Plasma Physics and Controlled Fusion, Vol. 33, No. 13, pp. 1579–1600, 1991.
Printed in Great Britain.
0741–3335/91 $3.00 + .00
IOP Publishing Ltd. and Pergamon Press plc.

HELIUM REMOVAL FROM TOKAMAKS

D. Reiter, H. Kever, G.H. Wolf, M. Baelmans*, R. Behrisch**, R. Schneider**

Institut für Plasmaphysik, Forschungszentrum Jülich GmbH,
Association EURATOM-KFA
Postfach 1913, D-5170 Jülich, FRG

ABSTRACT

Helium, as the ash of burning D-T plasma, is an unavoidable impurity component necessarily present already in near future tokamak experiments with significant alpha particle heating. Its efficient removal from the burning zone of a D-T fusion reactor (or lack thereof) will play a key role in the path towards achievement of economic fusion power production. A survey is given of the issues related to this question. Since there is as yet no experimental experience with thermonuclear plasmas significantly heated by fusion products, this review is based on results from simulation experiments of helium injection into hydrogen or deuterium tokamak plasmas, and from numerical transport code work. Both kinds of results are discussed with reference to handy ignition criteria obtained for steady D-T burning, which have been reformulated in terms relevant for the ash removal problem.

KEYWORDS

Tokamak, Helium Accumulation, Helium Removal, Divertor, Pumplimiter

1. INTRODUCTION

Any kind of steady burning process requires both sufficient thermal insulation (to keep the temperature high enough) and, simultaneously, sufficient particle throughput (refuelling and ash removal). For example, a simple candle flame is choked within seconds by its own ash, if, under otherwise identical conditions, gravity and consequently boyancy driven flow is turned off. (This demonstration experiment is occasionally carried out during astronaut training programs on parabolic plane flights.) For a steady burning D-T plasma, these conditions translate into the following requirements:

* The quality of heat insulation (ie. energy confinement time τ_E) must be high.
* The fast alpha particle confinement (τ_α^{fast}) must be long, for efficient heating.
* However, the thermal particle confinement (τ_p) must be small, for high particle throughput.

These three core plasma issues define a new and challenging transport problem in itself, especially since in present experiments particle and energy transport are closely connected.

* Department of Mechanical Engineering, Katholieke Universiteit Leuven, Celestijnenlaan 300A, B-3001 Heverlee BELGIUM

** Max-Planck-Institut für Plasmaphysik, EURATOM-Association, D-8046 Garching, FRG

Unfortunately, a positive correlation is most apparent in the transition to "improved confinement" regimes. Simple burn criteria, reformulated to explicitly focus on this correlation, will be used to quantify the problem in the next section.

* Furthermore, the particles must not only be transported to the plasma edge, but they must also be efficiently removed from there.

Not removing the helium ash from the plasma would lead to a relative helium concentration growth rate of about $0.01/\tau_E$. This would be much too large for economic reactor operation (with burn durations of many hundreds of τ_E times). Hence pumping speeds of the order of 300 to 1000 cubic meter per second are discussed for the ITER design. The vacuum pumping problem of a fusion reactor will probably require some novel solution, and the tritium throughput, depending on the relative pressure of helium to fuel particles in the pump duct region, will be an important safety issue.

* Even under ideal conditions, there will only be a small probability for each particle incident onto a target surface to be pumped in a single event. Therefore, repenetration of unpumped particles must be controlled by establishing appropriate plasma edge conditions. This means, shielding efficiencies against repenetration far in excess of what has been observed in present experiments must be accomplished.

In this paper we will first discuss problems related to helium transport in the main plasma, which determines the amount of fusion power degradation due to fuel dilution. As present day experiments are not with burning plasmas, for extrapolation one has to resort to either simulation of burning plasmas in computer experiments (transport modelling), or to simulation of alpha particles in real experiments e.g. by helium neutral beam injection, helium gas puff or He^3 minority ion cyclotron resonance heating.

Considering the ITER conceptual design and helium exhaust issues identified therein, we will then turn to edge related phenomena, such as helium pressure build-up and possibilites for helium enrichment in pump ducts. With regard to this question, some experimental data are available as well as validated computer models for neutral particle and plasma transport. We will conclude this paper with a summary and a listing of critical issues to be addressed in the near future.

We would like to point out that the aforementioned problems and related questions have been addressed recently at an international workshop on Helium Transport and Exhaust held in April 1991 at Gatlinburg, Tennessee. This meeting has covered the general topic areas of helium burn conditions, helium transport and exhaust experiments and modelling of H- and L-mode operation, helium detection methods (CER spectroscopy), helium NBI, conventional and more novel helium pumping schemes, as well as helium retention and release from relevant tokamak wall materials (Hillis, 1991 and Hogan and Hillis, 1991).
A significant amount of the material reviewed in the present paper is based on information given by about 50 scientists representing 15 institutes during this meeting. A meeting summary report is in preparation (Hogan and Hillis, 1991).

2. SIMPLE ESTIMATES

Zero-dimensional models are widely used as tools in both fission and fusion reactor studies to eludicate various aspects of the otherwise very complex systems.
On the basis of such point reactor studies, effects of alpha particles on the power balance of a burning D-T plasma have often been considered in the literature. Probably the first extensions of ignition conditions or of the Lawson criterion, to include fractional burnup, alpha particle slowing down and helium pressure, have already been carried out more than 20 years ago (Rose and Clark, 1965; and Rose, 1968). Different confinement systems have been considered in these papers. It has been found even for open mirror systems with intrinsicly large particle throughput "...that the build-up of He^4 is alarming." (Rose

and Clark, 1965). These estimates have been refined and completed since then for next step reactor designs (e.g. Meade, 1974; Bromberg *et al.*, 1979; Engelmann, 1980), and supplemented by investigations of thermal stability characteristics.
All these models, as well as the widely used POPCON plots (e.g.: Furth, 1990), are derived from power balance considerations alone, sometimes with prescribed fractional helium densities $f_{He} = n_{He}/n_e$ to account for fuel dilution and helium radiation losses.

Only recent reformulations of such models (Behrisch *et al.*, 1990, Reiter *et al.*, 1990a and 1990b) account for the fact that f_{He} is not an independent parameter, as are other impurity levels, but determined by the coupling of heat and ash production rates.

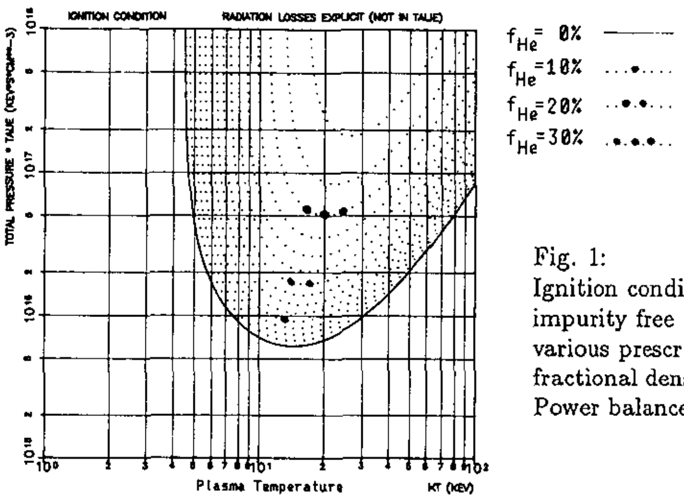

Fig. 1:
Ignition condition for an otherwise impurity free D-T plasma, and for various prescribed helium ash fractional densities f_{He} = 0,35,2.5%. Power balance equation alone

Combining the alpha particle balance equation with the power balance equation results in a cubic equation for f_{He}, with two physically meaningful roots, both strongly dependent on plasma temperature and confinement regime. This changes the topology of ignition contours (fusion parameter $n \cdot \tau_E$ or $n \cdot T_e \cdot \tau_E$ vs. plasma temperature T), which become closed, replacing the open curves which result if f_{He} is taken as an independent parameter (fig. 1). Depending on the confinement regime, the helium exhaust efficiency and the other impurity levels, these closed contours shrink in size and finally vanish if the control parameter $\rho = \tau_\alpha^*/\tau_E$ exceeds a critical value ρ_{crit} (fig. 2). Here τ_α^* denotes the global alpha particle confinement time in the burning core region. The resulting interrelation between minimum helium exhaust requirements and maximum tolerable impurity contamination is illustrated in fig. 3, where ρ_{crit} is plotted against the fractional impurity density $f_z = n_z/n_e$ for selected light impurities typical in tokamaks. Even under the optimistic assumption of an otherwise pure D-T plasma (f_z =0.0) and perfect thermalization of all fusion alphas, one finds $\rho_{crit} \leq 15$, and, for example, for 2% carbon contamination, fig. 2 and fig. 3 show that ρ_{crit} becomes ≤ 9.
A simple analysis allows one to crudely relate the confinement time ratio ρ to transport and exhaust parameters. In the case of perfect helium pumping one may write:

$$\rho_{crit} \geq \rho = \tau_\alpha/\tau_E = \chi/D + C$$

This formula expresses the fact that the helium accumulation level in the core is (aside from pumping conditions to be discussed later) directly correlated with helium transport in

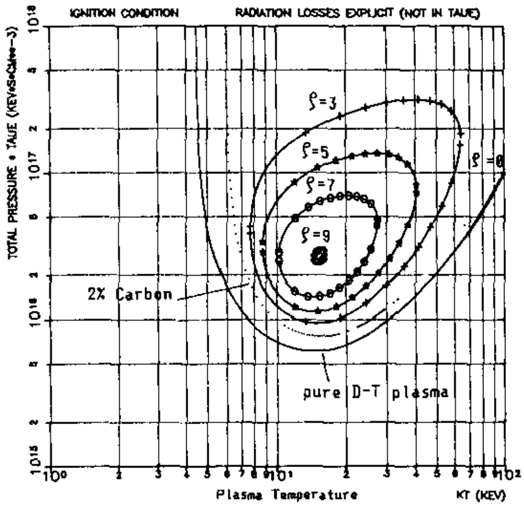

Fig. 2:
As figure 1, but f_{He} selfconsistently computed from particle and power balance equation.
Also exemplarily included:
2% carbon contamination

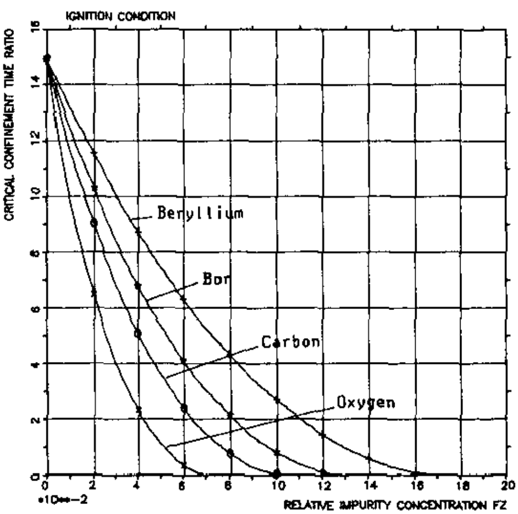

Fig. 3:
Interrelation of critical confinement time ratio $\rho_{crit} = \tau_\alpha^*/\tau_E$ and maximum tolerable impurity $f_z = n_z/n_e$

the core plasma. There is, in general, a contribution from diffusive (χ/D) and convective (C) transport. χ and D denote thermal and particle diffusivities, respectively.

The temperature multiplier C for convective heat transport is not very well defined, but it may be difficult to imagine values of C less than 1. The ratio χ/D has been studied in various experiments (see Post *et al.*, 1990, section 3.2.9.1 and references therein). Heat and density pulse propagation analysis (after sawtooth crashes) have shown $5 < \chi/D < 12$ in JET and $\chi/D \sim 3$ - 6 in JT-60. Smaller values of $\chi/D \sim 2$ - 3 have been reported from JET from profile analysis, and $\chi/D \sim 4$ for L-mode and $\chi/D \sim 1$ for supershot conditions in TFTR.

An inward pinch velocity V_{pinch} is often needed to explain experimentally observed peaked profiles. The anomalous pinch velocity is frequently expressed as $V_{pinch} = 2 \cdot C_v \cdot D \cdot \frac{r}{a^2}$,

where C_v plays the role of a profile peaking factor. Values of $C_v \sim 1$ - 2 for helium and lower values for hydrogen were reported from JT-60. Similarly $C_v \sim 2$ and ~ 1.6 for helium and deuterium, respectively, have been deduced from TFTR, and $C_v \sim 0.8$ - 1.4 from TEXTOR.
Direct experimental results for τ_α/τ_E are reported from TEXTOR and TFTR, with values in the range ~ 2 - 3 in both cases.

Nonperfect pumping can globally be described by a (nonvanishing) recycling coefficient R, which, realistically, will be closer to 1 than zero at the target plates.
However, replacing τ_α by $\tau_\alpha^* = \tau_\alpha/(1-R)$, as it is often justified in present day experiments, would grossly overestimate the effect of recycling on core plasma properties in the case of efficient edge shielding. If, for example, a value of R=0.998 (as specified for the ITER divertor target plates) would be taken as the characteristic recycling coefficient for the confined plasma region, then this would result in a multiplication factor for ρ of 500, ie. confinement time ratios far beyond anything meaningful for ignited plasma operation.

Clearly, the complex relation between τ_α, τ_α^* and an effective recycling coefficient R for the confined plasma region is controlled by local transport properties and detailed plasma edge physics. An expression

$$\tau_\alpha^* = \tau_{\alpha 1} + R/(1-R)\tau_{\alpha 2}$$

was proposed by (Reiter *et al.*, 1990a), with $\tau_{\alpha 1}$ and $\tau_{\alpha 2}$ being the characteristic lifetime in the burning core region of first generation (ie. fusion generated) and for wall recycled alpha particles, respectively. This distinction of alpha particles may appear somewhat arbitrary. However, it is a well established result that τ_α^* depends critically (amongst other parameters) on the radial distribution of the particle source, which certainly will be very different for the two types of particles introduced above. Furthermore, these particles differ in their velocity distribution, and probably also in collective behaviour, as e.g. fast alphas can excite Alfven waves (see section 3). Suffice it here to state that the above relation has been confirmed by an extensive BALDUR transport code study, carried out for ITER relevant conditions (Hu and Miley, 1991). The resulting lifetimes characteristic in the BALDUR transport model for the inner and outer plasma region, respectively, were indeed found to be quite different, namely $\tau_{\alpha 1} = 8.18$ s and $\tau_{\alpha 2} = 0.55$ s for the ITER transport model described in (Redi and Cohen, 1990a). A further discussion of R in terms of pumping and shielding efficiencies will be given below.

3. HELIUM IN THE PLASMA CORE

Alpha particle physics will be an important new subject in next generation tokamak experiments. Present day experimental findings from JET and TFTR (Keilhacker, 1991 and Hawryluk, 1991) of classical slowing down times for fast helium ions, which at present are either created by neutral helium beam injection or result from D-D fusion reactions, are promising in terms of α heating efficiency. However, the slow thermalization ($\tau_{\alpha s} \sim 0.5$ - 1 s) would lead to a significant contribution of fast α particle pressure in a reactor. For example $\beta_{f_\alpha}/\beta \sim 30\%$ is expected for ITER. Furthermore, this leads to a vulnerability of the fast alphas to radial transport during the thermalization process.

For a review of α particle physics such as α transport coefficients, loss mechanism, collective α instabilities or fusion heat driven collective phenomena the reader is referred to the excellent compilations of Kolesnichenko and Ya, 1980, Post *et al.*, 1990, section 2.4 and Furth *et al.*, 1990. Also discussed there is the new issue of burn control in α particle heated plasmas.

Assessments of these issues using both global and $1\frac{1}{2}$-dimensional transport models with *emphasis on α particle effects have* been carried *out for various planned burning plasma* experiments (Uckan *et al.*, 1988). Critical issues identified in this work, in addition to the (thermal) helium removal problem, are the contribution of the fast alpha population to the plasma β limit, the influence of instabilities and turbulence on alpha thermalization (with alpha-driven fishbones probably being the most threatening instability for fast alpha losses) and radial transport of energetic alphas.
Direct losses of fast alphas will clearly have an impact on fusion reactor design, although power balance considerations would allow rather high losses (5 - 10 % for ITER). The actual limitation is related to restrictions for localized heat load on the first wall, induced by the toroidal field ripple. Assuming such ripple losses, however, to be the most critical channel, a fraction of below 1% is found for ITER from numerical code results, and this is considered acceptable.
For the rest of this paper we will, therefore, deal exclusively with thermalized α particles and their removal from tokamaks.

The point reactor models mentioned in section 2 are useful tools to examine and illustrate general features of a burning plasma, and especially allow the identification of critical parameters such as the ratio χ/D or the effective helium recycling coefficient R_{He}, but (this is their weakness) they have to be based on ad hoc assumptions of profile shapes for all relevant quantities. These assumptions may not necessarily be consistent with any realistic physical transport models. Results from complex numerical transport code simulations can at least be related to a (in most cases empirical) transport model. However, relevant physics is often hidden behind the large number of adjustable "knobs" in such codes and by the fact that each run yields only a single point in a multidimensional parameter space. Nevertheless, recent attempts have been made to address the helium poisoning problem by systematic 1-d radial transport studies.
Probably the most complete assessment has been carried out for ITER ignited scenarios (see Redi and Cohen, 1990a, 1990b and references therein). For these BALDUR transport code simulations, an anomalous transport model was chosen, which was derived from JET H-mode data, and normalized to the ITER89P τ_E scaling (see below, section 4). A ratio $\chi_e(r)/\chi_i(r) = 2$ was assumed, together with the radial dependence

$$\chi_e(r) = \chi_{eo}(3 + 5 \cdot r/a) \ .$$

The absolute value of χ_{eo}, computed from confinement time scaling for the baseline ITER 1990 design (I_p=22MA, B=4.85T, R=6m, a=2.15m, elongation κ=2, P_α=200MW, P_{Fus}= 1000MW) results in energy confinement times τ_E ranging from 2 - 4 s. The ratio χ/D was varied, over the range 0.44 to 7., ie. covering most experimental data except those very high values (around 12) reported from particle and heatpulse propagation experiments in JET.
A particle pinch velocity was included, again defined by the relation $V_{pinch} = 2 \cdot C_v \cdot D \cdot r/a^2$. Transport of all species was assumed to be the same. As is obvious already from section 2, another critical quantity must be the effective recycling coefficient R. Justification for the range R_{He} =0.5 - 0.998 chosen in these studies comes from 2-d plasma edge simulations (Cohen *et al.*, 1990). However, as such fluid models provide only net flows, even if spatially resolved, it is intrinsically impossible to infer ratios of in and outgoing particle fluxes from these calculations (unless a particle, which has crossed a certain surface, say, the separatrix, is marked as a different "species").

As could have been anticipated on the basis of the estimates in section 2, it was found that this transport model does not result in sustained ignition unless edge recycling is sufficiently low, depending on the ratio χ/D. At low edge densities ($n_e(a) = 10^{18}m^{-3}$)

with $\chi/D = 7$, ignition was quenched after 10 to 30 seconds, even for the most effective helium removal $R_{He} = 0.5$ investigated. Furthermore, even the lowest ratio $\chi/D = 0.44$ did not result in sustained ignition for the high recycling coefficient $R_{He} = 0.998$, which is the estimated recycling coefficient at the target plate. This coefficient, however, is considerably over-pessimistic due to the neglect of any shielding action of the divertor plasma in this case. Stationary ignited plasmas were found for intermediate values, e.g. χ/D=3.5 and for efficient helium removal: $R_{He} = 0.5$.
At high edge densities ($n_e(a) = 3 \cdot 10^{19} m^{-3}$), these conditions are somewhat relaxed due to less efficient repenetration of unpumped ashes. For example, for a confinement time ratio $\chi/D = 3.5$ an effective recycling coefficient of $R_{He} = 0.9$ is found tolerable.
Another critical parameter was found to be the inward pinch factor C_v (as expected from the simple 0-d model of section 2, because C_v strongly effects the confinement time ratio τ_α^*/τ_E). Alpha power generation drops sharply for $C_v > 1$. Often quenching is found for $C_v \sim 2$, and sustained ignition does not occur if $C_v \sim 3$ regardless of R_{He}.
Inclusion of simple models for MHD effects on transport due to sawteeth and the β limit indicated that ignited operation is more attainable by reducing helium accumulation, however the physical credibility of such models and their effects on particle and energy transport has still to be demonstrated experimentally. Figure 4 and 5 show results from two such cases, with $R_{He} = 0.9$ and sutained ignition in fig. 4, and with $R_{He} = 0.998$ and quenched burn in fig. 5. No inward pinch was assumed in these two cases (taken from Redi and Cohen, 1990b), 90% of the α-heating power was given to the electrons, 10% to the ions.

We finally note here, that, according to what will be stated for ITER edge conditions in section 5, the choice of $R_{He} \geq 0.5$ might prove to be too pessimistic, since it does not take due account of the 2-d edge transport mechanisms leading to large flux amplification near the pumping station, and to very the low repenetration probability of unpumped helium particles. Correct accounting of these effects could result in an effective recycling coefficient of R_{He} close to zero if R_{He} is determined by the radial flow conditions near the separatrix.

Another transport code study addressing the helium ash accumulation problem was carried out on the basis of JET experimental results (Rebut, *et al.*, 1990 and Boucher *et al.*, 1991). The database from this tokamak supports a model for anomalous transport based on a single phenomenon and MHD limits. This "Critical Electron Temperature Gradient Model" features

- the degree of confinement degradation is determined by the electrons
- ion anomalous transport with heat diffusivity χ_i linked to electron heat diffusivity χ_e
- anomalous particle diffusivites, D, for ions and electrons, proportional to χ
- anomalous particle "pinch" for impurities alone.

This model has been validated over a wide range of JET operational conditions, such as L and H mode discharges (with H mode triggered "by hand" in the model by turning on a classical transport barrier in the high shear region near the separatrix), heat and density pulse propagation following sawteeth collapses, and various heating and fuelling scenarios. No anomalous particle pinch was needed to describe the density evolution (except for impurities).

Application of this model to a predictive DEMO reactor study was carried out, under the following operational conditions: $T_i \sim T_e$; L-mode confinement; D-T mixture including helium ash; sawteeth; β limit instabilities; adequate impurity control. It was shown that ignition is maintained in DEMO (R = 8 m, a = 3 m, 4.5 T, 30 MA, elongation $\kappa = 2$) after switching off the 50 MW of ICRH (fig. 6). However, this requires adequate He pumping, which translates into the need for pumping about 1 kg of D-T per hour.
The H-mode model was also applied for the same DEMO parameters. In this case, it was found that the short term benefits of H-mode for approaching ignition are offset by the long

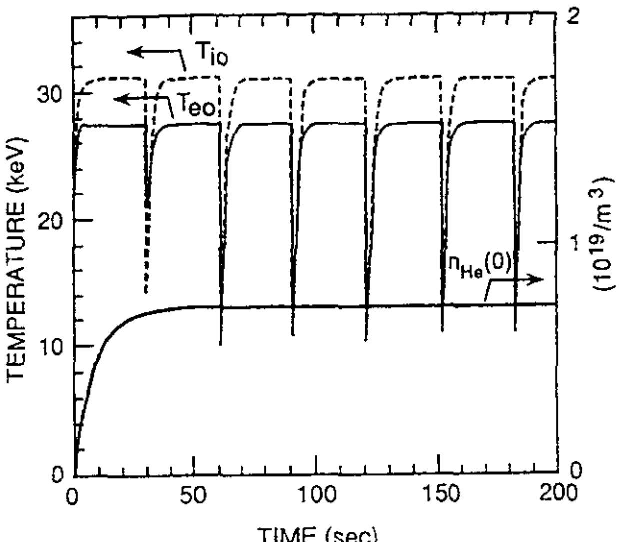

Fig. 4:
Evolution of central electron and ion temperatures and central helium accumulation
R_{He}=0.9, χ/D=4
ignition sustained
(BALDUR code simulation)

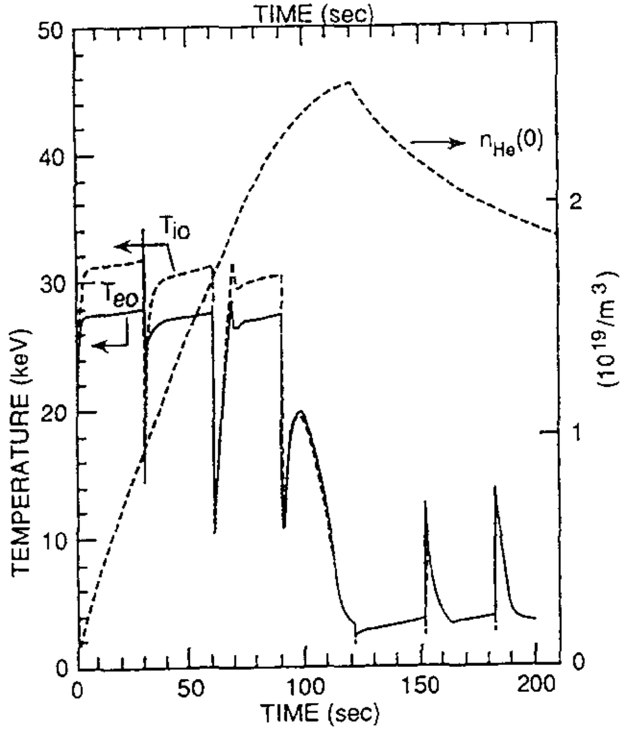

Fig. 5:
As figure 4, but R_{He}=0.998
ignition quenched
This figure and fig. 4 are taken from
Redi and Cohen (1990b)

term deficiencies due to *helium poisoning* (fig. 7). Eventually ignition is quenched, due to fuel dilution, even though the improved energy confinement in this confinement mode has lead to initially more alpha particle heating.

4. ITER HELIUM REMOVAL ISSUES

ITER is the first experiment for which helium exhaust is a decisive problem. Code calculations summarized in the previous section indicate a burn duration of 10 to 30 s only, if the helium content cannot be controlled, well short of the planned 200s.

In this section we briefly review the ITER relevant helium removal issues, which will also

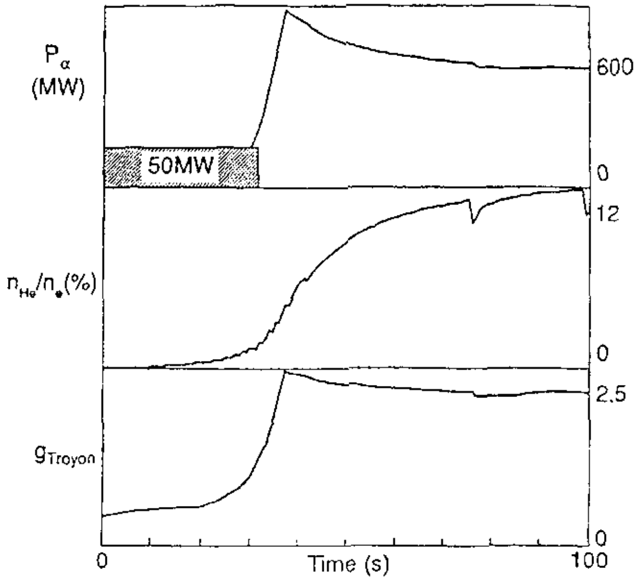

Fig. 6:
Model of DEMO plasma using L-mode transport model which has been tested against JET results.
This and fig. 7 are taken from Rebut *et al.* (1990)

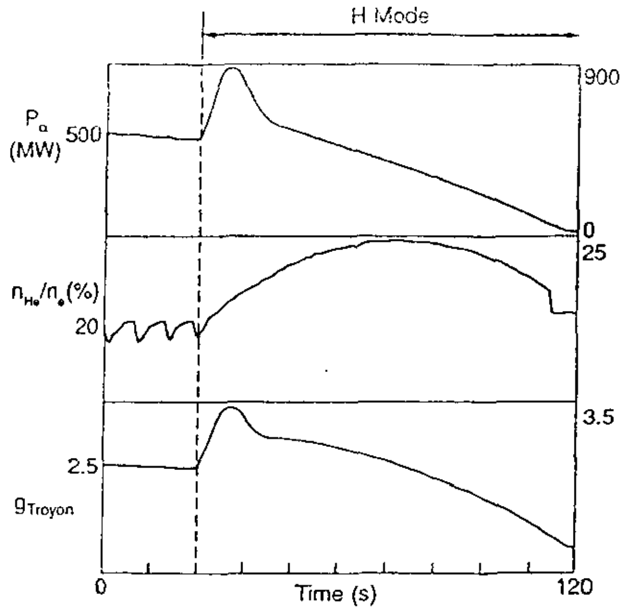

Fig. 7:
As Fig. 6 but with edge transport barrier included (H-mode model), showing long term deficiencies due to helium poisoning in H-mode

serve as the rationale for change-over from core to edge related problems. For further details the reader is refered to (Post *et al.*, 1990), and (Hogan and Hillis, 1991).
The major parameters and features of the ITER conceptual design are as specified in the ITER transport model discussed above. Pure H-mode confinement has become subject to a debate because of uncertainties in impurity and ash accumulation problems. It is planned for ITER, instead, to operate with high frequency, low amplitude ELM's and sawtooth control by current drive schemes. ITER will produce about $4 \cdot 10^{20}$ alpha particles per second.
The crucial quantity characterizing helium poisoning is again taken to be the ratio ρ of global alpha particle confinement time (e.g. at the separatrix) to energy confinement time τ_E. ITER will require ρ ratios not exceeding 10 for high Q operation (Q = 5 – ∞).
As for τ_E, the standard H-mode dataset (with and without ELM's), based on ASDEX,

DIII-D, JET, JFT-2M, PDX/PBX-M H-mode data, is the ITER-89 Power law scaling, given by:

$$\tau_E\ (H-mode) \approx 2.1 \cdot \tau_E^{ITER89-P}\ (L-mode)$$

and

$$\tau_E^{ITER89-P}\ (L-mode)\ = 0.048 \cdot I^{0.85} \cdot R^{1.2} \cdot a^{0.3} \cdot \kappa^{0.5} \cdot \bar{n}_{20}^{0.1} \cdot B^{0.2} \cdot A_i^{0.5} \cdot P^{-0.5}\ .$$

The database for particle confinement (τ_p, τ_α, τ_α^* etc.) is by far less well characterized. In particular the dependences on size, on B, and in the case of H-mode on heating power, are not clear.
For the extrapolation of present experimental data to ITER conditions there is a further dilemma, because τ_p^* will be strongly affected by the scrape off layer plasma conditions. ITER, in contrast to present experiments, will operate with plasma edge conditions in which penetration of neutral particles inside the separatrix is strongly reduced. Therefore, local diffusion and inward pinch coefficients are needed for extrapolation, in combination with a realistic SOL model, rather than global particle confinement times and their dependence on global plasma parameters which are usually provided from present experiments.
Experimental data for χ/D and the peaking factor C_v have been summarized in section 2 already. On this basis, the following scenario is anticipated and supported by extensive computer code modelling for the removal of helium from the core through the SOL:
The alpha particle flux of $4 \cdot 10^{20}$ particles/s will be amplified in the SOL to values around 10^{23} particles/s at the divertor targets, due to the formation of an "ultrahigh recycling" regime. Atomic physics estimates show that from the lower outer target, neutralized helium particles will reach the pumpduct entrance at a rate of about 10^{22} particles per second. Of this flux only the small fraction of a few percent needs to be pumped in order to obtain very small effective helium recycling coefficients $R_{He} \sim 0.1$. This fraction of 2 - 5 % can be achieved with pumping speeds of 100 — 1000 m^3/s at the pumps, depending on various details of the near target plasma edge conditions and pump duct conductancies.

5. HELIUM IN THE PLASMA EDGE

As pointed out in the previous section, edge plasma transport properties will play a key role for the problem of helium removal from tokamaks. Since the ITER poloidal divertor edge plasma conditions are expected to be very different from present day experiments (not to mention those in future fusion reactors), the predictive computer simulations take over an essential part here.
Historically, this was first addressed by INTOR divertor modelling studies (Seki *et al.*, 1980, Heifetz *et al.*, 1981). Unfortunately, these Monte Carlo studies, both carried out for similar configurations and plasma conditions, gave quite contradictory results. Helium enrichment factors of 2 or more in the pumping duct were found by Seki *et al.* using the DICON code, whereas a general tendency towards helium de-enrichment was predicted by DEGAS code calculations.
This discrepancy was attributed to different modelling details such as the atomic physics data used by the two codes. Recently, (Reiter in: Hogan and Hillis, 1991) this issue was reinvestigated by the EIRENE code, taking the same geometries and plasma parameters, even using the old surface and atomic data available at that time in the two codes DICON and DEGAS.

Both results were confirmed. It was found, that, although the more complete atomic and, in particular molecular physics in the DEGAS code are important, the apparent discrepancy resulted from accounting for edge plasma shielding effects (by an electrical field and by friction), which was included in a Monte Carlo Fokker Planck trace ion approximation in

DICON but not in DEGAS. Neither the DICON nor the DEGAS code was coupled to a fluid model for the fixed background plasma mixture of 90% hydrogenic and 10% helium ions. But in the DICON code those ions, which were created by reionisation of recycled particles in the divertor region, were also followed and had a certain chance to be pumped at a "later generation".

This conclusion is consistent with earlier EIRENE code validation experiments carried out at TEXTOR (Hardtke *et al.*, 1989) in which a deuterium and helium gas puff into the pumping chamber of the ALT-I pump limiter provided a well defined "recycling source" for neutral plasma interactions inside the pump limiter scoops. It was confirmed that the experimentally observed pressure decay time in the pumping chamber (as a result of back-flow conductance of neutrals reduced by the edge plasma streaming into the pumplimiter scoops) could qualitatively and quantitatively be modelled with EIRENE, however, only if reionized particles were followed as well (accounting for an electrical presheath with voltage corresponding to $0.5 \cdot T_e$). If, on the other hand, a stand-alone neutral particle calculation was carried out on a prescribed but fixed plasma background (as is usually done in helium pump efficiency assessments by Monte Carlo codes), in which reionization is treated as a loss from the test particle community, the experimental and theoretical trends were just the opposite of each other.

Similar modelling studies have recently been repeated, investigating deuterium and helium transport for the ALT-II pumplimiter experiments at TEXTOR. Previous findings were confirmed (Corbett *et al.*, 1989, and Reiter in Hogan and Hillis, 1991). The 3-d model geometry is represented in fig. 8. Shown are two of eight pumping stations, attached to the TEXTOR vessel in the lower outer poloidal section. Also included in the model is a detailed description of the two sided scoops (electron and ion drift side), even including diagnostic probes in these scoops, and the toroidal belt limiter, under which these plasma collector scoops are mounted. This Monte Carlo model is routinely benchmarked against ALT pumplimiter experiments, over the full range of operational regime from pure vacuum to high particle flux conditions.

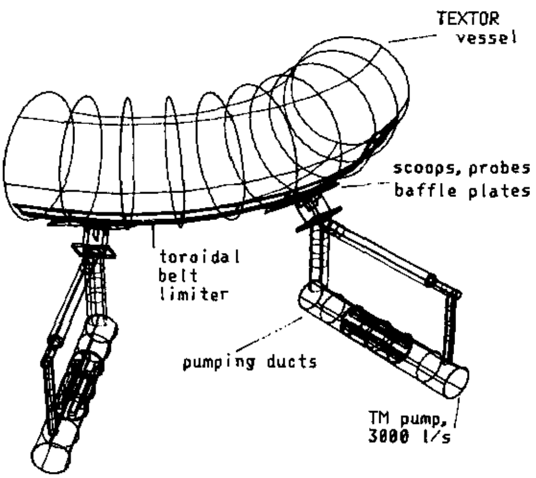

Fig. 8:
3-d geometrical model of ALT-II pumplimiter at TEXTOR, for EIRENE Monte Carlo code studies

It is concluded from these modelling studies, that the plasma streaming into the ALT-II scoops (typically edge plasma conditions: $T_e = T_i =$ 10-15 eV, $n_e = 4 - 8 \cdot 10^{17} m^{-3}$ in the helium removal studies), reduces the helium pumping efficiency as compared to hydrogen pumping efficiency by about a factor of 2, but only for the first generation of neutrals. Retention of reionized particles in the scoops by the presheath electric field (and possibly also by frictional forces, which were not included in this model) is more effective for reionized helium than for reionized hydrogenic particles, such that the net effect on overall pumping performance vanishes (in this special experimental situation). This is consistent with the experimental findings discussed below, in which about equal exhaust efficiencies for helium and hydrogen were obtained.

Summarizing, the present situation, also consistent with other modelling and experimental results published so far, may be described as follows:

* Due to different atomic physics (less efficient ionization and charge exchange processes, no molecular dissociation processes), neutral helium particles travel further away from their target birthpoint than hydrogenic particles, at least in a predominantly hydrogenic plasma. This results in helium enrichment at large distances (in terms of mean free paths) from the target, e.g. at the divertor entrance in the DEGAS calculations quoted above. Helium de-enrichment is observed near the target (same DEGAS results, at the pump duct entrance, and, e.g., EIRENE ALT-II modelling results).

Therefore, if ε_{He} and ε_H denote the pumping probabilities for neutral helium and hydrogenic particles, respectively, one may have, on the basis of different atomic physics and largely depending on geometrical and plasma conditions, ratios $\varepsilon_{He}/\varepsilon_H$ larger or smaller than 1. This effect has for example led to the proposal of an "inverted pumplimiter configuration" for preferential helium removal (Prinja and Conn, 1985).
However, things turn out to be more complicated:

* Upon ionization, in general, hydrogenic particles are much more energetic than helium (e.g. mean energies of 80 eV for D and T and only 4.4 eV for reionized helium particles were found in the benchmark study between DICON, DEGAS and EIRENE mentioned above in which a constant divertor plasma temperature of 250 eV was assumed).

This trend is observed in the narrow pumplimiter scoops and in divertor configurations, in which neutral particles experience many surface collisions before they are lost. This follows from the more efficient neutral particle heating for hydrogenic particles than for helium, again by charge exchange processes. Helium particles can not recover the energy which they lose in collisions with the divertor chamber walls.

* Consequently, reionized helium may be more efficiently trapped in presheath fields or screened by friction, as long as ionization does not take place too far away from the target.

Ie. if one writes γ_{He} and γ_H for the probabilities for repenetration into the plasma for helium and hydrogenic ions respectively, charged particle transport models (kinetic, as in the DICON code or fluid (Braams, 1987)) will in general yield ratios $\gamma_{He}/\gamma_H < 1$.

Overall helium and D or T removal is determined by both the atomic physics and pumping speed controlled number ε, and by the plasma edge transport determined number γ. In Reiter *et al.* (1990a) it was shown that the coefficient R/(1-R), in the relation for τ_α^* discussed above, can be interpreted as the mean number of return events to the confined plasma for a single particle due to nonperfect pumping. This number may be written as:

$$A = R/(1-R) = \gamma \cdot (1-\varepsilon)/\varepsilon \quad .$$

Poisoning of the core plasma due to recycling helium depends on the ratio A_H/A_{He}. Helium enrichment may then be characterized by $A_H/A_{He} > 1$. It becomes obvious that both helium enrichment or de-enrichment may result for $\varepsilon_{He}/\varepsilon_H$ either above or below 1. Ie. addressing this question not only depends on geometrical and other experimental details, but it also requires a consistent description of both neutral and plasma transport. This mutual action of different transport mechanisms is especially critical if both ε and γ are small numbers (then: $A \sim \gamma/\varepsilon$), as is expected to be the case in the ITER conceptual design.

5.1 Experimental Results

Probably partly triggered by the apparent discrepancy in the modelling predictions, the helium ash exhaust of a divertor was investigated 10 years ago (Shimada *et al.*, 1981) in the DIII expanded boundary divertor configuration experimentally. Helium gas was injected into the vacuum vessel in a pulse of 5 ms duration at 600 ms in the discharge.
After the helium pressure reached equilibrium levels ($\geq$ 200 ms after the puff), the fractional helium density f_{He} in the core plasma and the fractional helium pressure in the divertor chamber were monitored. For f_{He} = 1.6% the neutral helium pressure in the divertor region was found to be $1 \cdot 10^{-4}$ Torr. This was a first indication of the possibility of helium ash exhaust with poloidal divertors and with pumping ducts of practical size.
The helium enrichment factor was defined as the ratio

$$\eta = (P_{He}/2 \cdot P_{H_2})_{divertor}/(n_{He}/\bar{n}_e)_{core}$$

of fractional helium pressure in the divertor to fractional helium density in the main plasma. It was found that η was always significantly below 1 (fig. 9), and decreasing with $\bar{n}_e$. This finding was consistent with the model predictions on the basis of different atomic physics: neutral helium reionization mean free paths were larger than the divertor dimensions, and only the hydrogenic particles, especially molecules with their much shorter dissociation lengths, were trapped in the open divertor chamber. The beneficial effect of preferential helium ion retention did not come into action because the helium ions were not at all created in the divertor region. If argon was puffed instead of helium, for example, enrichment factors $\eta_{Ar} > 1$ were always found. This can be understood from the fact that argon is effectively reionized within the near divertor target region, and argon ions are more efficiently trapped than hydrogen.

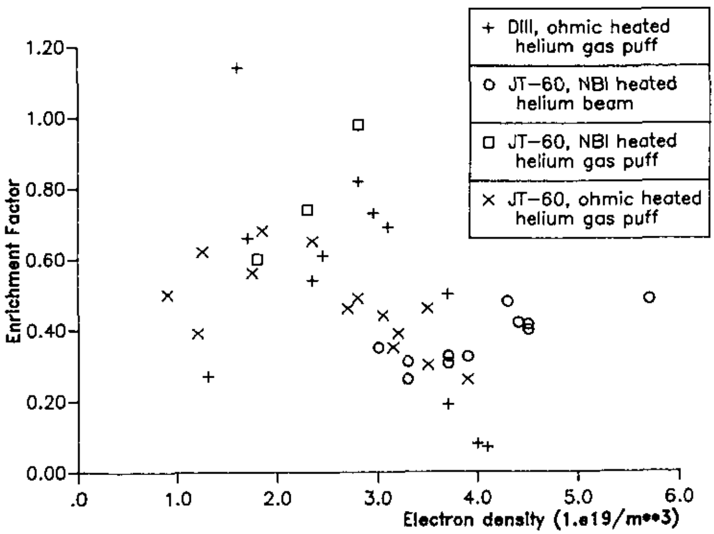

Fig. 9: Helium enrichment factors vs. density $\bar{n}_e$, from experiments in DIII and JT-60

Very similar results have recently been reported from JT-60 helium exhaust studies (Nakamura in: Hogan and Hillis, 1991). Both helium gas puffs and helium NBI are used in ohmical and beam heated (L-mode) plasmas. In addition to confirming the early DIII results for the ohmic heated plasmas ($\eta \approx 0.5$ and $\eta \sim \bar{n}_e^{-1.5}$), it was found that in beam heated plasmas η is still less than 1 but increases with density. Larger values of η (near 1) are found for gas fuelling than for helium NBI ($\eta \sim 0.3$), see figure 9. This is consistent with the general picture that fuelling of limited penetration assists helium (and impurity) removal, because, due to forced flows parallel to the open fieldlines in the SOL, the hydrogenic component flushes the boundary plasma (e.g. Kaufmann *et al.*, 1985, Keilhacker *et al.*, 1991a).

In these experiments gettering materials or cryopumping was used, which do not pump helium. The first active helium removal was demonstrated with the ALT-II pumplimiter system at TEXTOR, which uses turbomolecular pumps (Hillis *et al.*, 1990). In this experiment helium was injected as a short gas puff (20 ms) in beam heated deuterium plasmas, leading to a fractional helium density of 3 - 5 % in the core within 100 to 200 ms after the puff (fig. 10).

This helium density remains constant without pumping ($R_{He} \approx 1$). By turning on successively more ALT-II pumps, the helium signals from core, edge and the pumping chamber decay rapidly, and with full pumping on all eight stations, 3000 l/s each, helium is removed within 1 s from the discharge (fig. 11). Despite this the electron density stays fairly constant not only with feedback control on gas fueling, but also with preprogrammed fueling. This can easily be attributed to the large inventory of deuterium in the tokamak walls and limiters, which replenishes much of the deuterium exhausted by the pump limiter.

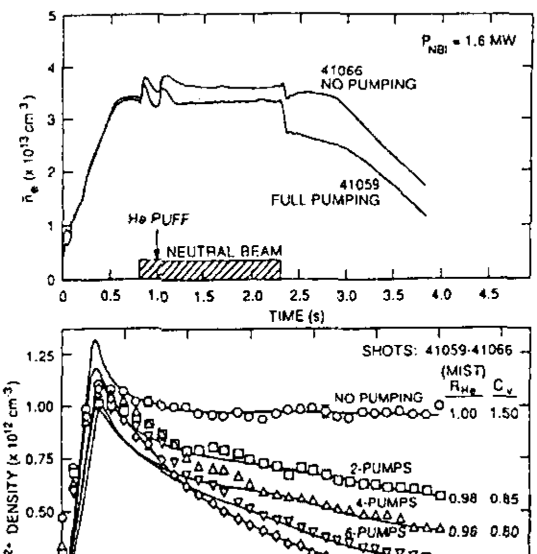

Fig. 10:
Line averaged electron density $\bar{n}_e$ vs. time, with feedback control on external D gas fueling to keep $\bar{n}_e \approx 3.2 \cdot 10^{13} cm^{-3}$ during NBI phase

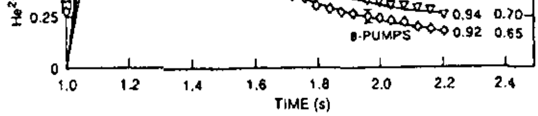

Fig. 11:
He^{++} density measured with CXE spectroscopy vs. time, with and without ALT-II pumping.
Solid lines:
MIST transport calculations

Similar exhaust efficiencies ε_D and ε_{He} for deuterium and helium, respectively, of 8 - 10 % were found, ie. $R_{He} \sim 0.92$ with full pumping. τ_α^*/τ_E, the figure of merit, is estimated for these L-mode conditions, to be in the range $10 < \tau_\alpha^*/\tau_E < 30$, ie. just marginally within the acceptable range for reactors (section 2).

Very preliminary results (Hillis in: Hogan and Hillis, 1991) and (van Nieuwenhoven *et al.*, 1991) have been reported from similar studies at TEXTOR with improved confinement

conditions. It was shown that the various confinement conditions did have significant influence on helium removal capabilities with the TEXTOR ALT-II system.
The results showed (loc.cit.) that in most (but notably not in all) cases of improved confinement helium removal was aggravated, but at different levels, depending on the polarity of the biassing probe which was used to trigger the various confinement regimes. More quantitive results are expected in the near future.

5.2 Edge Modelling

As pointed out in the preceding section, successful helium control via poloidal divertors (as e.g. foreseen in ITER) will depend on the formation of very high recycling conditions. The flux amplification factor F defined as the ratio of the core efflux to target plate fluxes is expected to be of the order of several hundreds.
These plasma edge conditions are determined, aside from core plasma efflux conditions, by the combined action of several physical effects, the dominant ones being radial outward diffusion and competing parallel hydrodynamic flow along the fieldlines towards the targets. This renders the plasma edge transport problem (at least) two-dimensional, because characteristic times for perpendicular and parallel transport are comparable. Recycling of plasma ions, which recombine at the targets into the neutral state, complicate the problem by introducing large nonlocal source terms and by inducing genuinely 2-dimensional flow patterns (flow reversal). Several edge physics reviews, discussing these topics, have been presented only recently, (e.g. Neuhauser, 1989, Stangeby and McCracken, 1990, and Post and Behrisch, Ed., 1986).
Various 2-d plasma transport codes have been developed in the past, see e.g. Post *et al.* (1990, section 3.3) for a very recent compilation. In particular, isolated aspects, such as the divertor impurity (and helium) retention based on parallel force balance and the removal of neutralized helium particles from the divertor, have been addressed by numerical analysis very recently (Keilhacker *et al.* (1991), and Abou-Gabal and Emmert (1991)).
Probably the most widely used multifluid plasma edge model is implemented in the B2 code (Braams, 1987), which describes an electrically neutral and current free plasma, with each fluid component governed by a Navier-Stokes system of equations for density, momentum and ion and electron energy. Classical transport is assumed parallel to the magnetic field (Braginsky, 1964) and a mixture of anomalous and classical transport terms is prescribed radially. Coupling between species enters through ionization, recombination, interspecies friction, electric and thermal forces and temperature equilibration. A highly simplified divertor target recycling model is available, but often replaced by (or adjusted according to) more complex Monte Carlo code simulations.
An example for such a consistent 2-d multispecies neutral and plasma transport analysis is described next.
In these calculations we have used the EIRENE Monte Carlo code (Reiter, 1984) which follows the trajectories of neutral atoms, molecules and trace ions. The number and kind of species is kept optional, and the code communicates automatically with external atomic databases (used here: Janev et al., 1987, supplemented by a collisional radiative model for ionization of helium atoms in the ground level, denoted by (1|1S), and the two metastable levels (2|1S) and (2|3S) respectively (Fujimoto (1979) and Reiter *et al.* (1991)) as well as with databases for surface reflection models. In the case presented here, a recent enlargement of the TRIM code database (Eckstein and Heifetz (1989), and Reiter *et al.* (1991)) was used.

The EIRENE code is equipped with several options especially useful for the CPU time consuming procedure of iterating with a 2-d plasma edge code:

- various statistical procedures (collision and tracklength type estimators, allowing high efficiency over a wide range of neutral-plasma collisionality regimes.

- it can use a computer generated 2-d mesh, which may easily be supplemented by arbitrary *3 dimensional structures (pumping ducts, etc.).*
- correlation sampling technique, which freezes statistical fluctuations between iterations without introducing bias.
- computes profiles of arbitrary (user specified) moments of the neutral particle distribution function, such as density, collision rates, momentum and energy exchange rates, and all other profiles needed for consistent coupling to plasma fluid models.

Fully consistent coupling of the B2 code and the EIRENE code has been achieved, ie. both codes operate on the same mesh (provided by the "magnetic interpolator code LINDA" (Maddison *et al.*, 1991) and boundary conditions and exchange data in a fully automatic way (see fig. 12).

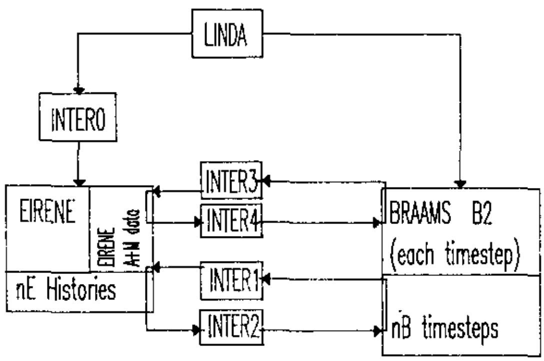

Fig. 12:
Schematic of multifluid 2-d plasma edge transport code system

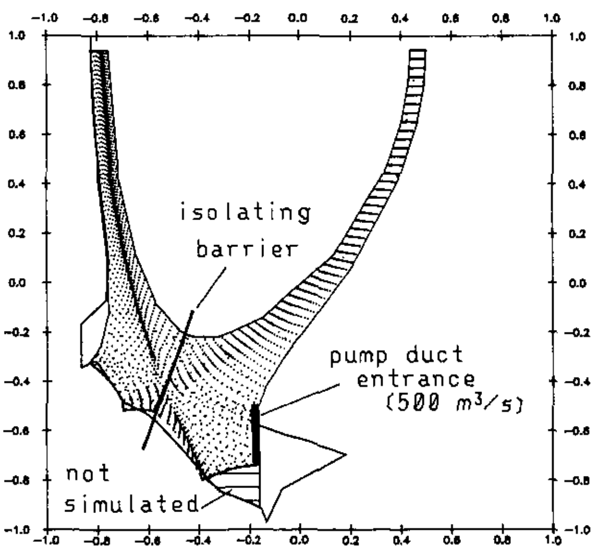

Fig. 13:
Lower half of two-dimensional SOL geometry (LINDA code), used for B2-EIRENE ITER modelling study

The neutral particles may leave the 2-d plasma grid (31 $\times$ 101 radial and poloidal cells, respectively) and travel outside in the 3 dimensional pump duct region or between the plasma and first wall.

In contrast to numerical experience with such coupled transport codes in lower recycling regimes (Gerhauser et al., 1988), in which convergence is fairly easy to achieve, the very strong neutral plasma interaction in front of the divertor targets, with flux amplification factors of several hundreds, creates many numerical booby traps (Reiter *et al.*, 1990c and Maddison *et al.*, 1991). The fluid code calls to EIRENE are automatically controlled by changes in the plasma profiles. In between such full Monte Carlo runs, and at each B2-timestep, only the EIRENE atomic data package is executed to recompute neutral source terms on the basis of the previous source term profiles and actual plasma profiles (see fig. 12: "implicit coupling technique"). Convergence can be monitored e.g. by the frequency of requests for full Monte Carlo runs, which can be quite high in the initial phase (almost every 5 timesteps) but finally becomes very low (ie. nB = many 1000 timesteps of 10^{-5} seconds in the B2 code without new Monte Carlo runs).

We discuss here the results from an ITER edge simulation study, employing the ITER specifications (Post et al., 1990) for an ELM-Y-H-mode edge transport model.

Aside from geometrical details this model is characterized by the following parameters (ITER operating scenario A1 "reference ignition"):

$\chi_{e\perp} = 2\ m^2/s$, $\chi_{i\perp} = D_\perp = \frac{2}{3}m^2/s$, $V_{pinch} = 0$, $n_e(a) = 3.5 \cdot 10^{19} m^{-3}$, $f_{He^{++}}(a) = 0.1$, $\Gamma_{He^+}(a) = 0$, $Q_e/Q_i = \frac{3}{1}$, $Q_{innerSOL}/Q_{outerSOL} = \frac{1}{4}$, $(Q_e + Q_i)_{SOL} = 116\mathrm{MW}$, Bohm target-sheath conditions.

Particle species accounted for in the plasma fluid model are D-T ions (effective mass number: 2.5), He^+ and He^{++} ions, and in the EIRENE test particle code: D, T, D_2, DT, T_2, D_2^+, DT^+, T_2^+, He(1|1S), He(2|1S) and He(2|3S).

Ionization and recombination rates for the helium ions are provided by a link to the atomic data files of the STRAHL impurity transport code (Behringer, 1987).

Starting from a fully converged single fluid (hydrogenic) solution obtained with a simplified analytical recycling model (Harrison *et al.*, 1991), about 100 full Monte Carlo runs (each with nE = 7500 and 15000 test particle trajectories launched from the inner and outer divertor target, respectively, and each run consuming about 45 CPU seconds on a CRAY X-MP/48) were needed before fully converged solutions were obtained (rather than about 5 – 10 Monte Carlo cycles requested for low recycling edge conditions as e.g. in TEXTOR-like open limiter configurations).

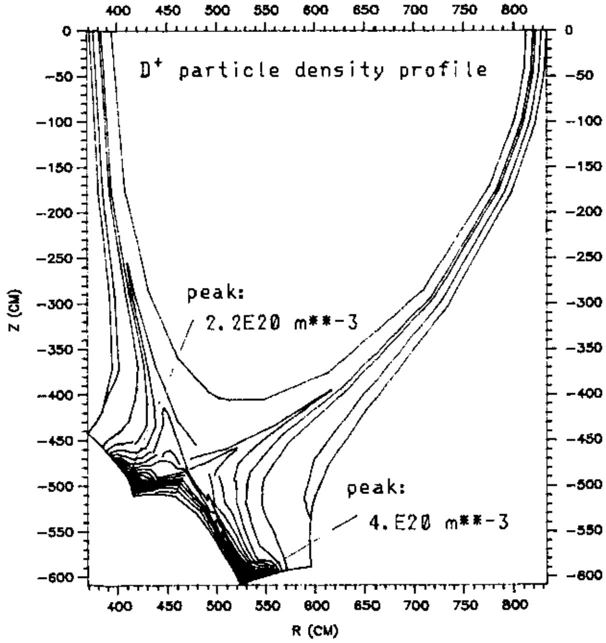

Fig. 14:
2 dimensional profile of D^+ density
(same as T^+ density).

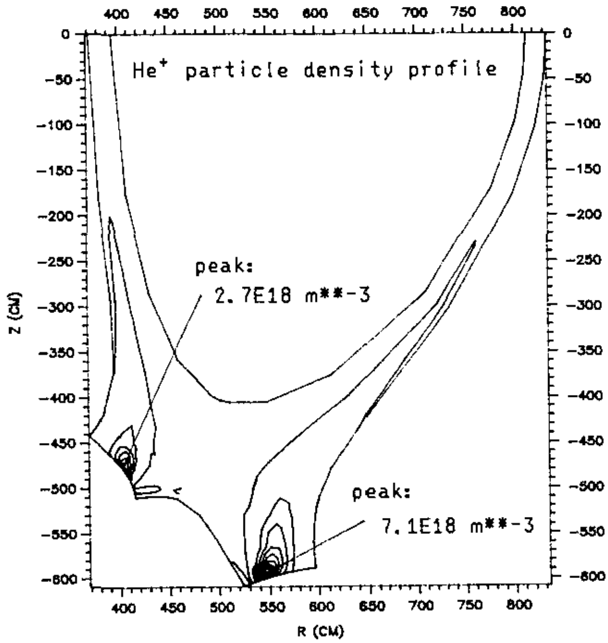

Fig. 15:
as fig. 14,
for He^{+} density.

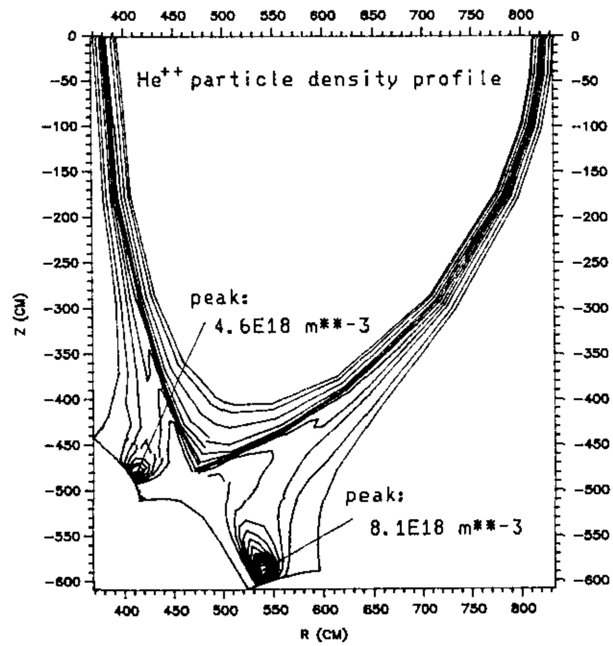

Fig. 16:
as fig. 14,
for He^{++} density.

The computations tend to confirm the estimates for helium pumping from an ITER edge plasma due to the efficient retention of particles, even though both targets have been assigned a recycling coefficient of one. I.e. different from previous ITER edge modelling studies, pumping (hydrogen or helium) is not treated by assigning an appropriate divertor plate porosity, but consistently by the Monte Carlo transport analysis of neutral particle flow towards the pumps more than 100 times during the time-evolution of the SOL plasma profiles. The only pumping occurs near the outer lower target, where a small part of the outermost radial surface is given a finite pumping speed of 500 cubic meters per second. Although the flux enhancement factors turn out to be smaller for helium than for hydrogenic ions ($F_{D-T} \approx 500$ and $F_{He} \approx 150$), the encouraging result is that we find similar pumping efficiencies for fuel and helium particles. This is because neutralized helium particles reach the pump more effectively (0.8% for helium and 0.1% for hydrogenic particles, of the target plate fluxes) and the SOL plasma is perfectly opaque for recycled neutrals of all species considered. Furthermore, similar to the hydrogenic profiles, a large concentration of helium near the target plates is found (fig. 14, 15, 16), and the peak target electron temperatures stay below 20 eV (inner target) and 30 eV (outer target).

The efflux from the core plasma into the SOL (equal to the pumped flux, since the transport model enforces particle balance) is about $2.2 \cdot 10^{20}$ helium particles/s, and $5 \cdot 10^{21}$ hydrogenic molecules/s. The fluxes to the outer target are $3.1 \cdot 10^{22}$ helium ions/s and $4.4 \cdot 10^{24}$ D-T ions/s, respectively. On the outer target the helium ion flux has its maximum about 5 cm further outward than the hydrogen flux, which is peaked near the separatrix strike point. Maximum helium ion densities near the targets are $4.6 \cdot 10^{18} m^{-3}$ (inner) and $8.1 \cdot 10^{18} m^{-3}$ (outer) for He^{++} ions, and $2.7 \cdot 10^{18} m^{-3}$ (inner) and $7.1 \cdot 10^{18} m^{-3}$ (outer) for He^{+} ions. These peaks arise at a distance of a few centimeters away from the targets, whereas the hydrogenic ions (with $4.4 \cdot 10^{20} m^{-3}$ (inner) and $8 \cdot 10^{20} m^{-3}$ (outer)) show a much sharper peaking, and at a distance of only 0.1 cm from the strike point.

Details of the flow to the pumps, however, should certainly not be inferred from such computations. The main defects still are the limitations to orthogonal target plates in the fluid model. Any "adjustments by hand" to account for inclined target plates reduces the physical consistency in the model.

One may speculate that for an inclined target model the flow towards the pumps may be enhanced. But probably at the same time the flux enhancement at the target and therewith the particle retention efficiency might be reduced. As pointed out earlier, the impact on overall helium removal is then described by the ratio of two small numbers ε_{He} and γ_{He}. Therefore, the final modelling answer to this question will have to be postponed until plasma fluid models are developed which can deal with more realistic boundaries.

The problem of possible helium enrichment in ITER pump ducts is particularly outside the predictive capacity of present day edge models, given its sensitivity to geometrical and plasma model details and at the same time the impossibility of assessment by stand alone Monte Carlo models (which do have the geometrical flexibility needed for realistic simulations). There is, however, modelling evidence that the various competing forces can cause decoupling between helium and hydrogen flow in the pumped divertor leg in such a way that helium removal might be assisted. This effect is a consequence of the larger mean free paths for neutral helium particles, which can penetrate the subsonic region outside the high recycling zone, in which the frictional coupling between D-T and helium flow is weaker.

6. SUMMARY AND CONCLUSIONS

Helium removal is a critical issue for all next generation tokamak experiments or reactor studies, which aim at significant fusion power selfheating. It has been shown that even under otherwise ideal conditions such as no impurities (except ashs) and 100% α heating efficiency, the helium particles must be removed from the plasma core within 10 to 15 energy confinement times. This makes it possible to relate minimum helium exhaust requirements to different confinement regimes.

Present day simulation experiments and numerical studies are addressing basically three questions with regard to this problem:

i: bulk plasma transport properties, characterized e.g. by the ratio χ/D of local thermal heat diffusivities to particle transport diffusivites, and by a density profile peaking factor C_v due to particle convection; MHD effects on transport, especially on ash accumulation.

ii: Helium removal from the edge plasmas, determined predominantly by the interrelation of parallel forces on the helium flow towards neutralizer plates, anomalous radial transport, and neutral helium recycling from the target surfaces, governed by atomic and surface physics.

iii: target plate and pumping duct engineering: the shape and orientation of target plates must not only match heat load constraints, but must also allow for sufficient particle flow towards the pumping duct entrances. These may be competing requirements, because, for efficient edge plasma cooling, neutral particles should also be confined in a well localized high recycling zone in front of the targets. Geometrical considerations are required to provide high vacuum conductances (and low backstreaming) in the vacuum ducts, which channel the neutralized particles to the pump locations.

1-d transport modelling studies for the core problem and 2 or 3 dimensional computations for the plasma edge and the pump ducts, based on present knowledge of confinement, parallel transport and atomic physics, have identified operational scenarios consistent with the ITER specifications, but also confirm the critical issues elucidated by the simple zero-dimensional estimates.
The suitability of pure H-mode conditions for a reactor is, due to the helium ash poisoning problem, at least highly debatable. ITER relies on "ELM-Y-H-mode" to degrade particle confinement. But empirical scalings for density profiles under such conditions, necessary for predictive computer modelling studies, are not yet established.

The plasma edge modelling studies have revealed the possibilities of obtaining ultrahigh recycling conditions in divertors. Under such conditions, both the target heat loads and the particle flow rates to the pumps are found to be in an acceptable range. The important coupling between neutral and charged particle transport in the near target region becomes numerically accessible, but geometrical restrictions in present-day 2-d plasma edge transport models (aside from physical uncertainties and approximations) are severe: only orthogonal targets have been investigated so far in a fully consistent manner. Furthermore, the credibility of the computed flow characteristics is diminished due to the lack of a computational linkage between the inner (unpumped) and outer (pumped) target plates.

Sufficient helium exhaust from the edge can be achieved with either

* high removal efficiencies (ε=10%), typical for pumplimiters, and little or no plasma edge screening.

  or

* lower removal efficiencies (ε=0.5%) but high edge screening against repenetration, as achievable predominantly in divertors operating at high densities.

This latter regime might turn out to be more vulnerable to bypass of particle flow to less well shielded components of the first wall or from unpumped divertor legs and, locally, even to flow reversal, whereas in the former case the leading edge heat load problem may restrict operation to short periods of time or to a position radially too far outside for sufficient particle flow into the limiter scoops. The achievability of helium enrichment in pump ducts depends critically on many configurational and plasma physics details and can neither be predicted nor be ruled out in general terms.
There are plans to experimentally investigate these issues in the next two years in the tokamaks JET, JT-60U, TFTR, DIII-D, ASDEX-U and TEXTOR.
Finally we would like to draw attention again to the simple example of the candle flame given in the introduction and the necessary conditions mentioned there for steady burning.

Whether or not circumstances in a steady burning D-T plasma flame will "by nature" or at least by active means be such that these requirements are met, can be addressed by predictive computer modelling techniques and validation by current and future experiments. However, past experience with extrapolations in the field of controlled fusion research, suggests that the relevant physics will finally only be sorted out by direct experiments.

REFERENCES

Abou-Gabal, H.H., Emmert, G.A., (1991). *Nucl. Fusion* 31, 407

Bateman, G., (1980). *"Distribution of Neutrals scattered off a Wall"*, Princeton Applied Physics Report No. 1

Behringer, K., (1987). *"Description of the Impurity Transport Code STRAHL"*, JET Report JET-R-(87) 08

Behrisch, R., Prozesky, V.M., (1990). *Nucl. Fusion* 30, 2166

Boucher, D., *et al.*, (1991). Proc. 18th Eur. Conf. Contr. Fusion and Plasma Physics, Berlin 1991, Vol 15c, part I, 177

Braams, B.J., (1987). *"A Multi-Fluid Code for Simulation of the Edge Plasma in Tokamaks"*, NET-Report 68, EUR-FU/XII-80/87/68

Bromberg, L. et al., (1979). *Nucl. Fusion* 19, 1359

Cohen, S.A. et al., (1990). *J. Nucl. Mat.*, 176 & 177, 909

Corbett, J., (1990). *J. Vac. Sci. Technol.* A8(3), 1772

Eckstein, W. and Heifetz, D.B., (1986). *"Data Sets for Hydrogen Reflection and their use in Neutral Transport Calculations"* Garching Report IPP9/59

Engelmann, F., (1980). Comments on Plasma Physics and controlled Fusion, 5, 261

Fujimoto, T., (1979). *J. Quant. Spectrosc. Radiat. Transfer*, 21, 439

Furth, H.P., et al., (1990) *Nucl. Fusion* 30, 1799

Gerhauser, H., *et al.*, (1988). *Contrib. Plasma Phys.* 28, 359

Harrison, M.F.A., *et al.*, (1991). *"Modelling of NET/ITER Edge Plasmas"*, NET-report EUR-FU/80/91-100

Hardtke A. et al., (1989). *J. Nucl. Mat.* 162 - 164, 661

Hawryluk, R.J., (1991). *Plasma Physics and Controlled Fusion*, this issue

Heifetz, O.B., et al. (1981). *" A Monte Carlo Model of Neutral Particle Transport in Divertor Plasmas"*, Princeton Report PPPL-1843

Hillis, D.L., (1990). *Phys. Rev. Let.* 65, 2380

Hillis, D.L., (Chairman) (1990). *"International Workshop on Helium Transport and Exhaust"* ITER report ITER/US/91/PH-13-02

Hogan, J., Hillis, D.L., (1991). Workshop summary report for *"International Workshop on Helium Transport and Exhaust"*, Gatlinburg, Tn, US, April 1991, submitted to *Nucl. Fus.*

Hu, S., Miley, G.M., (1991). *"Helium accumulation effects using benchmarked 0-d model"*, Draft, to be published.

Janev, R.K. et al., (1987). Springer Series on Atoms and Plasmas, Vol. 4

Kaufmann, M., et al., (1985). *Nucl. Fusion* 25, 89

Keilhacker, M., *et al.*, (1991a). *Nucl. Fusion* 31, 535

Keilhacker, M., *et al.*, (1991b). *Plasma Physics and Controlled Fusion*, this issue

Kolesnichenko, Ya.I., (1980). *Nucl. Fusion* 20, 727

Maddison, G. et al., Proc. 18th Eur. Conf. Contr. Fusion and Plasma Physics, Berlin 1991, Vol 15c, part III, 197

Meade, D.M., (1974). *Nucl. Fusion* 14, 289

Neuhauser, J., *et al.*, (1989). *Plasma Physics and Controlled Fusion* 31, 1551

Post, D.E., (1991) *ITER Physics Basis*, ITER Documentation Series No. 21, IAEA, Vienna (Draft)

Prinja, A.K., Conn, R.W. , (1985). *Nucl. Fusion*, 25, 1775

Rebut, P.H., *et al.*, (1990). *A Program towards a Fusion Reactor*, JET-Report JET-P (90) 75, to be published in *Physics of Fluids*

Redi, M.H. and Cohen, S.A., (1990a). *Effects of Particle Transport on Helium Ash Accumulation and Sustained Ignition in the ITER Design*, Princeton Report PPPL-2641

Redi, M.H. and Cohen, S.A., (1990b). *J. Nucl. Mat.* 176 & 177, 262

Reiter, D., (1984). *"Randschicht Konfiguration von Tokamaks: Entwicklung und Anwendung stochastischer Modelle zur Beschreibung des Neutralgastransports"*, JÜL-Report JÜL-1947

Reiter, D., *et al.*, (1990a). *Nucl.Fusion*, 30, 2141

Reiter, D., *et al.*, (1990b). *J. Nucl. Mater.*, 176 & 177, 756

Reiter, D., *et al.*, (1990c). *"Assessment of Neutral Particle Effects by Means of Monte Carlo Techniques"*, Final report on NET Contract 326/88-10/Fu-D, May 1990, unpublished

Reiter, D., *et al.*, (1991). *"Database for Recycling and Penetration of Neutral Helium Atoms in the Boundary of a Fusion Plasma"*, JÜL-Report, in preparation

Rose, D.J., Clark, M.,Jr., (1965). *Plasmas and controlled fusion*, M.I.T. Press

Rose, D.J., (1968). *"On the feasibility of power by nuclear fusion"*, ORNL-Report ORNL-TM-2204

Seki, Y., *et al.*, (1980). *Nucl. Fusion* 20, 1213

Shimada, M., *et al.*, (1981). *Phys. Rev. Let.* 47, 796

Sigmar, J., (1987). *Physica Scripta* Vol. T16, 6-15

Stangeby, P.C. and McCracken, G.M., (1990). *Nucl. Fusion*, 30, 1225

van Nieuwenhoven *et al.*, (1991). Proc. 18th Eur. Conf. Contr. Fusion and Plasma Physics, Berlin 1991, Vol 15c