IMPLEMENTATION OF 3-D EFFECTS OF THE PLASMA-FACING COMPONENTS IN A 2-D REAL-TIME MODEL-BASED APPROACH FOR WALL HEAT FLUX CONTROL ON ITER

H. Anand¹, R. A. Pitts¹, P. C. De Vries¹, J. A. Snipes¹, L. Kos⁴, Y. Gribov¹, F. Nespoli¹, C. Galperti², S. Coda², B. Labit², L. Zabeo¹, I. Nunes¹ and M. Brank⁴

¹ITER Organization
Route de Vinon-sur-Verdon, CS 90 046, 13067 St.-Paul-lez-Durance Cedex, France
Email: himank.anand@iter.org
²École Polytechnique Fédérale de Lausanne, Swiss Plasma Center
CH-1015 Lausanne, Switzerland
³Aix Marseille University, CNRS
Centrale Marseille, M2P2, Marseille, France
⁴University of Ljubljana
Askerceva 6, 1000 Ljubljana, Slovenia

Abstract

A control oriented approach including the effect of 3-D geometry of the plasma-facing components for monitoring the power flux density, based on real-time equilibrium reconstruction for ITER has been successfully developed. The model-based approach, in the simplest case, describes the deposited heat flux as a poloidal flux function with two parameters to be specified by the modeller: the power exhausted across the plasma boundary and the scrape-off layer width. An additional module containing weighting factors accounts for the true 3-D geometry of the first wall panels. Integration of the 3-D effect is performed by using a new GUI interface, SMITER, hosting a magnetic field line tracing code and permitting import and appropriate meshing of full CAD descriptions of the plasma-facing components. The paper reports on the typical surface heat load distributions for the ITER limiter start-up plasma phase on an inboard midplane first wall panels with a double exponential radial heat flux profile, according to the findings of recent multi-machine studies of inboard limiter plasmas. Furthermore, it discusses the heat loads on the first wall during a vertical movement of the plasma in the ITER baseline scenario with the effect of ELMs on the radial profile of the parallel heat flux.

1. INTRODUCTION

The use of water cooled plasma-facing components (PFCs) [1] during burning plasma operation [2] in ITER, imposes limits on the heat flux deposition. Thus, a robust and reliable real-time (RT) monitoring and control of PFC heat fluxes is mandatory for the ITER tokamak [3]. At ITER, the monitoring and protection of PFCs will be performed by the wide angle viewing system (WAVS) comprising visible (VIS) and infrared (IR) cameras. A sophisticated off-line field line tracer package, SMITER (uses the SMARTDA kernel [4]) has also been successfully developed to allow power deposition mapping on the full 3D CAD geometry of ITER. The demanding computational load restricts its application in RT, so a control oriented heat flux monitoring system accounting for the effect of 3D PFCs, based on Matlab/Simulink software [5] was developed for the ITER plasma control system (PCS) [6].

Various tokamaks have successfully demonstrated the capability of RT control to prevent the overheating of PFCs based on imaging diagnostics [7]–[9]. In addition, RT model-based techniques not relying on imaging systems have also been successfully tested for estimation of the PFC heat flux deposition [10], [11]. In addition, approaches relying on Magnetic field line tracing codes [12], [13] have been used routinely to estimate the heat flux deposition on the ITER PFCs [14], [15].

Plasma current ramp-up in a limiter plasma configuration on the Beryllium (Be) FW panels (FWPs) is foreseen for all ITER discharges, with a preference for the inner-wall (IW) surfaces [16], [17]. Current scenario design aims at transitions to divertor configuration for plasma currents of order I_p = 3.5 MA after a duration ~ 10 s [18]. Depending on the achievable blanket alignment, limiter phase heat flux densities on the shaped FWPs in the vicinity of plasma contact may approach the maximum design values [14] and hence the deposited heat flux must be monitored and carefully controlled.

In the diverted phases, constituting the majority of plasma operation time, the baseline use of equilibria at high triangularity will also impose high heat flux densities on the FWPs in the upper regions of the chamber.
intersecting the second separatrix. During diverted operation, even at moderate input power, heat flux monitoring and plasma position/shape control are mandatory at all times as a consequence of the relatively low power handling capability of the actively cooled panels of \(2.0 - 4.7 \text{ MW/m}^2\) compared to that of the divertor target (10 MW/m²). Unlike the inertially cooled PFCs common to many of today's devices, ITER's actively cooled Be FWPs cannot sustain intense heat flux densities for long before the critical heat flux is reached at the cooling interface.

**FIG. 1.** (a) Poloidal cross-section of the ITER tokamak with the referencing numbers and associated heat flux design limits for the FW PFCs (b) and detailed 3-D view of an inner wall FWP.

**FIG. 1(a)** shows a portion of the ITER first wall, illustrating the modular FWPs attached to massive stainless steel shield blocks. Illustrated in **FIG. 1(b)** is a FWP comprising a double-winged structure in the toroidal direction symmetrically disposed about a central, poloidally running slot that provides space for mechanical and hydraulic connections as well as for various plasma diagnostic systems \([19], [20]\). The physics heat load specifications for the FW and divertor \([17]\) have led to a requirement for different power handling capabilities at different regions of the poloidal cross-section. As a result, the FWPs are categorized as: a ‘normal heat flux’ (NHF) technology (up to \(2 \text{ MW/m}^2\)) and an ‘enhanced heat flux’ (EHF) technology (up to \(4.7 \text{ MW/m}^2\)). **FIG. 1(a)** shows the heat load specifications over all the FWPs. Rows 1-2-6-10-11-12-13-18 are equipped with NHF panels, while rows 3-4-5 and 14-15-16-17 are equipped with EHF panels \([19], [20]\).

The RT model-based approach, in the simplest case, describes the heat flux deposited on PFCs as a poloidal flux function with two parameters to be specified by the modeler: the power exhausted across the plasma boundary, \(P_{SOL}\), and the scrape-off layer (SOL) width, \(\lambda_q\). These two parameters are used to constitute an exponentially decaying radial parallel heat flux profile \(q_\parallel(r)\) imposed at the last closed flux surface (LCFS) or separatrix on the outboard midplane, which is then magnetically mapped to the points of FW contact. We assume that the heat flux flows along the field line and that no local diffusive cross-field heat transport near the plasma facing components takes place. Assuming only heat flow parallel to magnetic field lines, the power density ultimately deposited at any point of a given PFC surface depends on the angle with respect to the incident field line and on the local flux expansion. ‘Global power balance’ is used to determine \(P_{SOL}\):

\[
P_{SOL} = P_{OHMIC} + P_{AUX} - P_{RAD} - \frac{dW}{dt}
\]

where, \(P_{RAD}\) is the power radiated in the core plasma, \(P_{OHMIC}\) is the Ohmic power, \(P_{AUX}\) the external heating power input and \(dW/dt\) is the rate of change of the stored energy. The expression for the power density perpendicular to the surface, \(q_{dep}\), is as follows:

\[
q_{dep}(\psi) = \frac{FP_{SOL}}{2\pi R_{OMP} \Delta_q} \frac{B_\theta}{B_{\theta,OMP}} \frac{\sin(\alpha)}{\sin(\zeta)} \exp \left( \frac{-(\psi - \psi_b)}{B_{\theta,OMP} R_{OMP} \Delta_q} \right)
\]
where, $\psi$ is the poloidal flux, $\psi_p$ is the poloidal flux at the LCFS/separatrix, $B_{\phi,omp}$ is the poloidal magnetic field at the OMP (outer midplane), $B_\theta$ is the poloidal magnetic field at the point of calculation, $\zeta = \arctan \left( \frac{B_\psi}{B_\theta} \right)$ is the field line pitch angle, $\alpha$ is the angle between the field line and the physical surface and $B_\phi$ is the component of the field in the toroidal direction at the point of calculation.

An additional module containing weighting factors has been implemented in the 2-D RT model based approach to include the true 3-D geometry of the FWP. This is an essential modification if a more realistic value for the true maximum heat flux is to be correctly predicted. The maximum will always be found in the simple model at the point where the product of the decay heat flux by the incidence angle in the poloidal plane is maximum. However, FWP shaping means that the real angle of incidence will be different from that estimated by the toroidally symmetric assumption, leading to a different peak heat flux density on the PFC surface in magnitude and location, when 3-D effects are considered. Integration of the 3-D effect into the algorithm is performed by offline determination of the heat load distribution on the full 3-D poloidal sector using a new utility, SMITER, developed at the ITER Organization, in which the SMARDDA field line tracing code [4] has been embedded in a GUI interface permitting import and appropriate meshing of full CAD descriptions of the FW geometry. For a given magnetic equilibrium, weighting factors associated with the position in the poloidal plane and magnitudes of the peak heat flux are extracted for implementation into the 2-D model based approach. The improved RT model-based approach has been experimentally tested and successfully validated on the TCV tokamak using infra-red measurements of the central column surface power flux density [21].

2. HEAT LOADS DURING THE ITER LIMITER START-UP PHASE

The model-based algorithm has been used to estimate the power flux density on the inner FWPs 3-5 during the limiter start-up phase. The tokamak simulation code DINA [16] provides the poloidal flux distribution by solving the Grad-Shafranov equilibrium equation. FIG. 2 illustrates the early current ramp-up phase, full scenario simulation obtained with DINA, used here to test the control algorithm. The heat loads on the FWPs are estimated with an approximation of a double exponential radial profile of the parallel heat flux at the OMP consisting of two heat flux channels: the ‘far-SOL’ with e-folding length $\lambda_{q,far}$ and the additional ‘near-SOL’ featuring the e-folding length $\lambda_{q,near}$. The respective e-folding lengths are shown in FIG. 2 (b-c) and are derived using the recent multi-machine studies of inboard limiter plasmas [19]. The evolution of $P_{SOL}$ during the start-up phase is estimated using a simplified radiation model accounting for the line radiation of impurities using the model described in [23] and is shown in FIG. 2(d).

FIG. 2. (a) The plasma boundary at various time instances during the start-up phase obtained from a DINA code simulation. Time evolution of the (b) near-SOL width, (c) far-SOL width and (d) $P_{SOL}$.

FIG. 3 shows as an example of the heat load distribution obtained from SMITER for a limiter start-up plasma magnetic equilibrium on the inboard midplane FWPs 3-5. FIG. 3(b-c) also shows the comparison between the peak power flux density obtained from SMITER and the model-based approach for various plasma configurations. A reduction in the peak power flux density is observed during the plasma scenario with both, SMITER and the model-based based approach due to changes of the plasma shape as the plasma current
increases. It is evident that the 2-D model assuming a cylindrically symmetric FW grossly underestimates the magnitude of the peak heat fluxes and, of course, has no knowledge of their toroidal localization (it has knowledge in the poloidal plane but not the toroidal plane). This is a major drawback of an approach in which complexity (in this case the 3D geometry) is sacrificed in exchange for computational speed (cycle time ~ 3 ms). However, a typical magnetic field line trace on the full 3D poloidal sector of ITER tokamak typically takes ~ 300 s, restricting its application in RT. Thus, a weighting factor, $W_F$, defined as the ratio of the peak power flux density obtained from SMITER and the model-based approach is calculated and implemented in the 2-D model-based approach to include 3-D effects.

3. HEAT LOADS DURING A RIGID VERTICAL MOVEMENT OF THE ITER BASELINE SCENARIO PLASMA

The power density fluxes due to the ELM (edge localized mode) [22] filament-first wall interaction is studied during rigid vertical displacement of the ITER baseline plasma equilibrium (plasma current $I_p = 15$ MA and edge safety factor $q_{95} = 3$) [23], taking into account the full, three-dimensional structure of the FWPs, thus including all magnetic shadowing effects. Type I ELM heat loads on the ITER main chamber PPCs are
investigated for the plasma equilibria shown in FIG. 4(a). Assuming the baseline plasma equilibria as a rigid body, subsequent magnetic equilibria are constructed assuming an exponential growth of the plasma vertical position with a typical time constant \( \sim 0.2 \) s and an initial perturbation of 0.01 m. FIG. 4(b). The zoomed view of the primary separatrix and its proximity to the FWP 11 for various vertical plasma displacements is shown in FIG. 4(c). The baseline burning plasma equilibrium under investigation assumes a distance of \( \Delta_{sep} = 9 \) cm between the primary and the secondary separatrix. This was considered appropriate to ensure low power fluxes on the upper part of the first wall, but a more refined estimate of this stationary loading, including the inter-ELM and average ELM heat flux densities is required, which is performed here. The heat load specifications for the inter-ELM heat flux are adopted from [17], [24]. The model for estimating the power densities due to ELM transients represents a rather simple approximation to the filament description, where the filaments are assumed to propagate coherently across the SOL without fragmentation and are launched into the SOL at the OMP, with no poloidal spreading. The details of the model are described in [25]. The radial profile of the maximum ELM-averaged parallel heat flux density in the SOL, \( \langle q_{\|,ELM} \rangle \) as is follows,

\[
\langle q_{\|,ELM} \rangle = \left( \frac{1}{17} \frac{W_{fil}}{W_{fil0}} f_{ELM} \right) \left( \frac{2\pi R_{OMP}}{4\pi R_{OMP} \lambda_{q,\|}} \right) \left( \frac{B_p}{B_{tot,OMP}} \right) \left( \frac{n_{fil}}{n_{fil0}} \right) \left( \frac{\lambda_{ELM}}{\lambda_{q,\|}} \right) \]

\text{Eq. 3}

where, \( \left( \frac{W_{fil}}{W_{fil0}} \right) \) is the fraction of the ELM energy reaching a given radial distance from the \( 1^{st} \) separatrix, \( f_{ELM} \) is the ELM frequency, \( R_{omp} \) is the major radius at the OMP, \( \frac{B_p}{B_{tot,OMP}} \) is the magnetic field line pitch angle at the OMP, \( \lambda_{q,\|} \) is the local ELM energy decay length, \( n_{fil} \) is the number of filaments per ELM, \( w_{fil} \) is the poloidal filament width, and \( d \) is the poloidal separation of the filaments, assuming \( w_{fil} = d/2 \). The factor \((1/17)\) in Eq. 3 accounts for the fact that the filaments strike the first wall at random toroidal position but can overlap. In case of the lowest overlap (i.e. the minimum ELM-averaged heat load), the heat load to the first wall is reduced by 1.7 compared with the heat load that would arise if all filaments were to strike the wall at the same toroidal location. The first term in the brackets in Eq. 3 accounts for the toroidally-uniform poloidal ELM heat flux density at the OMP (outer midplane) (i.e. the common power balance formula used to estimate the radial profile of the parallel heat flux for the steady-state plasma conditions) while the second term in the brackets allows to increase the toroidally-uniform poloidal ELM heat flux density across the OMP due to the fact that the filaments fill (and thus carry the ELM heat load within) only \( \left( \frac{B_p}{B_{tot,OMP}} \right) \) of the toroidal circumference.

For the present work, the ELMs are characterized by \( W_{fil0} = 1 \) MJ, \( f_{ELM} = 40 \) Hz, \( n_{fil} = 10 \) and \( d = 0.6 \) m. A fluid model of the parallel ELM filament transport in the SOL (referred to here as the ‘parallel loss model’, PLM) [26] is used to compute \( \left( \frac{W_{fil}}{W_{fil0}} \right) \) and \( \lambda_{q,\|} \) (and, thus \( \langle q_{\|,ELM} \rangle \) ) as a function of the radial distance from the \( 1^{st} \) separatrix. In the model, the filament transport is described in the filament frame of reference by the temporal evolution of a Gaussian structure in which the initial particle and energy content decreases due to parallel losses along the flux tube connected at each end to the solid surface. The parallel particle and energy sink rate is proportional to \( c_s/L_{||} \), where \( c_s \) the ELM filament ion sound speed. \( L_{||} \) is the distance along the field lines between the OMP and the nearest surface. The radial distance from the filament birth location and the filament evolution time are related by the radial filament propagation speed, \( v_r \). We assume that the ELM filaments are ejected into the SOL from the \( 1^{st} \) separatrix with temperatures and densities equal to half of the pedestal top values \( T_{ped} = T_{ped} = 5 \) keV, \( n_{ped} = 7.5e+10^9 \) m\(^{-3} \). Furthermore, a constant value of \( v_r = 500 \) m/s, across the SOL is assumed in the PLM, and corresponds to the largest anticipated \( v_r \) for controlled ELMs in ITER. The input radial profile of \( L_{||} \) and \( \langle q_{\|,ELM} \rangle \) obtained from the PLM are shown in FIG. 5. The value of \( L_{||} \) used in the model corresponds to the shorter of the connection lengths to the inner and the outer target. The parallel connection length is characterized by a sharp drop due to the connection to the FWP 11 for all magnetic equilibria. In this region the parallel sink term becomes stronger. As a result, \( \left( \frac{W_{fil}}{W_{fil0}} \right) \) steepens, though not as much as the local energy decay length \( \lambda_{q,\|} \). This leads to the local increase of \( \langle q_{\|,ELM} \rangle \) at the radius of an intersecting FWP, which appears as a bump on the tail of the \( \langle q_{\|,ELM} \rangle \) profile. This behaviour is the result of our modelling assumptions, by which instantaneous heat fluxes are transformed into average fluxes; in reality a flattening of the power flux profile in that region will occur. In is important to note that as the plasma moves vertically upwards, the contribution of the ELM-averaged heat flux density becomes dominant in comparison to the inter-ELM heat flux profile in the far-SOL.

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The surface heat loads to the FWPs are determined using SMITER and the model based approach and comprises the static inter-ELM heat flux and intermittent ELM filament heat pulses. The SMITER analysis focuses on the deposited heat flux in the region of FWP 8, which is closest to the primary separatrix and FWP 11 that encounters intersection with magnetic field lines as the equilibria are displaced vertically. FIG. 6 shows a typical surface heat load distribution obtained with SMITER for FWP 8 and 11 for magnetic equilibria characterised by different vertical displacements. FIG. 6(a, b) shows the deposited heat flux for the ITER baseline plasma scenario; as expected the peak value of the power flux density on FWP 8 and 11 is well below the power handling capability (4.7 MW/m$^2$ and 2.0 MW/m$^2$). However, for a magnetic equilibrium with a vertical displacement of 20 cm, the deposited heat flux distribution on the FWP 11 exceeds its power handling capability. It is important to note that for such large displacements (beyond 16.5-17 cm), the plasma vertical position cannot be recovered by the use of the in-vessel vertical stability coils and the plasma will eventually disrupt. The intersection of the magnetic field lines and the dominance of the maximum ELM-averaged parallel heat flux density profile in the far-SOL result in high heat loads on FWP 11 and negligible heat flux density on FWP 8 (FIG. 6(e)). FIG. 6(b, d and f) illustrate the change in the power flux distribution pattern and peak deposited heat flux on the ELM-wetted FWP 11 during a vertical plasma displacement.

FIG. 6. Deposited power flux density on the ELM-wetted FWP 8 and 11 for different vertically displaced magnetic equilibrium.
FIG. 7 shows the comparison between the peak power flux density on FWP 11, obtained from SMITER and the model-based approach for various plasma configurations during the vertical displacement event. As expected, the evolution of the peak value of the power flux density from SMITER (FIG. 7(b)) and model-based approach (FIG. 7(c)) are similar during the event; however the model-based approach grossly underestimates the magnitude of the peak heat fluxes. This drawback of too low absolute values of power flux density is compensated by implementing the weight factor, W_F. The study also shows that the peak power flux density estimated by SMITER on the FWP 11 is below the design limit of 2 MW/m² for a magnetic equilibrium with a rigid displacement of 17 cm; this is the maximum value that the plasma vertical position can recover by the use of the in-vessel vertical stability coils. Our result shows that displacements of the 15 MA / 5.3 T Q = 10 plasma up to 17 cm can be thus controlled by the vertical stability coils without leading to unacceptable loads on the first wall.

4. SUMMARY AND CONCLUSION

A control oriented approach for estimating the first wall power flux density in ITER has been presented in this paper. The performance of the system has been tested on the ITER limiter start-up plasma phase and during a vertical displacement event of the ITER baseline scenario. Appropriate weight factors for both cases were determined by accounting for the full, 3-dimensional structure of the ITER first wall, including magnetic shadowing effects. For the ITER limiter start-up plasma phase on the inboard midplane first wall panels with the radial heat flux profile derived, according to the findings of recent multi-machine studies of inboard limiter plasmas, the steady state surface heat load stays below the maximum power handling capability of the FWPs. The simulation of the heat loads during the vertical displacement event of the ITER baseline scenario is studied using a simple ad-hoc fluid model of the ELM filament parallel transport. Due to uncertainties associated with the incomplete physical model of the ELM heat loads to the first wall, conservative assumptions are adopted wherever possible. Under the assumption of rigid vertical plasma displacement, the heat load analysis on the FWP 11 indicates that controlled ELM filaments in ITER will not lead to melting or significant evaporation of the beryllium surface for a baseline magnetic equilibrium shifted by 17 cm from the plasma magnetic axis.

DISCLAIMER

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BIBLIOGRAPHY


