Experimental implementation of a real-time power flux estimator for the ITER first wall on the TCV tokamak

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Abstract

A control-oriented approach to the monitoring of wall power flux densities on ITER has been successfully developed. It is based on real-time equilibrium reconstruction in 2-D which is then used to describe the deposited heat flux as a poloidal flux function with user specified parameters for the power exhausted into the scrape-off layer (SOL) and the SOL heat flux width. To account for the real 3-D geometry of the plasma-facing components (PFC), appropriate weighting factors are derived from magnetic field line tracing in 3-D. Integration of the 3-D effect is performed with a new GUI-based software environment, SMITER, incorporating a field line tracer and permitting import and meshing of PFC CAD models. The paper discusses the experimental demonstration of the model-based wall heat flux algorithm on the TCV tokamak, reporting on the benchmarking of the new code package, SMITER against infra-red camera heat flux measurements and the derivation of the component shaping weighting factors. A comparison of the real-time estimation of the peak power flux and its spatial location against the off-line infra-red measurement for limiter plasma configurations is presented.

1 Introduction

Burning plasma operation in ITER targets the achievement of high fusion gain, $Q_{DT} = 10$, for durations in the range 300 – 500 s [1]. During such plasma discharges, the resulting power flux densities on the PFCs will be managed under stationary, long pulse conditions using active cooling, with water the chosen medium on ITER [2]. The use of water cooling, particularly in the nuclear environment of ITER, imposes strict limits on PFC heat fluxes, making a robust and reliable real-time (RT) monitoring and control of PFC heat fluxes mandatory [3].

Various tokamaks have successfully demonstrated the capability of RT control to prevent PFC overheating based on imaging diagnostics [4, 5, 6, 7]. In addition, RT model-based techniques not relying on imaging systems have also been successfully tested for estimation of the PFC heat flux deposition [8, 9]. At ITER, the monitoring and protection of PFCs will be performed by a comprehensive wide angle viewing system (WAVS) comprising visible (VIS) and infrared (IR) cameras [10]. In addition, approaches relying on magnetic field line tracing codes [11, 12] have been used routinely to estimate the heat flux deposition on the ITER PFCs [13, 14]. Recently, a sophisticated Graphical User Interface (GUI), SMITER [15] that uses the SMARDDA [16] kernel and allows power deposition mapping in the full 3-D CAD geometry of the tokamak, taking as input user-defined specifications for parallel heat flow in the scrape-off layer (SOL) has been successfully developed. However, these field line tracing codes, including SMITER, are computationally demanding, restricting their application in a RT environment. Recently, a Matlab/Simulink software based control oriented heat flux monitoring system accounting for the effect of 3-D PFCs [17] was successfully developed for the ITER plasma control system simulation platform (PCSSP)[18, 19].

The generally low incidence angles of attack of magnetic field lines on PFCs in tokamaks, coupled with engineering tolerances, mean that the PFC front surfaces must usually be shaped to avoid the appearance of leading edges on which plasma would impact at near normal incidence. This is particularly important for the actively cooled components on ITER. As a result, a simple 2-D approach to a wall heat flux control algorithm, in which PFCs are assumed to be cylindrically symmetric, must be augmented to account properly for the 3-D shape which leads to distinct magnetically wetted areas and peak heat flux densities that can be considerably higher than the peak cylindrically symmetric values.

This paper describes the implementation and experimental demonstration of the RT FW heat flux algorithm developed for ITER on an operating device - the TCV tokamak [20].

Figure 1: (a) TCV limiter plasma boundary for the plasma discharge # 51399 used in the analysis. The HIR system field of view (FOV) is delimited by the red lines. (b) Deposited heat flux obtained from the HIR surface temperature measurements using the THEODOR code. The dark blue spots represents the areas where measurements are deficient as explained in the text.

2 Model-based power flux density descriptors

In the simplest case, the RT model-based approach 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 SOL power width, λ_q . These two parameters are used to constitute an exponentially decaying radial parallel heat flux profile $q_{||}(\psi)$ imposed at the last closed flux surface (LCFS) or separatrix on the outboard midplane (OMP), which is then magnetically mapped in the 2-D poloidal cross-section to the points of first wall (FW) contact. The model contains no transport physics and thus assumes that heat flow occurs as pure parallel convection, with the power density ultimately deposited at any given point on the FW surface a function only of the magnetic field line incidence and on the local flux expansion. Global power balance is used to determine P_{SOL} :

$$P_{SOL} = P_{\Omega} + P_{AUX} - P_{RAD} - \frac{dW}{dt} \quad (1)$$

where, P_{RAD} is the power radiated in the core plasma, P_{Ω} is the Ohmic power,

P_{AUX} the external heating power input and dW/dt is the rate of change of the stored energy. The radial power density profile in the SOL is given as follows:

$$q_{||,dR}(\psi) = q_{||,OMP} \exp\left(\frac{-(\psi - \psi_b)}{B_{\theta,OMP} R_{OMP} \lambda_q}\right), q_{||,OMP} = \frac{P_{SOL}}{2\pi R_{OMP} \lambda_q} \quad (2)$$

where, $q_{||,OMP}$ is the power density at the OMP. Taking into account the magnetic flux expansion, the power density perpendicular to the surface, q_{dep}^t is written as follows:

$$q_{dep}^t(\psi) = F \frac{P_{SOL}}{2\pi R_{OMP} \lambda_q} \frac{B_{\theta}^t}{B_{\theta,OMP}} \frac{\sin(\alpha)}{\sin(\zeta)} \exp\left(\frac{-(\psi - \psi_b)}{B_{\theta,OMP} R_{OMP} \lambda_q}\right) \quad (3)$$

where, the subscript 't' represents target, $F = 0.5$ represents the power flow fraction, assuming symmetric heat flux in the two directions of the field, ψ is the poloidal flux in Wb/rad , ψ_b is the poloidal flux at the LCFS/separatrix in Wb/rad , $B_{\theta,OMP}$ is the poloidal magnetic field at the OMP, R_{OMP} is the radial coordinate of the OMP, $\zeta = \tan^{-1}(B_{\theta}/B_{\phi})$ the field line pitch angle, α the angle between the field line and the physical surface and B_{ϕ} , B_{θ}^t are respectively the components of the field in the toroidal and poloidal directions at the point of calculation. q_{dep}^t is computed at the apex of the PFCs and the 2-D profile of the power density is assumed to be toroidally symmetric.

3 Experimental overview and analysis

The TCV device is a medium sized tokamak ($R_0 = 0.88$ m, $B_0 = 1.43$ T) with unique plasma shaping capabilities thanks to an array of 16 independently powered poloidal field coils and an open vacuum vessel structure [20], entirely covered with graphite protection tiles. On the central column (CC), these tiles have a very specific toroidal contour, shaped to avoid exposed edges, but optimized as far as possible to manage heat fluxes in both limiter and diverted configurations [21]. This is because TCV unlike many tokamaks, frequently uses the central column as a high field side divertor target as well as for inner wall limited plasma configurations. The TCV Matlab/Simulink based digital control system [22, 23, 24] and the deployment of a new, sub-ms, real-time magnetic equilibrium-reconstruction algorithm, RTLIUQE [25] positions the device as an ideal platform for testing the ITER wall heat flux estimation algorithm.

The performance of the control oriented model as an estimator for the power flux on the TCV inner wall is verified on a set of Ohmic limiter plasma discharges. The main diagnostic used in this experimental test is the horizontal (HIR) infrared (IR) camera [26], the field of view (FOV) of which is shown in figure 1(a), along with an example of the limiter configuration used in this study. Table 1 compiles the key important plasma parameters for the 3 discharges used. All data from the IR camera were obtained during the stationary phase of the discharge, $dW/dt = 0$ and $P_{AUX} = 0$.

Figure 2: (a) Estimated component of the deposited heat flux parallel to the magnetic field for the plasma discharge # 51399. (b) Sum of the background heat flux and the component of the deposited heat flux perpendicular to the magnetic field for the plasma discharge # 51399.

#	I_P (kA)	$\bar{n}_e$ ($\times 19 \text{ m}^{-3}$)	Z_{ax} (m)	κ	δ	P_{SOL} (kW)	P_{RAD} (kW)	P_{Ω} (kW)
51392	140	1.5	-0.05	1.35	0.0	127	35	162
51399	140	1.3	-0.05	1.17	0.0	155	35	190
51400	140	1.2	-0.05	1.09	0.0	136	33	169

Table 1: Plasma and shape parameters for the three limiter plasma discharges used in this study. κ and δ are the plasma elongation and triangularity respectively.

The heat flux distribution on the central column tiles, q_{dep} is computed from the IR images using the THEODOR finite difference inverse heat transfer code [27], with a procedure similar to that described in [28]. The outcome of the procedure is the temporal evolution of the 2-D map of the heat deposited onto the tiles $q_{dep}(R\phi, Z, t)$, where Z is the vertical coordinate, R , the major radius coordinate and ϕ , the toroidal angle. The deposited heat flux is averaged over time during the intervals where all plasma parameters listed in table 1 are kept constant. An example of the resulting $q_{dep}(R\phi, Z)$ 2-D map is shown in figure 1(b), where only the portion of heat loading above the plasma midplane is shown. There is a similar loading on the lower side of the tile, displaced toroidally to the left of a vertical axis corresponding to the tile centre. The up down heat loads will not generally be symmetric in a real experimental situation due to plasma transport effects. The analysis excludes several zones on the tiles (dark blue regions, figure 1(b)): the zone where the TCV port is imaged, the region shadowed

by the neighbouring tiles, the regions where the tile thickness is much smaller than the average tile thickness (THEODOR assumes a constant tile thickness) and the horizontal gap between the two tiles. The exclusion of these regions is also a cause of the under-estimation in the deposited power on the tile-surface.

Integration of the q_{dep} 2-D map yields the total power deposited on the CC, $P_{dep} = 32 \iint 2q_{dep}dR\phi dZ$, where the factor 2 accounts for heat deposition on the lower side of the limiter, assumed to be symmetric for simplicity, and 32 is the number of tiles making up a full toroidal CC ring. For the plasma discharge in Figure 1(a), P_{dep} , together with the total radiated power P_{RAD} obtained from bolometric measurements, accounts for 58% of the Ohmic power P_{Ω} . The missing power is lost through transport along field lines to other PFCs in the device beyond the strongest interaction points on the CC and may also be partially transferred to neutrals. Some underestimation is also due to certain regions of the experimental data being excluded (see below). In addition, the TCV bolometers underestimate the radiated power by 13%-15% [29].

Figure 3: Shadowing of TCV CC neighboring tiles, edited [21]. A close-up of the top view of two tiles on the CC is shown, together with a magnetic field line in the SOL (black line) and its first intersection with the tile (red dot).

For the sake of comparison between different plasma shapes and parameters, each point on the tiles is remapped onto magnetic coordinates, namely the OMP radial coordinate, $dR = R - R_{OMP}$ and the angle, α between the magnetic field and the plane tangent to the tile surface. The mapping uses the magnetic equilibrium reconstruction code LIUQE [30]. Following the procedure described in [31], the resulting 2-D distribution of the deposited heat flux, $q_{dep}(dR, \alpha)$ is then used to compute the perpendicular and parallel components of the heat

Figure 4: Parallel heat flux profile (red square), fitted with equation 5 (blue dotted line) for the plasma discharge # 51399.

flux. The deposited heat flux is modelled as the sum of heat flux components perpendicular, $q_{dep,\perp} = q_{\perp} \cos(\alpha)$ and parallel, $q_{dep,\parallel} = q_{\parallel} \sin(\alpha)$ to the magnetic field, and a background component, q_{BG} :

$$q_{dep}(dR, \alpha) = q_{\parallel}(dR) \sin(\alpha) + q_{\perp}(dR) \cos(\alpha) + q_{BG} \quad (4)$$

where $q_{\parallel}$ and $q_{\perp}$ is a function of R . The background results from a combination of different processes such as IR reflections and heating of the tiles by the radiation coming from the plasma. The parallel, $q_{\parallel}(dR)$, and perpendicular, $q_{\perp}(dR)$, heat flux radial profiles at the OMP are computed using the simplified method described in [31] for the determination of the wetted area and field line angles. The method locates the point of tangential incidence on the target tile and assumes a straight field line trajectory, until intersection with the neighbouring shadowing tile surface. It would be demonstrated later that this is not necessarily a good approximation.

The result of the analysis is shown in Figure 2, which breaks down the components of the total deposited heat flux due to thermal plasma parallel (Figure 2(a)) and perpendicular (Figure 2(b)) to the magnetic field. The sum of the deposited powers, together with P_{RAD} obtained from bolometric measurements, accounts for 62% of P_{Ω} . The sum of the deposited power exceeds the experimentally measured total deposited power since the analysis accounts for the heat flux deposition on the tiles that were excluded in the IR image analysis due to the reason explained earlier in the text. Furthermore, the power corresponding to the perpendicular component of the heat flux contributes 41% of the total deposited power on the tile surface. The reasons for this significant contribution of the perpendicular component at grazing incidence are discussed in detail in [31]. Since SMITER and the model-based approach assume only heat flow parallel to magnetic field lines, the code estimated tile surface heat flux distribution can be compared only with the parallel component of the experimentally measured heat flux (Figure 2(a)).

4 Power flux estimation on TCV using SMITER

Figure 5: (a-b) Parallel component of heat flux distribution derived from IR measurement and SMITER for the plasma discharge # 51399, along with the location of toroidal and poloidal cut used for comparing heat fluxes. (c-d) Comparison of poloidal and toroidal profile of the heat flux distribution estimated from the IR measurements and SMITER for the plasma discharge # 51399.

For the TCV magnetic equilibrium of Figure 1(a), assuming $P_{SOL} = P_{dep,\parallel}$ as the deposited power is carried by parallel heat flux and for a given specification of $q_{\parallel}(dR)$, SMITER is used to simulate the inner wall tile surface heat loads. Figure 3 illustrates the tile model used for the field line tracing. Experimental IR data are used to extract the OMP $q_{\parallel}(dR)$ profile according to the prescription described in [30] [31]. An example of the measured profile, together with coefficients obtained from fitting the data with a double exponential OMP radial heat flux profile are shown in Figure 4 for pulse # 51399. This form of the OMP heat flux profile has been the subject of intensive study in recent years, with a dedicated multi-machine effort [13] finding it to be universally present on all tokamaks running inner wall limiter plasmas, leading to the change of ITER first wall design. It is characterized by two heat flux channels, the ‘main-SOL’ component, $q_{\parallel,main}$, with e-folding length $\lambda_{q,main}$ and a ‘near-SOL’ heat flux channel, $q_{\parallel,near}$, featuring an e-folding length $\lambda_{q,near}$, but carrying an appreciable fraction of P_{SOL} [13]:

$$q_{\parallel}(dR) = q_{\parallel,main} + q_{\parallel,near} = q_{\parallel 0,main} e^{(-\frac{dR}{\lambda_{q,main}})} + q_{\parallel 0,near} e^{(-\frac{dR}{\lambda_{q,near}})} \quad (5)$$

with

$$R_q \equiv \frac{q_{\parallel 0,near}}{q_{\parallel 0,main}} \quad (6)$$

Applying this $q_{||}(dR)$ profile in SMITER for the magnetic equilibrium of Figure 1(a) gives the tile surface heat load distribution shown in Figure 5(b). The plots in Figures 5(c) and 5(d) show respectively poloidal and toroidal cuts in the computed distribution compared with the measured heat loads (due to the parallel component only) obtained from Figure 2(a), which is repeated in Figure 5(a). The discrepancy between the toroidal and poloidal heat flux profiles is expected and can be attributed to the different algorithms used in the determination of the shadowed regions and field line angles on the target tile surface. The shadowing algorithm in the IR analysis locates the point of tangential incidence on the target tile and assumes a straight field line trajectory, until intersection with the neighbouring shadowing tile surface. In contrast, SMITER solves the field line equation to trace the real trajectory of the field lines from the target.

5 Model-based power flux estimation for the TCV tokamak

Figure 6: Matlab/Simulink schematic for wall heat flux estimation and definition of input/output data for the application on TCV.

The Matlab/Simulink architecture of the ITER wall heat flux estimator [17] has been adapted to estimate the TCV inner wall tile power flux densities during high field side limiter plasma operation. The model is implemented on a dedicated core of a multi-core computational node of the TCV digital control

system [24, 32] and cycles at a sample time of 0.6 ms. The Matlab/Simulink implementation of the algorithm, accommodating various inputs and outputs is shown in figure 6, along with a description of the inputs/outputs used for the application to TCV.

Figure 7: (a) The three inner wall limited equilibria used in this study (see also Table 1). (b-d): Comparison of poloidal cuts of the deposited power flux density at the toroidal locations corresponding to the peak of the power density computed from SMITER and at the tile apex with the output of the 2-D model-based approach. (e) Evolution of the weight function with plasma elongation.

The real time magnetic equilibrium reconstruction algorithm, RTLIUQE provides the information at sub-ms time steps for the poloidal flux distribution, ψ_x , poloidal flux at the last closed flux surface or separatrix, ψ_b , radial coordinate of the OMP, R_{OMP} and vertical coordinate of the magnetic axis, Z_a . Additionally, the signals for the loop voltage, plasma current, the toroidal field, B_t , at the major radius, R_0 itself are available directly as RT signals inside the TCV digital control system. The P_Ω is approximated as the product of the loop voltage, V_{loop} times the plasma current, I_p , whereas due to the non-availability of the bolometric signals in RT, the power radiated in the core is approximated as a fraction of (typically, 20 %) of the P_Ω . The output of the algorithm is P_{SOL} , α and q_{dep} .

The model-based approach was applied to the set of dedicated Ohmic discharges listed in Table 1. For each plasma equilibrium, the magnitude of the peak heat flux from both SMITER and the model-based approach is determined, choosing an equilibrium from the stationary phase of the discharge flattop for the SMITER calculation. A weighting factor, W_F is then implemented into the model-based approach, defined as the ratio of the peak power flux density

obtained from SMITER and the model-based approach. Table 2 lists the parameters used to specify the OMP radial heat flux profile in SMITER derived from the experiment for the three discharges in Table 1

#	$P_{dep,\parallel}$ (kW)	$\lambda_{q,near}$ (mm)	$\lambda_{q,main}$ (mm)	R_q
51392	38	2.3	22	3.7
51399	48	2.8	30	3.0
51400	52	2.6	33	4.2

Table 2: Experimentally extracted parameters for the radial profile of parallel heat flux for the three discharges studied.

Figure 8: (a-f) Evolution of the various inputs required by Matlab/Simulink architecture for the plasma discharge # 51392. P_{Rad} is approximated as a fraction of the P_{Ω} (20 %).

The LCFS for the three equilibria studied with SMITER and the model-based approach are shown in figure 7(a). Figures 7 b-d compare poloidal power deposition profiles along the inner wall tile surfaces at the toroidal locations of the tile apex and at the peak of the power density computed with SMITER, which accounts for the full 3-D geometry of the problem. The profiles computed using the 2-D model-based approach match closely the SMITER values at the tile apexes, as expected, since this is the point at which the 2-D wall contour position used in the model is defined. At this point the effects of toroidal tile surface geometry are absent and the cylindrical symmetry assumed in the 2-D approach is exactly satisfied.

At the SMITER computed peak of the power deposition, there is a significant deviation from the 2-D approach, again as expected, since the toroidal curvature of the tile increases the field line impact angle. The actual value of the ratio between the 2-D and 3-D approaches depends on both the tile and magnetic geometry, but also sensitively on the assumed $q_{\parallel}(dR)$ profile (derived here directly from experiment). Figure 7e shows that the W_F is in the range 4.8-5.5 for the

3 discharges chosen here and this is essentially due to the small variation of plasma elongation and Ohmic power in the study. The average value of these weight factors is implemented into the RT algorithm to include the geometrical effect of the wall geometry.

The algorithm was used to simulate the heat flux distribution on the CC above the plasma midplane for pulse number # 51392. The specification of the OMP radial heat flux profile for the discharges is defined using the parameters listed in Table 2 derived from post-shot IR analysis. However, it is possible to have a RT estimation of the heat flux profile parameters [33] with the help of the multimachine scaling for inner wall limiter plasma [34]. Furthermore, the heat flux estimation is sensitive to the variation of λ_q [17]. For the case under consideration a systematic relative error of 20% in λ_q can result in a relative error of 19 % in the heat flux estimation. The time evolution of various inputs required for simulating the power flux density is shown in Figure 8(a-f). Figure 9 shows the evolution of the P_{SOL} , maximum value of the grazing angle and peak deposited power flux density on the inner wall CC. The perturbation at 0.76 s is a result of a step change in current in one of the poloidal field coils on the high field side as it changes its polarity.

Figure 9: Evolution of the various outputs of the Matlab/Simulink architecture for the plasma discharge # 51392. (a) Evolution of P_{SOL} during the discharge. (b) Evolution of the maximum field line angle at the apex of the CC tile. (c) Maximum deposited heat flux density evolution.

Figure 10(a) compares the steady state poloidal power deposition profile along the inner wall tile surfaces at the toroidal locations of the peak of the power

density derived using the IR measurement. The profiles computed using the 2-D model-based approach match closely the IR values. The shift in the poloidal location for the peak deposited power density is due to the evaluation of the profiles at different toroidal locations. Figure 10(b) shows a good agreement of the simulated peak power flux density with the experimental IR measurement. Additionally, the response of the algorithm with respect to a perturbation in the poloidal flux at $t = 0.76$ s is found to be satisfactory and a good tracking of the heat flux estimation is obtained with respect to the IR measurement.

Figure 10: (a) Comparison of a poloidal cut of the deposited power flux density at the toroidal location corresponding to the peak of the power density derived from the IR measurement with the output of the 2-D model-based approach applied at the apex of the tile at $T=0.6$ s. (b-c): Comparison of the evolution of the peak heat flux and its location in the poloidal plane from the IR measurement with the 2-D model based approach applied at the apex of the tile for the plasma discharge # 51392.

6 Conclusions

A successful experimental implementation of a wall heat flux estimator algorithm developed for ITER has been demonstrated on the TCV tokamak. The real-time Matlab/Simulink architecture has been successfully implemented in the TCV digital control system for estimating the heat flux distribution on the TCV tile surface. Good agreement with the IR diagnostic data is achieved with respect to estimation of the peak heat flux and its location in the poloidal plane

for limiter plasma discharges. Furthermore, the heat flux distribution on the 3-D geometry of the TCV tokamak determined using the SMITER has been successfully benchmarked against experimental IR diagnostic data.

The experimental validation of the model based approach accounting for 3-D effects of the plasma facing components on the TCV tokamak demonstrates the RT operational feasibility of the wall heat flux algorithm and sets the basic foundation for future development and its implementation in the heat load control system for the ITER plasma control system.

Disclaimer

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

This work was supported in part by the Swiss National Science Foundation.

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