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Simulations of ITER start-up and assessment of limiter power loads

G. Federici [a,*], O. Zolotukhin [a], M. Kobayashi [b], A. Loarte [c], G. Strohmayer [a], A. Tanga [a], A. Portone [c], L. Horton [d], Y. Feng [e], F. Sardei [e], Y. Gribov [f], M. Shimada [f], A. Polevoi [f], R. Mitteau [g], C. Lowry [a]

[a] ITER JWS Garching Co-center, Boltzmannstrasse 2, 85748 Garching, Germany
[b] National Institute for Fusion Science, 322-6 Oroshi, Toki 509-5292, Japan
[c] EFDA-CSU Garching, Boltzmannstrasse 2, 85748 Garching, Germany
[d] Max-Planck-Institut für Plasmaphysik, Boltzmannstrasse 2, 85748 Garching, Germany
[e] Max-Planck-Institut für Plasmaphysik, Teilinstitut Greifswald, 17491 Greifswald, Germany
[f] ITER JWS Naka Co-center, 801-1 Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken, Japan
[g] Association Euratom CEA–Cadarache, Saint Paul Lez Durances cedex 13108, France

Abstract

This paper presents the results of a modelling study conducted to estimate the power crossing the separatrix (P_{SOL}) in the ITER device during a standard start-up sequence. This is used to calculate the power intercepted by the start-up limiters and the resulting power load distribution. The models and methodologies applied to calculate P_{SOL} and the power loads on the limiters are described in detail elsewhere ([e.g., M. Kobayashi et al., Nucl. Fusion. 47 (2) (2007) 61]) and only a brief mention of some of the main results is included here. These assessments show that for the range of conditions analysed, the maximum P_{SOL} intercepted by the two ITER limiter start-up modules during the current ramp-phase is ~6 MW. The peak power load to each limiter is calculated to be ~5 MW/m^2, but these values depends on assumptions on physical quantities (e.g., transport coefficients, i.e., $D_\perp$ and $\chi_\perp$), which are uncertain and still await confirmation by experiments. Recommendations are made for modelling and experiments to extend the study presented here.

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Keywords: ITER; Beryllium; Limiter; Simulation; Power deposition

1. Introduction

The analysis of plasma current ramp-up and ramp-down phases in ITER is an important part of the design of the start-up limiters. Among other things, good accuracy is required to estimate the

* Corresponding author. Tel.: +49 89 32994228; fax: +49 89 32994110.
E-mail address: gianfranco.federici@iter.org (G. Federici).

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power crossing the separatrix (P_{SOL}) and the power load distribution on the limiter surfaces throughout the limiter phase of the discharge.

This paper summarizes mainly the results of a study conducted with the plasma transport code ASTRA [1] to calculate the power intercepted by the start-up limiters, and briefly mentions some typical results of an accompanying modelling effort by Kobayashi et al. [2] to calculate the resulting power load distribution at the limiter surface using the code EMC3–Eirene [3,4]. The governing equations and the methodology of the model applied here to calculate P_{SOL} are discussed in [5], together with the results of the underlying validation runs done against a set of well diagnosed standard Ohmic current ramp discharges in JET [6] and ASDEX Upgrade. In particular, the selection/calibration of some of the most important parameters and physics assumptions used in the simulation (i.e., global energy confinement time, impurity-line radiation) was made based on obtaining reasonable agreement with experimental data.

The present analysis should be viewed as more reliable for indicating trends, rather than providing firm quantitative predictions. There are still significant uncertainties and the study presented here should be extended in several ways. Recommendations are made on areas that need improvements.

A brief description of the design of the ITER limiter system is included in Section 2. The model and the simulation conditions are discussed in Section 3. The results of the analyses are discussed in Section 4. Finally, the main findings are summarized in Section 5 together with recommendations for further work.

2. Limiter design and start-up conditions

The current design of the ITER start-up limiter system consists of two modules, which protrude of

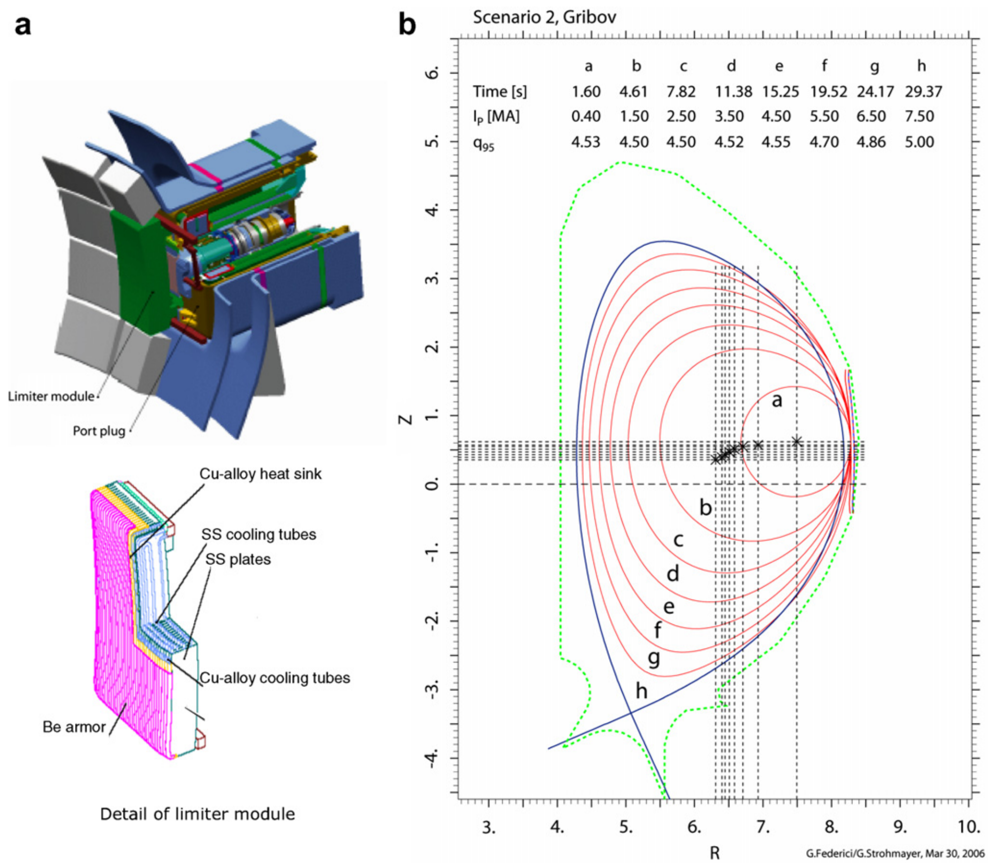

Fig. 1. (a) Limiter module located inside the equatorial port of the vacuum vessel. (b) Evolution of the plasma boundary during the current ramp-up for reference start-up scenario 2 [11].

few centimeters from the rest of the first wall and are located inside two opposite ports, at the equatorial level. These two limiters can be misaligned relative to each other by 1 mm. The function of these start-up limiters is to define/control the plasma boundary during the initial phase of and the final phase of each discharge. For the *reference* start-up sequence, described in [7], the plasma remains limited for about ~30 s (up to $I_p \sim 7$ MA), until an X-point is formed and a divertor plasma is established. The time evolution of the plasma boundary during the current ramp-up for this sequence is shown in Fig. 1(b) together with values of I_P and q.

These vertical limiter modules must fit through the horizontal ports since they are to be installed in the tokamak after the vacuum vessel is closed, and must be easily replaced. In general, the main advantages of this design are simplicity, flexibility and easy maintenance. The major disadvantage is that the contact area with the plasma is limited, which results in large heat loads unless the input power is small. The details of the engineering design for the present ITER limiter system (see Fig. 1(a)) can be found elsewhere (e.g., [8–10]) and only aspects influencing the power handling capability of the system are discussed here.

Each of the two port limiter modules are 2.1 m high, 1.65 m wide, comprising a plasma-facing part and stainless steel shield plates, and a structure with a plug shield, alignment and supporting system and attachments for the limiter module. The front part of the limiter consists of 4 mm beryllium armour in the form of small tiles attached to a Cu-alloy substrate plate internally cooled by water flowing in Cu-alloy tubes. A design solution is being explored [10], for fast retraction of the limiter during each shot and aligning it to the rest of the first wall.

3. Model description

The plasma transport code ASTRA [1] was used to estimate the power flux crossing the separatrix (P_{SOL}). Details of the model and underlying assumptions are discussed in detail in [5,11] (see also Table 1). Here, we only briefly discuss some of the assumptions, which were found to affect the results.

- *Global energy confinement time during current ramp*: several calibration runs were performed using different correlations for the global energy confinement time during current ramp to determine the best fit with experimental data (e.g., neo-Alcator scaling [12], and L-mode scaling [13], and a combination of the two according to the formulation recommended by Shimomura–Odajima [14] in cases with additional heating was tested). Pure Ohmic scaling, like neo-Alcator, is known to describe well the plasma confinement for small- and medium-sized tokamaks. However, for larger machines with high plasma

Table 1
Summary of main assumptions used for the ITER simulation (see also text)

Parameters	Assumptions	Comments/References
Average electron density	$\langle n_e \rangle = 0.2$–$0.5 n_G$	Scans
Fuel density	$n_D = n_T$	
Plasma resistivity		Neo-classical correction to the resistivity as given by Hirsman et al. [21] is used
Electron heat conductivity	$\chi_e^{anom} = \pi a^2 \kappa / 18\tau^*$	The electron transport is assumed to be neoclassical plus an anomalous term, while for the ions, only a neoclassical term is considered
Ion heat conductivity	$\chi_i^{anom} \approx 0$	The ion transport is assumed to be only neoclassical
Plasma confinement time	$\tau^* = \tau_{L-89}$	See text
Particle diffusion coefficient	$D_e = 0.2\chi_e$	
Ware pinch	$v = 0$	
Radiation losses	Bremsstrahlung + syncrotron + impurity line radiation	See text
Edge density $n_{e,\,b} = 0.2\langle n_e \rangle$	$n_{e,\,b} = 0.2\langle n_e \rangle$	
Edge temperature	$T_{e,b} = T_{i,b} = 50$ eV	

Input: $I_p(t)$, $a(t)$, $n_Z(0)$, $\kappa(t)$, $\Psi(t)$, $\vec{B}(t)$, $P_{aux}(t)$; given fraction of impurity species k in the plasma, ($f_k = n_z^k/n_e$). For the calculations discussed in this report, the concentration of impurities has been varied to have Z_{eff} in the range 2–3. $Z_{eff} = 2 \rightarrow f_{Be} = 2\%$, $f_C = 3\%$; $Z_{eff} = 3 \rightarrow f_{Be} = 2\%$, $f_C = 6\%$. *Output*: n_e, n_i, T_e, T_i, P_Ω, P_{RAD}, P_{lim}.
$n_G = I_p/\pi a^2$ Greenwald density.

temperature even during ramp-up (e.g., JET and probably ITER), the plasma is likely to be better described by a saturated Ohmic confinement law (SOC), which for JET is well fitted by L-mode scaling [13]. Thus, we set the electron anomalous transport in such a way that energy confinement time matches L89-mode scaling.

- *Radiation and impurities*: contrary to most of the simulations reported in the past (e.g., [15,16]), we assume that impurity ions are not fully stripped during start-up and that line-radiation plays an important role. This has substantial implications on the results presented here. The impurity model used is the basic corona model. The effect of the finite residence time of impurities, which enhances radiation compared to the corona model, was also taken in account by using the scaling proposed by Behringer for limiter conditions for JET and ASDEX Upgrade [17,18].

4. Results of the calculations and discussion

The results of these runs are compared in Fig. 2, where the time evolution of the density, the ohmic heating power P_{Ω}, radiation power P_{RAD}, total power to the limiter P_{lim}, P_{RAD}/P_{TOT}, where $P_{tot} = P_{\Omega} + P_{aux}$. To avoid computational problems the breakdown phase was not modelled, and we begin the ASTRA simulations at $t = 4.5$ s when the fast current rise ends and the plasma current is ~1.5 MA. Due to this reason all curves in Fig. 2 start at $t = 4.5$ s.

For the simulation of the ITER start-up sequence, we found that for the current ramp-up at relatively low plasma density (e.g., $<1 \times 10^{19}$ m^{-3}, which is about a factor of 2 lower than in current experiments), the power to the limiter remains relatively low (~3 MW), but there remain concerns with regard to sustainability of stable discharge conditions (e.g., likely onset of plasma instabilities due to slideway/runaway effects, locked-modes, etc., and Be runaway erosion, etc). The increased instability of the plasma column in the current ramp-up is likely to pose further limitations, apart from an acceptably safe trajectory in the empirical l_i–q diagram. All this requires more detailed assessments.

A sensitivity study with respect to density, impurity concentration and the level of auxiliary heating was conducted to explore possible ranges of operating parameters and viable window of operation. The parameters and the results of the density and impurity concentration scans are shown in Table 2 and

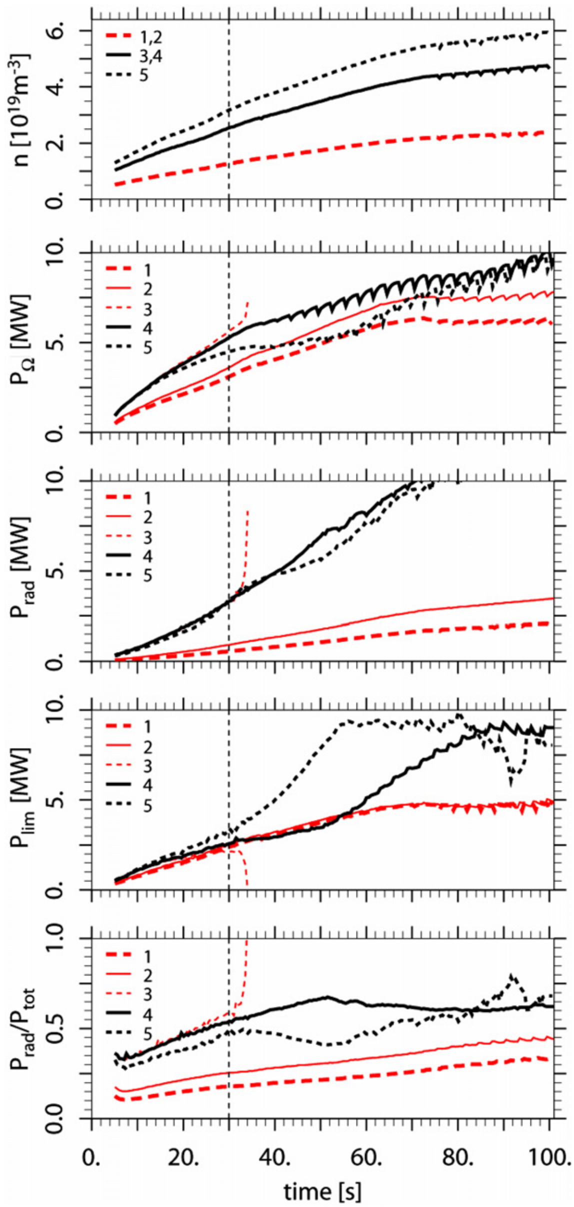

Fig. 2. Time evolution of n, P_{Ω}, P_{RAD}, P_{RAD}/P_{TOT}, P_{lim}, for all cases analysed. The numbers correspond to the cases listed in Table 2.

Fig. 2. In particular, impurity line radiation was found to play an important role because of the very large surface of the ITER plasma. For example, P_{RAD} in ITER can be easily calculated to be of the order of 5 MW, while in JET is of the order of 1 MW. On the contrary, the current density in ITER is ~1/4 of that of JET for the same current ($a_{ITER} \approx 2a_{JET}$) and as a consequence, the ohmic heating is lower, typically a half of that of JET. This

Table 2
Parameter scan for ITER limiter start-up simulations

Case	n (I)	Z_{eff} (I)	$t_{collapse}$ (s) (C)	P_{aux}* (MW) (I)	P_{Ω} max* (MW) (C)	P_{RAD} max (MW) (C)	P_L max (MW)
1	0.2 n_G	2	$>t_{ramp}$	0	3.1	0.5	2.4
2	0.2 n_G	3	$>t_{ramp}$	0	3.6	0.9	2.5
3	0.4 n_G	3	$\leqslant 30$	0	5.6	3.3	2.1
4	0.4 n_G	3	$>t_{ramp}$	$\leqslant 10$	5.2	3.2	2.5
5	0.5 n_G	2	$>t_{ramp}$	$\leqslant 10$	4.5	3.3	3.3

(I) input to simulation, (C) calculated, *end of limiter phase assumed here to occur at the plasma current of ~7 MA or before the plasma collapse. For cases 4 and 5, auxiliary heating is applied at beginning of current ramp.

is why in ITER during a purely Ohmic ramp-up at high densities (e.g., 0.4–0.5n_G, where n_G is the Greenwald density) the source of heating may not be enough to offset radiative losses (e.g., see case 3 in Table 2).

To avoid the onset of radiative collapse one has to apply very early in the discharge additional heating to the electrons. To preliminarily investigate this effect we added an additional heating power limited to 10 MW (cases 4–5 in Table 2) during the ramp-up and continuing throughout the reminder of the discharge. A scaling of the auxiliary power was selected to minimize the level of the additional power needed to sustain a discharge (as the ramp-up is assumed to be done at constant n/n_G, there is a need to compensate for the strong increase of radiation with density n^2 during the discharge).

The power to the limiter for the cases analysed could increase up to ~3.3 MW. None the less, given the remaining uncertainties, to be clarified by further work, it was deemed prudent to assume a factor of 2 uncertainty in the present estimates. In this case the upper bound of P_{lim} should be about 6 MW.

It should be noted that for the calculations shown here we assume, for the sake of simplicity, that no radial misalignment exists between the two limiters. This is a reasonable assumption, because for typical power scrape-off-width expected for ITER during current ramp-up (e.g., 10 mm < λ_q < 20 mm) a design misalignment tolerance of ±1 mm would lead to an increase of the total power on the most protruding limiter which would less than 10%. A shift of the limiters significantly larger than 1 mm would lead to a larger increase of the power. This is not addressed in the paper and requires further study.

Earlier parametric calculations performed by Pacher [19], which were based on a simple diffusion model and assumed a constant power e-folding length function of the square root of the average connection length for that particular limiter configuration, led to the conclusion that the peak power load on the limiter surface did not exceed the design value, as long as the power decay width was above 1 cm and P_{SOL} is limited. Recently, a more accurate assessment was conducted using the 3D plasma

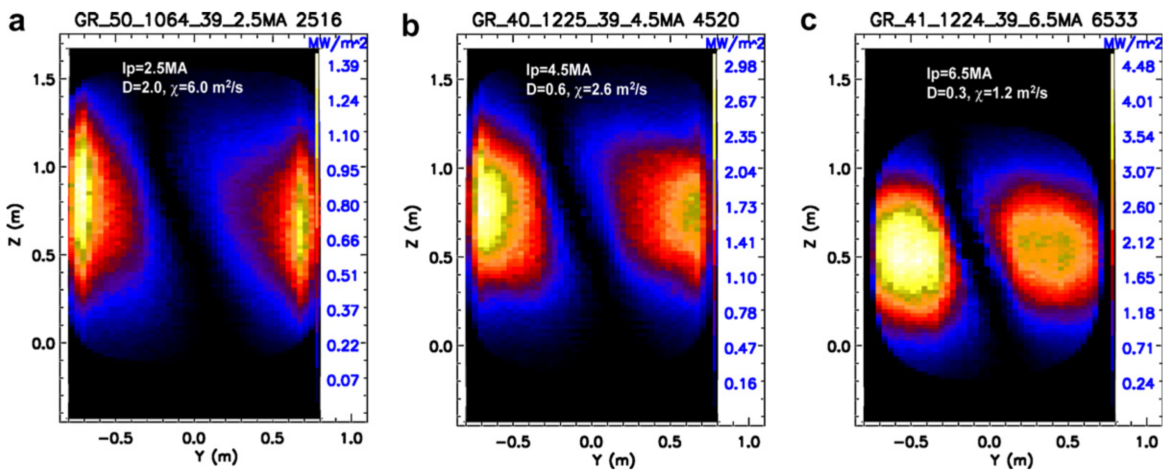

Fig. 3. Power deposition profiles vs. Y and Z on limiter calculated with the model discussed here with asymmetry 1:1 for (a) 2.5 MA; (b) 4.5 MA; and (c) 6.5 MA for a total power to the limiter SOL of 6 MW assumed to be equally distributed between 2 limiters [2].

transport code EMC3–Eirene [2] and the peak power load to each limiter was found to be of the order of 5 MW/m^2 at the end of the limiter contact phase for a total assumed power flowing in the SOL of 6 MW and a cross-transport diffusion coefficient of 0.4 m^2/s (see Fig. 3). Details of this work are discussed elsewhere (e.g., see [2,20]). The calculated peak heat flux is below the currently assumed design limit value, ($\sim$8 MW/m^2), based primarily on thermal fatigue considerations [8]. However, further design work is ongoing to re-examine the maximum design heat flux and to determine the available engineering margins for this or alternative solutions, as well as to identify conditions under which the power handling could become marginal.

5. Summary and recommendations for further work

We describe in this paper the results of a study conducted to estimate the power intercepted by the two ITER start-up limiter modules for the principal plasma start-up scenario and the resulting power loads.

For the range of conditions analysed, the maximum P_{SOL} intercepted by the two ITER limiter start-up modules during the current ramp-phase is less than $\sim$6 MW. The peak power load to each limiter is calculated to be $\sim$5 MW/m^2, but these values depends on assumptions on physical quantities (e.g., transport coefficients, i.e., $D_\perp$ and $\chi_\perp$ used in the models), which are uncertain and still await confirmation by experiments.

The present analysis should be viewed as more reliable for indicating trends, rather than providing firm quantitative predictions. In particular, the study presented here should be extended in several ways. Some of the modelling/design areas requiring improvements are: (i) modelling of impurity transport; (ii) determination of a proper stability criterion to set an upper limit for $P_{\mathrm{rad}}/P_{\mathrm{tot}}$ during the current ramp-up; (iii) compatibility of recommended level of auxiliary heating during the current ramp-up with available heating systems, hydrogen operation and limiter power handling; (iv) analysis of the potential implications of a reduced current ramp-up rate, which may arise from a plasma heated during the ramp-up phase in limiter configuration; (v) investigation of the margins and compatibility of early X-point formation; (vi) analysis of other plasma scenarios including ramp-down.

At the same time, further experimental information is required to characterize: (i) the SOL parameters, e.g., density and temperature (separatrix values and decay lengths); (ii) ohmic heating and radiated power proportion, plasma energy confinement and plasma control (iii) impurity production and plasma contamination (Z_{eff}); (iv) upper and lower operation density ranges during limiter start-up, and ramp down in present tokamaks, to compare them with the values during the flat-top, and with modeling.

All this would give important data for the design of the ITER start-up limiters.

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