Simulations with COREDIV Code of DEMO Discharges

R. I. Ivanova-Stanik, R. Zagórski and R. Stankiewicz

1Institute of Plasma Physics and Laser Microfusion, EURATOM Association, Hery 23, 01-497 Warsaw, Poland

Corresponding Author: roman.zagorski@ipplm.pl

Abstract:
The reduction of divertor target power load due to radiation of sputtered and externally seeded impurities in fusion reactor is investigated in this paper. The approach is based on integrated numerical modelling of DEMO discharges using the COREDIV code, which self-consistently solves 1D radial transport equations of plasma and impurities in the core region and 2D multifluid transport in the SOL. Calculations have been performed for inductive DEMO scenarios and for DEMO steady-state configurations with tungsten walls and Ar, Ne seeding. For all considered DEMO scenarios significant fusion power can be achieved. Increase of seeded impurity influx leads to the reduction of fusion power and Q-factor due to plasma dilution. Total radiation appears to be almost independent on puffing level and is dominated by core radiation (> 90%). The radiation due to seeding impurity is small and the results are weakly affected by seeded impurity type. For pulsed DEMO concepts, accessible seeding level is limited. There is no steady state solution for stronger puffing. The solution terminates due to helium accumulation, if confirmed by more detailed investigations, might strongly affect DEMO design.

1 Introduction

The long term aim of the EU fusion program is the development of commercial electricity-generating fusion power plant. For this goal to be achieved, the Power Plant Physics and Technology Department was established under EFDA with the main objective to develop tokamak based Fusion Power Plant conceptual design (DEMO), integrating the technology and physics aspects for optimal performance by addressing the key issues in physics and technology. A limited number of baseline DEMO design concepts with various degrees of extrapolation from today’s known underlying physics and engineering basis is currently being explored[1]. It appears that one of the most crucial and challenging issues of the fusion power plant is the development of reactor scenarios which satisfy simultaneously the requirement of sufficiently high power amplification with the needs for sustainable power exhaust. Independent of the plasma facing materials in DEMO reactor, the radiative exhaust of energy by sputtered and by externally seeded impurities is considered as possible way of spreading energy over large wall area. Radiating scenarios
have been already successfully developed for present day tokamaks, like ASDEX\cite{ref2} and for JET ITER Like-Wall (ILW) configuration\cite{ref3}, however it has to be checked if similar approach is applicable for burning plasmas, self-heated by $\alpha$-particles and with tungsten as the wall material.

This paper describes integrated numerical modelling for the first time applied to DEMO discharges with tungsten wall and argon/neon seeding using the COREDIV code\cite{ref4,ref5}, which self-consistently solves 1D radial energy and particle transport equations for plasma and impurities in the core region and 2D multifluid transport in the SOL. The model is fully self-consistent with respect to both the effects of impurities on the $\alpha$-power level and the interaction between seeded and intrinsic impurities. This interaction leads to a significant change in the intrinsic impurity fluxes and energy power balance, and it is found to be essential for a correct evaluation of the average power to the target plates. The code has been successfully benchmarked with a number of JET discharges, including the nitrogen seeded type I and type III ELMy H-mode discharges\cite{ref6,ref7} and recently, JET ILW configuration\cite{ref8}. It has been also applied to ASDEX discharges in the full W environment\cite{ref9}.

2 Model

Since the energy balance in tokamks with metallic walls depends strongly on the coupling between the bulk and the scrape-off layer plasma, modeling requires the transport problem to be addressed in both regions simultaneously. The physical model used in the COREDIV code is based on a self-consistent coupling of the radial transport in the core to the 2D multifluid description of the scrape-off layer. Since the model is relatively complex \cite{ref7,ref10} we point out here only the most important aspects of the model.

Core region

In the core, the 1D radial transport equations for bulk ions ($n_i$), for each ionization state of impurity ions ($n_{j,k}$) and for the electron ($T_e$) and common ion temperature ($T_i$) are solved. In particular, the following equations are solved:

\begin{align}
\frac{\partial n_i}{\partial t} + \frac{1}{r g_1} \frac{\partial}{\partial r} [r g_2 (-D_i \frac{\partial n_i}{\partial r} + W_i n_i)] &= S_i(r) \tag{1} \\
\frac{\partial n_{j,k}}{\partial t} + \frac{1}{r g_1} \frac{\partial}{\partial r} [r g_2 \Gamma_{j,k}] &= n_e [n_{j-1,k} \alpha_{j-1,k} - n_{j,k} (\alpha_j \chi_{j,k} + \beta^{j+1}_{j,k}) + n_{j+1,k} \beta_{j,k}^{j+1}] \tag{2} \\
\Gamma_{j,k} &= (-D_{j,k} \frac{\partial n_{j,k}}{\partial r} + W_{j,k} n_{j,k}) \tag{3} \\
3 \frac{\partial n_i T_i}{\partial t} + \frac{1}{r g_1} \frac{\partial}{\partial r} [r g_2 (-n_i \chi_i \frac{\partial T_i}{\partial r} + \frac{5}{2} \Gamma_{j,k} T_i)] &= P_{aux}^i + Q_{ei} \tag{4} \\
3 \frac{\partial n_e T_e}{\partial t} + \frac{1}{r g_1} \frac{\partial}{\partial r} [r g_2 (-n_e \chi_e \frac{\partial T_e}{\partial r} + \frac{5}{2} \Gamma_{j,k} T_e)] &= P_{OH} + P_{aux}^e + P_{a} - P_{B} - P_{synch} - P_{lin} - P_{ion} - Q_{ei} \tag{5}
\end{align}
where $j = 1, \ldots Z_k$ corresponds to ionization stages of impurity type $k$, with $Z_k$ being atomic number, $r$ is the radial coordinate and $g_1 = V'$ and $g_2 = |\nabla \rho|^2$ are metric coefficients. Densities of main plasma ions ($n_i$) and impurity ions ($n_k^j$) are given by the solution of the above radial diffusion equations with diffusion coefficients $D_i = D_j^k = 0.35 \chi_e$ and small anomalous pinch term $W_i = W_j^k$ the same for all species. Electron density $n_e$ is calculated from the quasi neutrality condition, whereas the ambipolarity condition defines the electron flux $\Gamma_e$.

$P_{\text{aux}}, P_{\text{OH}}, P_\alpha$ are the auxiliary, ohmic and the alpha heating powers, respectively. The energy losses are determined by bremsstrahlung ($P_B$), synchrotron radiation ($P_{\text{synch}}$), line radiation ($P_{\text{lin}}$) and ionization losses ($P_{\text{ion}}$). $Q_{\text{e}}$ is the collisional energy exchange between electrons and ions, and $\alpha_{\text{ion}}, \beta_{\text{rec}}$ are ionization and recombination rate coefficients, respectively. With respect to the energy sources, alpha heating is calculated self-consistently with the plasma dynamics, whereas for auxiliary heating parabolic-like deposition profile is assumed. The equation for the poloidal magnetic field has been neglected and thus the current distribution is assumed to be given in our approach. The electron and ion energy fluxes are defined by the local transport model proposed in Ref.[11] which reproduces prescribed energy confinement law. Similar model has been used for the simulations of the JET and ASDEX experiments [7, 9]. In particular, the anomalous heat conductivity is given by the expression

$$\chi_{e,i} = C_{e,i} a^2 \frac{\tau_E}{\tau_{E}} \times F(r)$$

where $a$ is the plasma radius, $\tau_E$ is the energy confinement time defined by the ELMy H-mode scaling law (IPB98(y,2))[12] and the coefficient ($C_e = C_i$) is adjusted to have agreement between calculated and experimental confinement times. The parabolic like profile function $F(r)$ is modified to provide transport barrier in the plasma edge with the barrier width of $\sim 10$ cm which results in the plasma temperature at the pedestal of the order of 4-8 keV. The source term takes into account the attenuation of the neutral density due to ionization processes: $S_i(r) = S_{i0} \exp(-\frac{a-r}{\lambda_{\text{ion}}})$, where $\lambda_{\text{ion}}$ is the penetration length of the neutrals, calculated self-consistently. The source intensity $S_{i0}$ is determined by the internal iteration procedure in such a way that the average electron density <$n_e>$ obtained from neutrality condition is fixed (input parameter). In the present simulations, the radial impurity transport is described only by anomalous contribution with small pinch term $W_j \propto \tau_E^2 D_i r/a^2$. That corresponds to the experimental situation with the ECRH control of the central $T_e$ [2] leading to mitigation of W accumulation in the core.

Scrape-off layer (SOL) plasma

In the SOL, the 2D boundary layer code EPIT is used, which is primarily based on Braginskij-like equations for the DT plasma[13] and on rate equations for each ionization state of each impurity species. In the past, the EPIT code was used to analyze FTU experiments resulting in satisfactory agreement between calculated and measured quantities[14]. For the sake of simplicity, the drifts are not included in the model. However, the influence of drifts on the ignition condition and on the global balance of a high density thermonuclear plasma is expected to be relatively small. The effect of the drifts would be concentrated in the SOL and in the boundary region close to the separatrix,
where - due to their interrelation with the radial electric field and its shear - they might influence transport significantly. These implications are, however, outside the scope of our work. Since the SOL model is fully described elsewhere\[15\], only the main points are briefly discussed in the following. For every ion species the continuity, the parallel momentum and two energy equations (for $T_e$ and common ion temperature $T_i$) are solved. An analytical description of the neutrals is employed based on a simple diffusive model. It takes into account the plasma (deuterium and seeded impurities) recycling in the divertor as well as the sputtering processes at the target plates including deuterium/tritium sputtering, self-sputtering and sputtering due to seeded impurities. The recycling coefficient is an external parameter and the energy losses due to interactions with hydrogenic atoms (line radiation, ionization and charge exchange) are accounted for in the model. A simple slab geometry (poloidal and radial directions) with classical parallel transport and anomalous radial transport ($D_i = \chi_i = 0.5 \chi_e = 0.25 \text{ m}^2\text{s}^{-1}$) is used and the impurity fluxes and radiation losses caused by intrinsic and seeded impurity ions as well as by He ash are calculated self-consistently. Schematically the integration domain and the boundary conditions used in EPIT code are shown in Fig.1. The standard sheath boundary conditions are imposed at the plates, whereas the boundary conditions are given by decay lengths ($\lambda_n = 3 \text{ cm}$, $\lambda_T = 4 \text{ cm}$) at the wall. The parallel velocities and the gradients of densities and temperatures are assumed to be zero at the midplane (stagnation point).

The coupling between the core and the SOL is made by imposing continuity of energy and particle fluxes as well as of particle densities and temperatures at the separatrix. The computed fluxes from the core are used as boundary condition for the SOL plasma. In turn, the values of temperatures and of densities calculated in the SOL are used as boundary conditions for the core module. In the core, the time dependent transport equations are solved. For each time iteration in the core, several time steps (10 - 20) are performed in the SOL to adjust the edge parameters \[4\].

### 3 Results and Discussion

The DEMO project is currently at the conceptual design stage and consequently no final configuration is defined. Among different DEMO concepts which are under discussion in Europe, we have chosen 4 configurations\[1\] corresponding to inductive (Pulsed) and Steady-State scenarios. The basic input parameters used for our simulations corresponding to four different demo concepts are shown in Table 1. In all scenarios, tungsten is assumed as the wall material and argon (Ar) or neon (Ne) is puffed as the seeding impurity...
in the divertor region. In order to keep the prescribed plasma density at the separatrix (at stagnation point), the hydrogen recycling coefficient \(0 < R_H < 1\) was iterated accordingly. Please note that following definition of the recycling coefficient is used in our model:

\[
R = 1 - \frac{\Gamma_{sep}}{\Gamma_{plate}}, \quad \text{where } \Gamma_{plate} \text{ is the total particle flux to the target and } \Gamma_{sep} \text{ is the total flux crossing the separatrix.}
\]

For helium, the recycling was assumed to be two times smaller than for hydrogen \(R_{He} = 2R_H - 1\) whereas for seeded impurities (Ar, Ne) constant value has been assumed \(R_{Ar,Ne} = 0.925\). Simulation results for basic demo scenarios (DEMO1-4) are shown in Fig.2 for different levels of Ar seeding. In all cases significant amount of alpha power is produced but \(Q\) -factor is the highest (~30) for pulsed DEMO configurations (DEMO1-2) characterized by small level of additional heating. For steady state scenarios (DEMO3-4), relatively large additional heating is required to drive the plasma current and consequently \(Q\) factor is low (<15). The increase of Ar influx is accompanied by increase of \(Z_{eff}\) and reduction of \(P_\alpha\) and corresponding decrease of \(Q\) factor (Fig.2a, d, e). This is caused by the changes to the density of

<table>
<thead>
<tr>
<th></th>
<th>Pulsed DEMO1</th>
<th>Pulsed DEMO2</th>
<th>Steady State DEMO3</th>
<th>Steady State DEMO4</th>
</tr>
</thead>
<tbody>
<tr>
<td>(R_0, a [m])</td>
<td>9.0, 2.25</td>
<td>9.6, 2.4</td>
<td>8.5, 2.83</td>
<td>8.5, 2.83</td>
</tr>
<tr>
<td>Plasma current (I_p) [MA]</td>
<td>16.44</td>
<td>18</td>
<td>19.75</td>
<td>23</td>
</tr>
<tr>
<td>Toroidal magnetic field (B_T) [T]</td>
<td>7.2</td>
<td>7.45</td>
<td>4.9</td>
<td>5.74</td>
</tr>
<tr>
<td>Elongation (\kappa_95), Safety factor (q_{95})</td>
<td>1.66, 3</td>
<td>1.66, 3</td>
<td>1.66, 3</td>
<td>1.66, 3</td>
</tr>
<tr>
<td>Density (&lt;n_e&gt;_{VOL}) (\times 10^{19} m^{-3})</td>
<td>8.8</td>
<td>9.1</td>
<td>7.5</td>
<td>9.1</td>
</tr>
<tr>
<td>Separatrix density (n_{es}) (\times 10^{19} m^{-3})</td>
<td>3.5</td>
<td>4</td>
<td>3.5</td>
<td>4</td>
</tr>
<tr>
<td>H-factor (IPB98(y,2))</td>
<td>1.0</td>
<td>1.2</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>Auxiliary heating (P_{aux}) [MW]</td>
<td>51.8</td>
<td>100</td>
<td>163</td>
<td>200</td>
</tr>
</tbody>
</table>

Figure 2: Plasma parameters versus gas puff levels: a) \(Q\)-factor, b) total radiation \(P_{rad}^{tot}\), c) power to target plate \(P_{plate}\), d) \(Z_{eff}\), e) alpha power \(P_{\alpha}\), f) power radiated by tungsten \(W_{rad}\).
fuel ions due to the dilution effect of argon and helium. We note, that at the same time the plasma temperature in the center remains almost constant not being affected by seeding (not shown here). The total plasma radiation depends very weakly on the argon influx, with majority of the radiation losses coming from the plasma core (> 90%, with significant tungsten contribution (above 50%) (Fig. 21). The radiation in the SOL is usually small (< 10%) but sufficient to achieve high recycling or semi-detached conditions in the divertor. It should be noted, that in the reactor with very high input powers, the plasma temperature at the target plates is set up mostly by the sputtering processes and remains slightly above the sputtering threshold for argon ions (≥ 5 eV) (see Fig. 3). This is the result of a self-regulating mechanism being specific feature of tungsten (metallic) targets. This mechanism regulates the tungsten production due to sputtering processes at the target plates by radiative cooling of tungsten ions in the plasma center. Since the radiation efficiency of tungsten is very high and simultaneously the dependence of the sputtering yield on the temperature (incident ion energy) is very steep the equilibrium between production and radiation appears at temperature values very close to the sputtering threshold. At much higher temperatures the production and corresponding radiation would be too large leading to discharge termination. This mechanism is also responsible for the fact that W radiation is relatively weak function of the puffing intensity. However, the low temperature conditions in the divertor do not guarantee strong enough reduction of the power to the target plates. Only for the highest seeding levels the heat load might be tolerable (Fig. 2c), again mostly due to reduction of the input power (Pα) with increased seeding and not by cooling action of argon ions. We note however, that for strong seeding the power crossing the separatrix approaches the power threshold for L-H transition. If the seeding impurity is changed from argon to neon the overall picture remains similar as it can be seen from Fig. 4 for DEMO1 configuration where the basic discharge parameters are shown for Ar and Ne seeding.

It should be stressed that for pulsed DEMO configurations (DEMO1-2), we observe a maximum value of the allowed influx of seeded impurity above which the steady state solution does not exist, whereas for the steady state configurations (DEMO3-4) the input flux of additional impurity might be increased up to the level when the detached conditions in the divertor are achieved (Fig. 2 - the last point for the DEMO3 configuration). However, in all cases, the maximum values of the seeding impurity puffing rate are relatively low, and consequently the contribution of the injected impurity ions to the total radiation is always small, even for detached plasmas. In addition, the detached conditions in divertor are achieved at very poor plasma performance with Q-factor ~ 5 and large values of Zeff (see Fig. 21). The reason for the limits of the puffing level for pulsed DEMO configurations is not fully clear for us. It is connected with nonlinear interaction

Figure 3: Sputtering yields for tungsten at normal incidence angle (Ion energy ~ 5T_e)
between plasma (helium) transport and fusion power production, which is particularly strong when additional heating is low, like for pulsed configurations. It is observed, that helium concentration increases together with the increase of the seeding intensity and at some puffing level burning is stopped by plasma dilution by helium. The termination of the solution is very sharp and caused by the negative feedback which starts when the puffing limit is exceeded: $\Gamma_{\text{puff}} \uparrow \Rightarrow P_a \downarrow \Rightarrow \tau_E \uparrow \Rightarrow \tau_{He} \uparrow \Rightarrow c_{He} \uparrow$ and due to dilution again $P_a$ is reduced. In Fig. 5a we show helium concentration $c_{He}$, energy confinement time $\tau_E$, helium confinement time $\tau_{He}$ and the product $S_{He} \times \tau_{He}$ versus argon puff level ($\Gamma_{Ar}^{\text{puff}}$) for DEMO1 configuration. It can be seen that the product $S_{He} \times \tau_{He}$ which drives helium concentration is growing strongly with puffing. Although the increase of $\tau_E$ is quite moderate (from 2.48 sec up to 2.76 sec), the corresponding change to $\tau_{He}$ is quite drastic (9.6 sec $\rightarrow$ 18.5 sec) and the reason is still to be identified. In fact, there are different physics processes in the loop which lead to the
instability and it is difficult to find out which one is the most important. However, careful
analysis of the simulation results indicates that the improvement of the plasma confinement
seems to be the main driver for the instability. In order to check this hypothesis we have performed simulations for different confinement condition. At the fixed argon puff level \((Γ_{puff}^{Ar} = 5 \times 10^{20})\) s\(^{-1}\), numerical scans were done with different \(H_{98}\) factors and plasma currents \((I_p)\) and the results are shown in Figs.5b-c. It appears, that steady state solution does not exist if the maximum values of \(H_{98} (= 1.17)\) and \(I_p (= 22\) MA) are exceeded. These results show that improved plasma confinement in the fusion reactor might lead to strong helium accumulation. Therefore it is necessary to find out how this problem can be mitigated. This objective is however outside the scope of this paper.

Acknowledgement: The authors would like to thank Dr. R. Neu and Dr. T. Pütterich from IPP Garching for providing them with atomic data for tungsten.

This work was supported by EURATOM and carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

References

[3] F.Romanelli et al., Overview of the JET Results with the ITER-like Wall, this conference (OV/1-3)
[12] ITER Physics Guidelines, ITER report N 19. FDR 1 01-07-13 R 0.1