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EXPERIMENTAL EVIDENCE FOR THE SUITABILITY OF ELMING H–MODE OPERATION IN ITER WITH REGARD TO CORE TRANSPORT OF HELIUM

ABSTRACT

Studies have been conducted in DIII–D to assess the viability of the ITER design with regard to helium ash removal, including both global helium exhaust studies and detailed helium transport studies. With respect to helium ash accumulation, the results are encouraging for successful operation of ITER in ELMing H–mode plasmas with conventional high-recycling divertor operation. Helium can be removed from the plasma core with a characteristic time constant of ~8 energy confinement times, even with a central source of helium. Furthermore, the exhaust rate is limited by the pumping efficiency of the system and not by transport of helium within the plasma core. Helium transport studies have shown that D_{He}/χ_{eff} ~ 1 in all confinement regimes studied to date and there is little dependence of D_{He}/χ_{eff} on normalized gyroradius in dimensionless scaling studies, suggesting that D_{He}/χ_{eff} will be ~1 in ITER. These observations suggest that helium transport within the plasma core should be sufficient to prevent unacceptable fuel dilution in ITER. However, helium exhaust is also strongly dependent on many factors (e.g., divertor plasma conditions, plasma and baffling geometry, flux amplification, pumping speed, etc.) that are difficult to extrapolate. Studies have revealed the helium diffusivity decreases as the plasma density increases, which is unfavorable to ITER's extremely high density operation.

1. INTRODUCTION

The fusion power generated in the plasma core of a fusion reactor such as the International TRhermonuclear Experimental Reactor (ITER) is integrally linked to the level of helium ash remaining in the core plasma, which for a given plasma density determines the number of fuel ions available for fusion processes. Hence, improvements in energy confinement (e.g., H–mode, VH–mode, and negative central shear (NCS) regimes) that are accompanied by even larger improvements in helium confinement are not necessarily favorable when extrapolating to a reactor. Studies on DIII–D in a variety of confinement regimes (including L–mode, H–mode, and VH–mode) have assessed whether helium buildup poses substantial problems in these regimes. The majority of these studies have been conducted in H–mode plasmas with edge localized modes (ELMs). The results obtained to date are encouraging for the successful operation of ITER in ELMing H–mode plasmas with conventional high-recycling divertor operation.

2. ITER REQUIREMENTS

Because the specific ITER requirements are difficult to scale to a present-day device due to differences in geometry, exhaust capacity, etc., this discussion will primarily concentrate on physics requirements, not specific engineering requirements. The helium exhaust problem is truly global in nature since the helium ash is generated in the plasma core and can only be removed from the system via pumping at the plasma edge. Since the source of thermal energy and the source of helium ash are the result of alpha particle thermalization, the energy and helium particle continuity equations (either the 0-D or 1-D forms) are coupled. Global transport studies have shown that in order to prevent unacceptable fuel dilution, the characteristic He removal time must be less than 7–15 energy confinement times, (i.e., τ^*_{He}/τ_E < 7–15), depending on the impurity content of the plasma [1]. Here, τ^*_{He} is the global helium confinement time and τ_E is the energy confinement time. In terms of local transport, one can combine the thermal and helium continuity equations (assuming that the alpha particles thermalize completely on the flux surface on which they were generated) to get the following equation for the helium density profile [2]:

$$\frac{dn_{He}}{dr} - \frac{V_{He}}{D_{He}} = -n\,\frac{\chi_{eff}}{E_\alpha D_{He}}\,\frac{dT}{dr} \qquad (1)$$

where V_{He} is the convection velocity for helium, D_{He} is the helium diffusivity, χ_{eff} is the single-fluid thermal diffusivity, E_α = 3.5 MeV is the alpha particle energy, and *n* and *T* are the plasma electron density and temperature, respectively. Note that this equation is only valid in regions where the recycling source is negligible. The primary parameters determining the helium density profile are the ratios V_{He}/D_{He} and D_{He}/χ_{eff} . Simulations of helium exhaust in ITER have shown that the helium content is extremely sensitive to the value of D_{He}/χ_{eff} with values of $D_{He}/\chi_{eff} \leq$ 0.5 being unacceptable [3].

3. HELIUM EXHAUST WITH A CENTRAL SOURCE OF HELIUM

Previous experiments on DIII–D with helium introduced via gas puffing at the plasma edge have shown that sufficient helium exhaust can be achieved ($\tau^*_{He}/\tau^*_E \approx 8$) simultaneously with good energy confinement in an H–mode plasma with ELMs [4]. To extend these results to a case with a central source of helium, as would be expected in a reactor, experiments with continuous helium neutral beam injection have been conducted on DIII–D. This central helium source coupled with simultaneous divertor exhaust provides a full simulation of the situation in a burning device. The DIII–D neutral beam system now includes the capability of 2.0 s steady-state He neutral beam injection (NBI). Divertor exhaust of helium is facilitated by condensing an argon (Ar) frost layer (~1.5 μm thick) on the liquid helium surface of the DIII–D divertor cryopump.

These experiments were carried out in a single-null divertor configuration with the ion grad-B drift towards the lower divertor, a plasma current of 1.0 MA, magnetic field of 2.1 T, and a major radius of 1.67 m. In these discharges (Fig. 1), the plasma density and temperature are held approximately constant during the period of interest through feedback control of plasma density via deuterium gas injection. Helium beam injection (3.0 MW at 75 keV) is initiated at 2.0 s and maintained for 1.4 s. The helium density in the plasma core, as measured by charge-exchange recombination (CER) spectroscopy, responds immediately to the initiation of helium NBI. Throughout the He NBI phase, the divertor outer strike point (OSP) is maintained in the optimal location for divertor exhaust. The time behavior of both the helium density and total core inventory of helium [Fig. 1(c)] are observed to follow the expected temporal evolution given by:

$$N_{He}(t) = N_{He}(t_o) + \left[S_{He}\tau^*_{He} - N_{He}(t_o)\right]\left\{1 - \exp\left[-\frac{(t - t_o)}{\tau^*_{He}}\right]\right\}, \quad (2)$$

where N_{He} is the total number of He ions in the plasma and S_{He} is the instantaneous He source rate. Least-squares fitting of the evolution of the helium inventory to Eq. (2) and standard energy balance analysis gives $\tau^*_{He}/\tau_E \approx 8.5$, well within the range necessary for ITER. The shape of the helium density profile remains essentially the same during the He beam injection phase after a brief transient. This observation and the flat profiles even in the presence of a central He source indicate high core transport rates and that transport does not limit the helium exhaust rate. This observation is corroborated by particle balance calculations based on CER and Penning gauge measurements, which show that the removal rate is determined by the equilibration time between the divertor plasma and the pumping plenum, which is observed to be on the order of 1.0 s, whereas the core-to-divertor equilibration time is ~ 100 ms. The enrichment factor γ (defined as ratio of the helium fraction in the pumping plenum to the helium fraction in the core) is in the range $0.2 \le \gamma \le 0.6$, meeting the present ITER design criterion that $\gamma \ge 0.2$.

4. HELIUM TRANSPORT STUDIES

Helium transport studies on DIII–D are carried out by injecting a small amount (~3%–5% of the electron density) of helium gas during an otherwise steady-state portion of the discharge. The helium density profile evolution subsequent to this puff is

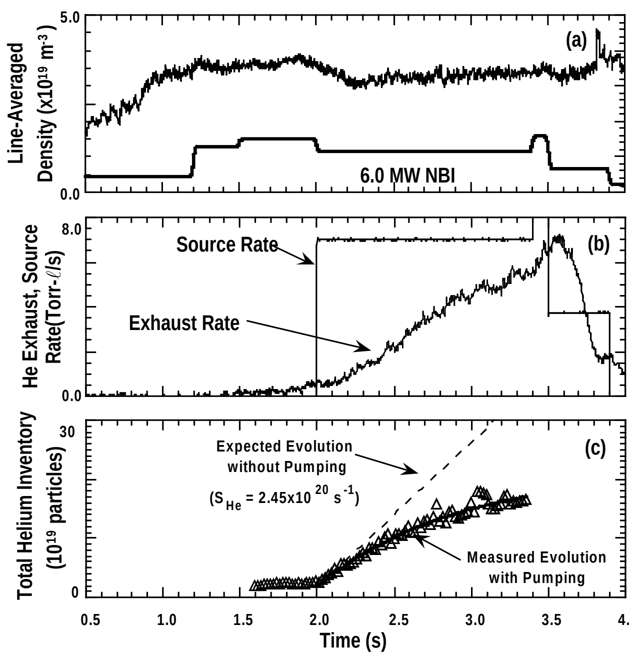

Fig. 1. Time evolution of the (a) line-averaged density, injected power, (b) helium source and exhaust rates, and (c) total helium inventory within the core plasma for a typical discharge with helium NBI. Helium NBI is applied starting at 2.0 s and maintained until 3.5 s. The predicted helium density and inventory if no helium exhaust were present are plotted as dotted lines in (c).

monitored by CER with a minimum time resolution of 5 ms. The local helium transport coefficients are then determined via regression analysis of the inferred local helium particle flux and the measured helium density gradients, assuming the flux takes on the general form, $\Gamma_{He} = -D_{He}\nabla n_{He} + V_{He} n_{He}$ [5].

As discussed in Section 2, the primary parameters of interest in helium transport studies are the ratios D_{He}/χ_{eff} and V_{He}/D_{He}. In general, the helium density profiles in ELMing H–mode plasmas are nearly flat in the inner regions of the plasma core ($\rho \leq 0.5$) . Hence, the parameter V_{He}/D_{He} is typically very small ($\leq 0.01\ \mathrm{m}^{-1}$). In all confinement regimes studied to date, the steady-state helium density profile has the same shape as the electron density shape, suggesting that preferential helium accumulation will not be a problem in ITER [5]. Studies on DIII–D have shown that $D_{He}/\chi_{eff} \sim 1$ for all confinement regimes studied to date including L–mode, ELM-free H–mode, ELMing H–mode, and VH–mode (Fig. 2). This suggests that the helium transport within the plasma core of ITER should not pose significant problems. However, helium transport and energy transport may scale quite differently from present-day devices to ITER. To address this issue on DIII–D, nondimensional scaling studies, which have primarily emphasized energy transport in the past, have been expanded to include helium transport studies. The premise in these studies is that the diffusivity (either thermal or particle) can be written in the form:

$$\chi = \chi_B \left(\rho_*\right)^{\alpha} F\left(\beta, \nu_*, q, R / A, \kappa, \ldots\right), \tag{3}$$

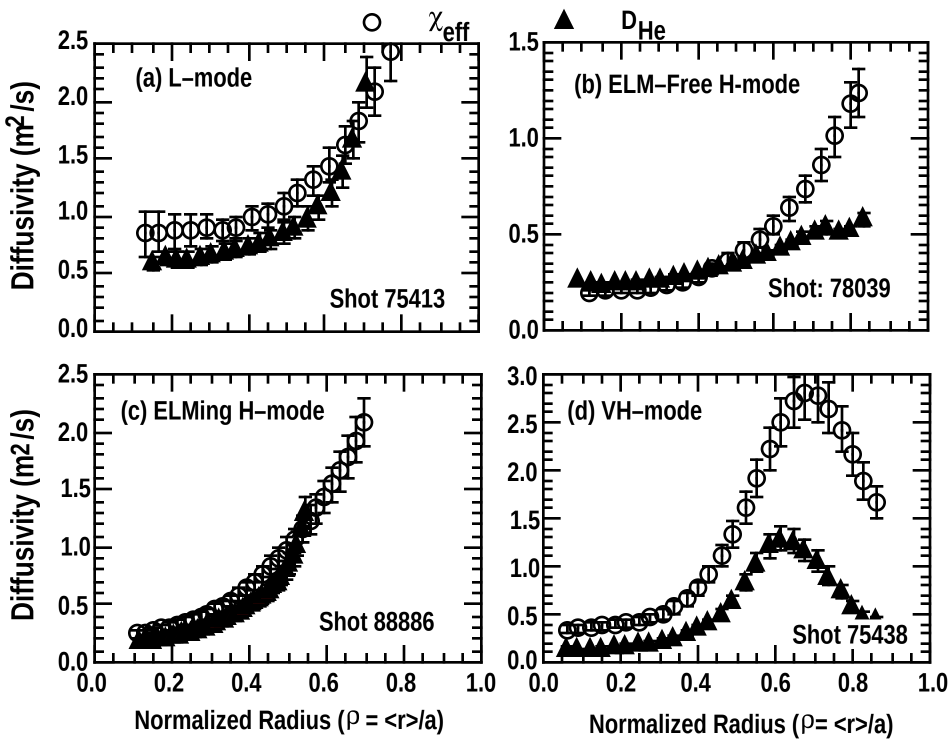

Fig. 2. Helium diffusivity (solid triangles) and single-fluid thermal diffusivity (circles) for (a) L–mode, (b) ELM-free H–mode, (c) ELMing H–mode, and (d) VH–mode discharge. Note that this data is from discharges with different plasma current, injected power, etc. so care should be taken in comparing the magnitude of the diffusivities.

where F is an arbitrary function of all of the relevant dimensionless parameters except ρ_*, the gyroradius normalized to the machine size. The diffusivity is normalized to the Bohm diffusion coefficient, $\chi_B \equiv cT/eB$, for convenience. The primary unknown in extrapolating from present-day devices to ITER is the dependence of transport on ρ_* since present-day experiments can operate at ITER-like values for all the other dimensionless quantities. Experimentally, one can determine the exponent α in Eq. (3) by varying ρ_* while holding F (or equivalently all the other dimensionless quantities) constant. Energy transport experiments of this kind in ELMing H–mode plasmas in the ITER shape and ITER-like dimensionless parameters on DIII–D have shown the scaling to be "gyro-Bohm"-like, with the characteristic transport scale lengths being on the order of a gyroradius [6, 7].

For the experiments discussed here, both the plasma shape (lower single-null divertor with κ = 1.68 and δ = 0.36) and the expected ITER parameters were approximately matched - β^{th} = 1.72 %, β_N^{th} = 1.67 % (MA/m-T), $\nu_{*i,min}$ = 0.01, and q_{95} = 3.8. To obtain a variation in the gyroradius of 1.5 while the other dimensionless parameters were held nearly constant, measurements were made in two cases: (1) B_T = 2.1 T, I_p = 1.14 MA, $n_e = 6.26\times10^{19}$ m^{-3}, P_{input} = 5.9 MW, and (2) B_T = 1.05 T, I_p =0.57 MA, $n_e = 2.74 \times 10^{19}$ m^{-3} , P_{input} = 1.2 MW . The helium diffusivity is inferred from analysis of the evolution of the helium density profile subsequent to a helium gas puff during an otherwise steady-state portion of the discharge. Note that because the propagation of the helium perturbation to ρ = 0.6 in the B_T = 1.05 T case is too rapid (~70 ms) to be followed accurately by the CER system, transport analysis of this discharge is limited to the interior regions of the plasma ($\rho \leq 0.5$). Using the premise of Eq. (3), the scaling of helium diffusivity with ρ_* is found to be approximately gyro-Bohm-like (Fig. 3). The transport analysis further suggests that inward convection (i.e., particle pinch) is considerably larger in the 2.1 T case outside ρ = 0.4. This would have the effect of decreasing the net transport rate in the 2.1 T case relative to the 1.05 T case. Therefore, if one considered the ratio of the net helium transport rate instead of simply the helium diffusivity, the scaling for $\rho \geq 0.4$ would be adjusted closer to gyro-Bohm. Energy transport analysis of the same discharges have shown the thermal diffusivity to scale with ρ_* in a similar manner as the helium diffusivity. Hence, D_{He}/χ_{eff} has at most a weakly linear dependence on ρ_*, which is favorable in the extrapolation to ITER.

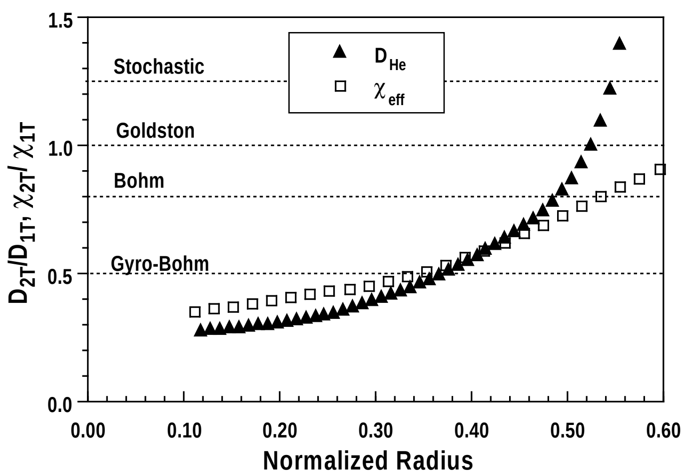

Fig. 3. Ratio of the helium and single fluid diffusivities inferred for the dimensionally similar discharges described in the text. See Ref. [7] for a more detailed description of the labels.

Scaling studies of the dependence of the helium transport coefficients with various parameters in ELMing H–mode plasmas have also been done. Studies in which the injected power was varied have shown that D_{He} and $|V_{He}|$ increase as the injected power is increased [5]. Both of these trends are favorable in terms of extrapolation to future, high-power devices. To assess the density dependence, helium transport data has been obtained during a density scan in ELMing H–mode in DIII–D in a lower, single-null diverted discharge with I_p = 1.0 MA, B_t = –2.1 T, q_{95} = 5.5. In the first case, the plasma density was maintained at 4.9×10^{19} m^{-3} (0.53 n_{GW} where $n_{GW} = I_p/\pi a^2$ (10^{20} m^{-3}, *MA*, *m*) is the Greenwald density limit [8]) with 6.5 MW of neutral beam power. In the second case, the density was reduced by approximately 40% to 3.2×10^{19} m^{-3} (0.34 n_{GW}) using the divertor cryopump on DIII–D, and the neutral beam injection power was reduced to 4.1 MW in order to maintain similar plasma temperatures in the two cases. Transport analysis shows that the helium diffusivity is approximately 50% lower in the higher density case. Energy transport analysis has shown that χ_{eff} changes only slightly in these cases, which is consistent with previous studies that have shown little dependence of global energy confinement on density [9]. No attempt was made in these experiments to maintain similarity in the dimensionless quantities. Therefore, it is possible that this variation is simply due to the parametric dependence of helium diffusivity on one (or even several) of the dimensionless quantities in Eq. (3), most notably collisionality. However, this data does present a concern because ITER seeks to operate at densities significantly above the Greenwald limit.

5. SUMMARY AND CONCLUSIONS

The results presented here are promising as far as helium ash removal in ITER is concerned. The successful demonstration of helium exhaust with a central source of helium in an ELMing H–mode plasma is encouraging, especially considering that the estimated exhaust efficiency is ~5%. An equally encouraging result is that D_{He}/χ_{eff} is found to be ~1 and there appears to be a weak dependence of D_{He}/χ_{eff} on normalized gyroradius, making the extrapolation less uncertain as far as core transport rates are concerned. These observations suggest that core transport rates for helium will be sufficient for ITER in ELMing H–mode plasmas.

However, helium exhaust is strongly linked to the exhaust capabilities of a particular device. The exhaust efficiency is dependent on plasma and baffling geometry, flux amplification, pumping speed, etc., which makes the extrapolation from present-day devices to ITER design dependent. Because of this reason, this review has not addressed the extrapolation of the results from detailed divertor studies of helium in DIII–D [10]. Also, the divertor plasma in ITER is expected to be in a detached state, characterized by low temperature and high density such that the mean free path against ionization of the helium neutrals recycling from the divertor surface will be quite long. Since this could lead to a reduction in the retention of helium in the divertor plasma., the helium exhaust results presented in Section 2 may not be nearly as favorable if done with a detached divertor plasma. In this regard, detailed studies of helium transport in the plasma edge and divertor plasma in combination with two-dimensional modeling are required to make a better assessment of the extrapolation of the present set of results to the specific ITER design.

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