Fusion Engineering and Design 137 (2018) 143–151

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Model-based real-time power flux estimator for the ITER first wall

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ARTICLE INFO

Keywords:
ITER
Plasma facing components
Real-time

ABSTRACT

The actively cooled technology used for the plasma facing components demands monitoring and control of heat flux on ITER first wall (FW) panels. Intense heat loads are predicted on the FW, even well before the burning plasma phase. Thus, a real-time (RT) FW heat load control system is mandatory from early plasma operation of the ITER tokamak. As a first step into this development, the paper presents a control oriented model, based on the RT equilibrium reconstruction for the ITER plasma control system. The paper discusses the model based approach and reports the Matlab/Simulink implementation of the algorithm. Key aspects of system integration and testing are reported, leading to the verification of the system in a RT environment.

1. Introduction

Baseline burning plasma operation in ITER targets the achievement of high fusion gain, $Q_{DT} = 10$, for durations in the range 300–500 s [1]. In such discharges, the thermal power crossing the plasma boundary corresponding to $\sim$100 MW is to be deposited on the plasma-facing components (PFCs). The resulting power flux densities can only be managed under stationary, long pulse conditions with active cooling, with water the chosen medium on ITER [2]. The use of water cooling, particularly in the nuclear environment of ITER, requires robust and reliable RT monitoring and control of PFC heat fluxes [3].

Various tokamaks have successfully demonstrated the capability of RT control to prevent the overheating of PFCs based on imaging diagnostics [4–7]. 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 the wide angle viewing system (WAVS) comprising visible (VIS) and infrared (IR) cameras [10]. The system views almost the entire first wall and divertor surface area and provides surface temperature measurements for hot spot monitoring, plus visible and IR images for physics analysis [3]. 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]. However, these codes are computationally demanding, restricting their application in a RT environment. This paper presents a control oriented model-based PFC heat flux monitoring system. The model has been implemented in the Matlab/Simulink environment and integrated into the ITER plasma control system simulation platform (PCSSP) [15,16].

Plasma current ramp-up in 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 [17,18]. Current scenario design aims at transitions to divertor configuration for plasma currents of order $I_p\bar{3}$. 5 MA after a duration > 10 s [19]. Depending on the achievable blanket alignment, limiter phase heat flux densities on the shaped FWP in the vicinity of plasma contact may approach the maximum design values [13] and hence the deposited heat flux must be monitored and carefully controlled. Fig. 1 (reproduced from [19]) compiles a series of magnetic equilibria derived from an example inner wall current ramp-up scenario using the DINA code [19].

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 MW/m^2 compared to that of the divertor target (10 MW/m^2). Unlike the inertially cooled PFCs common to many of today's devices, ITER's actively cooled Be FWPs cannot sustain the intense heat flux densities in the separatrix region for long before critical heat flux is reached at the cooling interface. Moreover, because high performance plasmas will be relatively tolerant to low Z Be influxes, plasma contamination itself is unlikely to be an effective manner with which to avoid off-normal situations.

Fig. 2(a) shows a portion of the ITER first wall, illustrating the modular FWPs attached to massive stainless steel shield blocks.

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https://doi.org/10.1016/j.fusengdes.2018.08.019
Received 5 June 2018; Received in revised form 30 July 2018; Accepted 30 August 2018

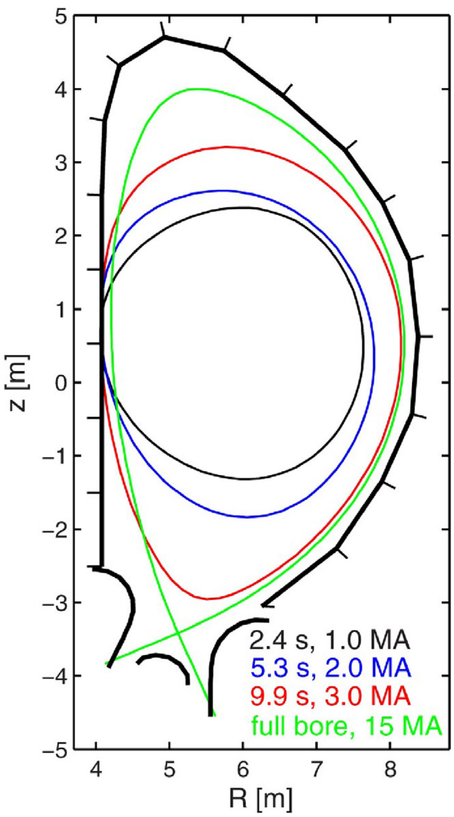

Fig. 1. Example of ITER start-up on the inner wall obtained from a DINA code full scenario simulation [19].

Illustrated in Fig. 2(b) is a FWP comprising a double-winged structure in the toroidal direction symmetrically disposed about a central, poloidally running slot which provides space for mechanical and hydraulic connections as well as for various plasma diagnostic systems [20,21]. The physics heat load specifications for the FW and divertor [18] has led to a requirement for different power handling capabilities at different regions of the poloidal cross-section. As a result, the FWPs are categorised as: a 'normal heat flux' (NHF) technology (up to 2 MW/m^2) and an 'enhanced heat flux' (EHF) technology (up to 4.7 MW/m^2). Fig. 2(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 [20,21].

For a given magnetic equilibrium, power crossing the plasma boundary, P_{SOL} and width of the scrape-off layer (SOL), λ_q, for heat flux density, $q_{||}(r)$ flowing parallel to the magnetic field lines, the heat flux density deposited on the PFCs can be obtained by imposing 0-D power balance at the outer midplane (omp) and constructing an exponential radial profile of $q_{||}(R)$ [18] as follows:

$$q_{||}(R) = q_{||,\mathrm{omp}} \exp(-(R - R_{\mathrm{omp}})/\lambda_q) \tag{1}$$

$$q_{||,\mathrm{omp}} = \frac{P_{\mathrm{SOL}}}{4\pi R_{\mathrm{omp}}\lambda_q}\left(\frac{B_\phi}{B_\theta}\right)_{\mathrm{omp}} \tag{2}$$

where $q_{||,omp}$ is the parallel heat flux at the separatrix or last closed flux surface (LCFS), R_{omp} is the radial coordinate and $(B_\phi/B_\theta)_{\mathrm{omp}}$ is the ratio of the toroidal to poloidal magnetic field at the omp and R is the major radius. 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, the pitch of the field line and on the local flux expansion.

Magnetic field line tracing codes [11,12] determine the heat flux density on the PFC by 3-D field line tracing within a given magnetic equilibrium to compute the plasma wetted area, accounting for shadowing of neighbouring components (including self-shadowing) [18,13]. Eqs. (1) and (2) are used to specify the parallel heat flux density, which is then projected onto the PFC surface over the wetted area. The difficulty lies in the efficiency of the intersection calculation, since a rather fine meshing of target components is often required and a number of neighbouring objects must generally be included. In view of the present computational capabilities, the intersection algorithm is computationally expensive and cannot yet be practically included in a RT environment. Concerning the simple model used to specify the heat flux, increased levels of complexity could of course be included, though the justification to do so is not strong for main chamber plasma interactions where, especially in diverted configurations, there is little or no dissipation of the parallel heat flow in the separatrix region where the flux densities are most intense.

The paper reports on a control oriented model, based on RT equilibrium reconstruction for estimating the heat flux on the ITER PFCs. The model-based approach follows some aspects of the WALLS system developed for JET [8,9] and, in the simplest case, describes the heat flux deposited on PFCs as a poloidal flux function with two free parameters: P_{SOL} and λ_q.

2. Model-based power flux density descriptors

'Global power balance' is used to determine P_{SOL}:

$$P_{\mathrm{SOL}} = P_{\mathrm{Ohmic}} + P_{\mathrm{Aux}} - P_{\mathrm{Rad}} - \mathrm{dW/dt} \tag{3}$$

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, $q_\perp$ perpendicular to the surface is as follows [22],

$$q_\perp(\psi) = \underbrace{F}_{\mathrm{Term1}} \underbrace{\frac{P_{\mathrm{SOL}}}{2\pi R_{\mathrm{omp}}\lambda_q}}_{\mathrm{Term2}} \underbrace{\frac{B_\theta}{B_{\theta,\mathrm{omp}}}}_{\mathrm{Term3}} \underbrace{\frac{\sin(\eta)}{\sin(\zeta)}}_{\mathrm{Term4}} \underbrace{\exp\left(-\frac{\psi - \psi_b}{R_{\mathrm{omp}}B_{\theta,\mathrm{omp}}\lambda_q}\right)}_{\mathrm{Term5}} \tag{4}$$

where ψ is the poloidal flux, ψ_b is poloidal flux at the LCFS/separatrix, $B_{\theta,omp}$ is the poloidal magnetic field at the omp, B_θ is the poloidal magnetic field at the point of calculation, $\zeta = \arctan\frac{B_\theta}{B_\phi}$ is the field line pitch angle, η is the angle between the field line and physical surface and B_ϕ is the component of the field in the toroidal direction:

$$B_\phi = B_0\frac{R_0}{R} \tag{5}$$

with B_0, the magnetic field at the major radius, R_0.

Term 1 in Eq. (4) represents the power flow fraction, $F = 0.5$, assuming symmetric heat flux in the two directions of the field, Term 2 denotes the power density at the omp, translated into the power density local to the tile by the Term 3, accounting for flux expansion. The Term 4 takes care of the incident angle and the pitch of the magnetic field lines. Term 5 describes the exponential decay of the power density across the magnetic flux surfaces. Term 3, Term 4 and the denominator of Term 2 can be combined together to define the wetted area, A_w:

$$A_w = \frac{2\pi R_{\mathrm{omp}}\lambda_q B_{\theta,\mathrm{omp}}\sin(\eta)}{B_\phi} \tag{6}$$

where the term $2\pi R_{omp}\lambda_q B_{\theta,omp}$ is constant along a magnetic field line.

The total power, $P(\psi_1, \psi_2)$ on a toroidally continuous ring with a surface of finite length between two points with total fluxes ψ_1 and ψ_2, is expressed as follows:

$$P(\psi_1, \psi_2) = F[P(\psi_2) - P(\psi_1)] \tag{7}$$

$$P(\psi) = P_{\mathrm{SOL}}\left[1 - \exp\left(-\frac{\psi - \psi_b}{R_{\mathrm{omp}}B_{\theta,\mathrm{omp}}\lambda_q}\right)\right] \tag{8}$$

A reduction in the distance between the plasma boundary and various locations on the FW is linked to an increase in the deposited