in DEMO. The aim was to establish the methodology which will allow to evaluate a larger and more precise list of these events, and to understand the implication of the plasma perturbations due to them. After a prelaminar optimization of the DEMO 2D poloidal contour and 3D shape of the wall, it seems to be possible to approach the technological FW limit of ≈1MW/m^2, presently considered for DEMO[CITATION FAr \m JAu \l 1033], including charged particles and radiation contributions. This is also partially due to the relative large minimum plasma-wall distance of 23cm that is presently considered for DEMO scenarios, while preserving the vertical stability controllability of the plasma. The margin to the HF limit appears to be nevertheless small, or insufficient if all the detrimental effects are accounted for, such uncertainties, misalignments and tolerances. Further improvements on the FW exhaust capability are needed. The outcome of this activity will help the FW designer regarding foreseeable high HF regions, mainly localized close to the upper inactive null, and in the baffle region, and the cooling system designers, which have to design the system to exhaust the radiation to the wall. A solution, similar to early ITER case[CITATION ACa \l 1033], with 3 discrete limiters was proposed and analyzed for the plasma ramp-up phases, resulting in heat flux densities up to 1.2MW/m^2, with negligible amount in the standard FW. A preliminary assessment was also carried out on the effect of unmitigated disruptions on the standard FW. The extreme heat flux densities foreseen for these occurrences indicates that specific high heat flux component solutions are needed to avoid the breaching of the standard FW. Some indication on the possible geometry and materials of possible concepts were derived from a broad range of thermo-hydraulic calculations, performed using the code RACLETTE. These analyses will be assessed in the future with specific codes, similarly with what is being done in ITER [CITATION Ser16 \m SPe17 \l 1033], as the effect of the vapor shielding is expected to help to mitigate the loss of armor estimated. Further activities are being performed regarding the sensitivity of the HF due to misalignments and tolerances, of the standard FW and discrete high HF components, together with studies on their impact on tritium breeding, preliminary remote maintenance, and integration concepts.

Acknowledgments

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