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Critical plasma–wall interaction issues for plasma-facing materials and components in near-term fusion devices

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Abstract

The increase in pulse duration and cumulative run-time, together with the increase of the plasma energy content, will represent the largest changes in operation conditions in future fusion devices such as the International Thermonuclear Experimental Reactor (ITER) compared to today's experimental facilities. These will give rise to important plasma-physics effects and plasma–material interactions (PMIs) which are only partially observed and accessible in present-day experiments and will open new design, operation and safety issues. For the first time in fusion research, erosion and its consequences over many pulses (e.g., co-deposition and dust) may determine the operational schedule of a fusion device. This paper identifies the most critical issues arising from PMIs which represent key elements in the selection of materials, the design, and the optimisation of plasma-facing components (PFCs) for the first-wall and divertor. Significant advances in the knowledge base have been made recently, as part of the R&D supporting the engineering design activities (EDA) of ITER, and some of the most relevant data are reviewed here together with areas where further R&D work is urgently needed. © 2000 Elsevier Science B.V. All rights reserved.

1. Introduction

Material selection and plasma-facing component (PFC) design remain major challenges in the safe and reliable operation of the International Thermonuclear Experimental Reactor (ITER), and are perceived to be potential obstacles to the successful development of any long-pulse and steady-state deuterium–tritium (D–T) fusion reactor. Many plasma–material interaction (PMI) issues remain to be resolved. Often, the design features and parameters of ITER [1] as documented in the ITER final design report and a more recent reduced technical objectives, reduced-cost design option, so-called ITER fusion energy advanced tokamak (FEAT) are often used in this paper to exemplify issues that are generic to any reactor-scale next-step experimental facility.

In this paper we identify the most critical issues arising from PMIs which become key elements in the selection of materials and the validation of the design of PFCs of next-step devices. Although this paper surveys results mainly from tokamaks, investigations in fusion devices based on alternative magnetic confinement schemes (e.g., helical devices, stellarators, etc.) are also making significant contributions to this complex field.

The organisation of this paper is as follows. Section 2 discusses some of the most critical PMI issues for a next-step device such as ITER, and the expected changes are

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compared to present fusion research devices. Section 3 highlights some of the more recent advances made in understanding PMI issues, especially in relation to the next-step fusion device, and points the reader to appropriate literature where the subject is dealt with in more detail. Section 4 identifies the critical needs and suggests directions and priorities for future R&D. Finally, a summary is provided in Section 5.

2. Plasma reactor performance and technology issues related to plasma–material interactions

2.1. General considerations

The critical role that PMIs play in the achievement of the technical objectives of a reactor-class fusion device can be easily understood from the consequences that these interactions have on the plasma performance and the surrounding material surfaces. Among the critical material issues is the erosion of PFCs – e.g., limiters, divertors, and wall armours – that come in direct contact with the plasma. The erosion process limits component lifetime, and also leads to impurity transport into the burning plasma, affecting its performance. Another critical issue is the trapping, re-emission, and retention of tritium, affecting plasma fuelling and tritium inventory, which affects fuel availability and safety.

Although the final design of the next-step device is still the subject of debate, the main goal of such a tokamak will certainly be the safe demonstration of sustainable fusion burn and significant fusion power production in a magnetic confinement device. These criteria alone have allowed recent PMI research to focus on the likely issues to be important for any such device, even though these issues may have little or no consequence in today's tokamaks.

The longer pulse duration and cumulative run-time, together with the high-heat loads and more intense disruptions, will represent the largest changes in operation conditions in future fusion devices and determine by far the greatest consequences, and open new design, operation and safety issues. For a more detailed discussion see Ref. [2]. Present-day machines operate in short-pulse mode, with plasmas maintained for periods of the order of seconds, between which intervals of 10–30 min are typical. The power and particle loads are sufficiently small that they can be handled by making the PFCs, e.g., limiter modules or the divertor plates (on which the diverted field lines impinge), of materials such as graphite, and cooling these divertor plates between discharges. In addition, erosion of the main chamber and divertor strike plates acts as a source for impurities in the discharge but, mainly due to their very low-duty cycles, they have no impact on the material lifetime. These erosion/re-deposition effects are on the scale of $\sim\mu$m for current tokamak run cycles ($\sim$1000–5000 s/operation year).

In contrast, in a next-step experimental device such as ITER, the plasma pulse duration and the cumulative experimental run-time should be between a 100 and a 1000 times longer. Because of the longer pulse lengths, active cooling of PFCs during the plasma discharge is required. Based on current erosion rates, modelling shows that erosion of PFCs (by sputtering, chemical reactions, ablation and melt-layer loss) will be on a cm-scale after relatively long operation time in future devices, and this represents a 3–4 orders of magnitude change from present tokamaks, where net-erosion rates are barely measurable. Under the most pessimistic assumptions, the erosion lifetime of the PFCs becomes sufficiently short that several replacements, by remote handling procedures, will be required during the lifetime of the machine. Similarly, fuel economy has never been an issue in deuterium-fuelled experiments and only recently have the limitations associated with the use of tritium, and its incomplete recovery, in recent experiments in the Tokamak Fusion Test Reactor (TFTR) and in the Joint European Torus (JET) brought the issue of fuel retention under closer scrutiny. Trapping of tritium in re-deposited layers, particularly on cold regions hidden from direct contact with the plasma, represents potentially a very serious problem. Operation experience with tritium in TFTR and in JET pointed clearly to the anticipated problem associated with the formation of T-rich carbon co-deposited layers ($\geqslant$50 μm) during operations. That experience showed that without means of removing the co-deposited tritium, ITER operations could be quickly terminated due to safety and fuel economy reasons [3].

2.2. Critical plasma–wall interaction issues for PFCs

Recent experimental and modelling results have indicated four key problem areas that will greatly affect PMI design of next-step devices. They are (i) dispersal of the power and control of the particles and impurities to provide an adequate lifetime of PFCs and a tolerable plasma contamination ($Z_{\mathrm{eff}} \leqslant 1.5$); (ii) mitigation of off-normal events such as disruptions to reduce their severe effects on PFCs; (iii) minimisation of the tritium accumulation and development of efficient means of tritium removal; (iv) minimisation and control of the production of dust. A detailed discussion of these problems is beyond the scope of this paper and only a few considerations are outlined below. The interested reader is referred to Ref. [2] for further details and relevant bibliography.

In general, there is a strong coupling between plasma parameters and the interaction with the walls, both in terms of causes and effects, e.g., erosion/re-deposition and hence tritium co-deposition rates and locations. An important recent example is the choice of the so-called 'partially-detached plasma divertor regime' which is considered the most promising and favourable for an ITER-class tokamak [4]. A detached or partially-detached plasma allows for a large heat flux reduction at the divertor plate, sufficient helium exhaust, and greatly reduces the incident particle energy at the divertor ($T_e < 5$ eV), reducing or eliminating physical sputtering of even low-Z materials. However, in this case, chemical processes are expected to dominate carbon erosion and the complex transport/dissociation of hydrocarbon molecules in the plasma must be understood to predict erosion and deposition rates. An additional uncertainty is added due to the expected (but not well understood) flux dependence on the carbon chemical erosion yield [5,6]. This changes the physical picture of carbon erosion/re-deposition considerably compared to the fairly well understood physical sputtering, atom/ion transport dominated in attached plasmas. For example, attached-plasma divertor scenarios, with high local re-deposition probability, are favourable (for low-tritium retention), whereas detached scenarios are unfavourable (resulting in high-tritium retention), since the resulting low-temperature plasmas allow a larger fraction of hydrocarbons formed at the target to escape and deposit on cold surfaces.

At the present time, there is much attention given in the fusion community to so-called 'advanced scenarios'. These scenarios, generally aimed at providing improved energy confinement in the plasma core, usually involve operating at lower plasma densities than envisaged, for example, in the ITER design [1]. There are clear implications for the first-wall in operation at lower density, including increased power densities at the divertor targets, possibly reduced interactions with the remainder of the first-wall, and less efficient removal of fuel and helium ash. Furthermore, the divertor becomes increasingly the primary source of plasma impurities in comparison to the main chamber wall, and recycling is also affected. It is possible that such advanced scenarios also require novel concepts to handle PMIs.

2.2.1. Power dispersal, particle and impurities control

Operation of the next-step tokamak experiment will require effective means to disperse the thermal power and to attain a sharp reduction in divertor heat flux to a level such that material surfaces can be designed for (i.e., about 5–10 MW m^{-2}) without causing deterioration of the bulk plasma from the highest performance conditions. The current strategy to achieve power dispersal is to convert most of the heat flux to impurity radiation in the so-called 'core-mantle' region – i.e., the outer periphery of the core plasma, as well as radiation from the scrape-off-layer (SOL), and re-distributing that heat flux over the relatively large side-wall area of the divertor region. This could be achieved, for example, by introducing impurity noble gas ions, such as neon, argon, or krypton (via controlled feedback loops for either pellet injection into the core plasma or gas-puff into the SOL). Power dispersal by impurity radiation has been successfully implemented in today's tokamak experiments. The so-called 'vertical target' adopted in the ITER design (see Fig. 1) provides an optimal geometrical configuration because it directs neutrals reflected off the divertor plate towards the separatrix field line, thus increasing the level of recycling.

The erosion lifetime of the divertor component subject to most severe plasma interactions is dependent on the maximum allowable thickness for the armour material and the net-erosion rate (material loss rate) due to sputtering and disruption vaporisation and melting. The predicted lifetime varies with the plasma edge conditions and the selected plasma-facing materials; (PFMs) the results for ITER are discussed in Refs. [7,8]. Based on these results and taking into account the combined effect of sputtering, disruptions and the so-called slow transients, the lifetime for ITER [1] would be below 10^5 s for Be at 10 MW/m^2, more than 2×10^6 s for carbon-fibre-composites (CFCs) and 1.5×10^6–4×10^6 s for W. Although there are still significant uncertainties involved in these estimates, particularly for detached plasma conditions, the results of these analyses are nonetheless sufficient to illustrate trends and to guide the current material selection.

As far as the plasma contamination is concerned, impurity production rate is not the only consideration. Equally important is the efficiency of impurity transport into the core plasma, which depends critically on location. It has been found, for example, that while local divertor impurity sources strongly affect the radiation in the divertor, for the core there is an additional contribution to impurity levels from first-wall surfaces that are physically closest to the core (e.g., protection limiters, antennas and other protruding parts) which receive considerable ion fluxes. In ITER, modelling shows that the core plasma is highly sensitive to neutral influx and the divertor is designed in such a way that maximum isolation is achieved between the highly radiating divertor with its high neutral density and the main plasma chamber, so as to attain the best plasma performance. In contrast to the divertor, erosion rates at the wall may be low enough that the PFCs do not need replacement [9], but because the area is larger, the total amount of eroded material may be significant. This material must go somewhere, most likely to the divertor, and this might affect the composition of the divertor surface and therefore affect the divertor performance. Also, there is some concern on the effects resulting from the impacts of energetic charge-exchange particles on the first-wall surfaces which can result in mechanical degradation of surfaces, production of dust, and also affect the tritium inventory [10]. However, for metals such as beryllium, such microdamage was found to

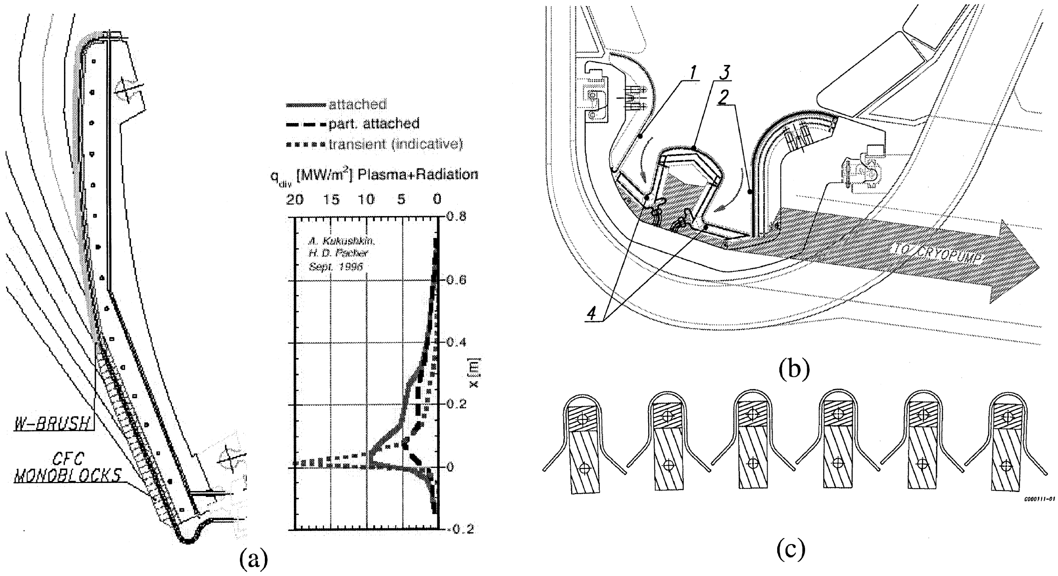

Fig. 1. (a) Vertical target and power loads on the ITER vertical target (plasma, neutrals, radiation) for typical attached and partially-detached divertor plasma cases; (b) cross-section of the reduced-size ITER FEAT divertor: (1) inner vertical target, (2) outer vertical target, (3) dome, (4) pumping duct liner; (c) detail of the radiative pumping duct liner.

have a favourable effect in reducing the implanted tritium inventory (see Section 3.2).

2.2.2. Disruption damage and mitigation issues

Disruptions occur frequently in today's tokamaks, chiefly due to their experimental, exploratory nature. Although the in-vessel components are designed to withstand structural integrity and meet general operational availability goals, the erosion damage is generally non-critical. This will no longer be the case in the next burning plasma device, mainly because of the large increases anticipated in both plasma current and the energy loss during a disruption. As an example, the 'specific energy', i.e., the plasma thermal energy divided by the surface area of the plasma, which provides a measure of the severity of the plasma–wall interactions in disruptions, will be up to 10 times larger in ITER than in existing large fusion machines. Major disruptions and vertical displacement events (VDEs) in ITER will then cause ablation and melting of surface material in the divertor target area and possibly a good fraction of the divertor baffle and first-wall. Calculations and experiments in plasma simulators show that a vapour shield should form in front of the divertor targets, dispersing the majority (>90%) of the incident energy flux to the divertor chamber walls via radiation which, in turn, could cause melting or sublimation on the nearby-surfaces. However, there are still large uncertainties in determining the erosion associated with disruptions, and the effectiveness of the vapour shield to control the power on the surfaces.

Disruptions and VDEs can also give rise to the production of runaway electrons with multi-MeV energy. In present experiments, the magnitude of runaway generation varies, from none detectable to up to $\sim$50% of the initial plasma current. The latter levels would be high enough to cause surface damage to the affected region and repeated runaway strikes would likely mandate repair or replacement of the affected surface.

The incentives for disruption avoidance are considerable and include both conservation of the finite operational lifetime of ITER in-vessel components affected by disruptions, and avoidance of the loss of operational time that wall reconditioning may entail. However, despite these significant incentives, disruption avoidance in tokamaks in general, and in ITER in particular, remains as a goal to be achieved, particularly in operational regimes close to known operation limits – density, beta and marginal radiation energy balance, for instance.

Therefore, when designing PFCs, there is an important need to distinguish between events that will be handled by avoidance versus events that must be included in the design. At present, the ITER divertor is being designed to accommodate disruptions for lengthy operating periods before replacement will be required. For this reason carbon has been selected near the strike points to withstand the power loads of disruptions without melting. This situation might lead to a generation of special limiters to be used during the early phase of ITER that are to be robust against runaway electrons of VDEs but lack other features (e.g., long lifetime, low-tritium inventory, etc.) that will be needed in later phases.

Ref. [11] details the extensive progress which has been made in understanding and quantifying the impact of disruptions on an ITER-scale device and gives the status of disruption mitigation studies. The injection of a solid impurity pellet composed either of moderate-Z or high-Z materials (commonly termed a 'killer pellet') into a tokamak plasma is considered, as it is capable of effecting a non-disruptive fast plasma energy and current shutdown. However, studies indicate that injection of high-Z impurities leads to the generation of potentially damaging large runaway currents. Deuterium injection avoids, in principle, this problem but the amount of injected material required is very large. Recently, massive bursts of helium gas have also been used to mitigate the deleterious effect of VDEs and density limit disruptions in the DIII-D tokamak [12]. Due to the high plasma densities achieved, no runaway electron production was expected and none was observed. Most of the energy flowing to the divertor is seen to be dissipated as radiation in the main chamber. The simplicity and expected reliability of this technique makes it a strong candidate for disruption mitigation in next-step devices. The injection of hydrogen or helium liquid jets has also been proposed as a new concept for disruption mitigation and fast shutdown in tokamaks [13]. Liquid jets can rapidly cool the plasma to reduce divertor heat loads and large halo current forces, while simultaneously raising the density, sufficiently to prevent runaway electron generation.

Unlike disruptions, which one eventually plans to eliminate (or sufficiently mitigate) in a working reactor, edge localised modes (ELMs), which result in impulsive flow of thermal energy and particles to localised areas of the surrounding material surfaces, may be necessary in order to provide particle and impurity control (their most useful function in current tokamak H-modes). Work is continuing on the development of techniques to control ELMs and reduce peak heat flux to PFCs, while maintaining the core plasma parameters required for ignited plasma [2,14]. Operation which results in type-I and/or giant ELMs must be avoided because the energy density of such events would surpass the most severe disruption in current tokamaks, and would, in contrast, occur every second or so (e.g., current projections show that ITER type-I ELMs liberate $\sim$ 50–200 MJ m^{-2} s$^{-1/2}$). This is a sobering number considering that ITER-class disruptions liberate > 1 GJ m^{-2} s$^{-1/2}$. Both will well surpass the ablative limit of carbon-based materials of 40–45 MJ m^{-2} s$^{-1/2}$.

2.2.3. Minimisation of in-vessel tritium retention and control of the tritium inventory

Tritium retention in PFMs and control of the in-vessel tritium inventory have emerged as primary concerns for next-step fusion devices and future fusion reactors fuelled with mixtures of D–T, with strong implications for the in-vessel component design, materials selection, operational schedule and safety.

Tritium retention and the control of the tritium inventory in ITER and future reactors strongly depends on the choice of PFMs and their operational conditions (e.g., temperature, flux density of impinging particles), plasma edge conditions, and geometry effects (e.g., gaps, shaded regions, etc.). On the basis of recent experimental and modelling results (see for example, [3,8]), it can be concluded that as long as C-based materials are used, even in limited regions of in-vessel components, the dominant mechanism for retention will be co-deposition of tritium with eroded carbon in colder regions of the divertor system. Retention by other mechanisms such as implantation and surface absorption, which may be significant for small short-pulse machines, is expected to rapidly reach saturation in ITER. To date, there are still large uncertainties in quantifying the in-vessel tritium inventory of a device like ITER. These arise mainly from the uncertainties of the plasma edge physics parameters, which strongly affect the erosion, deposition and co-deposition patterns and rates. Moreover, mixed-material effects, arising from the simultaneous use of different PFMs to protect the different parts of in-vessel components, introduce significant uncertainties for the operation of a tokamak like ITER. Nevertheless, it has become clear that the rate of tritium build-up in JET (and even more so in TFTR) is already too high for a next-generation fusion device. Such a machine will have to choose PFC geometry, materials and temperatures so as to minimise tritium co-deposition and to minimise hydrogenic content of deposited films. The ongoing analysis of present tokamak experience will be key in aiding such choices. An emerging concern is the formation of carbon films in cold regions hidden from the plasma [2,3,15] that can trap a significant amount of tritium that is very difficult to remove. A new concept of a radiative 'hot liner' has been proposed for the private region of ITER to minimise the tritium accumulation on the surfaces which will be kept hot (1000–1500 K) during normal operation [16] (see Fig. 1). An R&D programme is underway to validate and quantify the underlying processes.

As long as carbon is used even on limited areas of a next-step device, safe operation and fuel economy will necessitate the periodic removal of the tritium from the co-deposited layers, or perhaps the removal of the layers altogether, unless keeping the surfaces where co-deposition is expected to occur 'hot' can prevent the accumulation of tritium. Although several alternatives are being considered for the removal of the T-rich co-deposited layers, their removal from a next-step machine using carbon is a major unsolved problem. Techniques involving exposure to oxygen (e.g., thermo-oxidative erosion at temperatures above 570 K, or oxygen plasma discharges) have been found to be most effective in laboratory experiments to remove T from a carbon surface (by removing the T-containing films) [17,18]. Major drawbacks of techniques using oxygen, especially at elevated temperatures include collateral effects on other in-vessel components, and recovery time for normal plasma operation. No practical method of localising the oxidation to the area required (and avoiding oxygen exposure elsewhere) has been developed, although various ideas are being explored. Alternatively, high-temperature baking (>1000 K) under vacuum is sufficient to remove the trapped tritium, but is technically very difficult to achieve.

If carbon-based materials were to be eliminated from the divertor of a next-step device, the situation as far as tritium inventory is concerned could be radically different and the control of the tritium inventory much more manageable [19]. However, dust from other materials such as beryllium and tungsten might still be a safety issue. In the light of recent developments of copper-backed W which survive steady-state power loads of 15–30 MW m^{-2} [20,21], the primary candidate in lieu of carbon for high-heat flux regions is tungsten. However, the ability of tungsten to withstand the high-heat flux during transients without suffering damaging melting is as yet unclear and would require disruption mitigation. The predicted need for carbon PFCs, and the consequent allocation of a significant fraction of the operational schedule for detritiation, follows directly from the projected levels of thermal loads expected during attached-plasma transients and disruptions.

2.2.4. Minimisation and control of the production of dust

The subject of dust has in the past received little attention within the fusion community, mainly because dust is neither a safety nor an operational problem in existing tokamaks. The ITER design has highlighted the fundamental need to deepen understanding of the mechanisms, which lead to the production of dust in tokamaks, since this may affect directly the safe operation of a next-step device. The amount of dust generated in a next-step device is likely to scale up by 2–3 orders of magnitude along with the erosion and discharge duration. Dust represents a potential safety hazard for ITER because it can in principle cause steam- (or air) induced hydrogen explosions (e.g., the basic reaction for beryllium: $Be + H_2O \rightarrow BeO + H_2 \;\; -370$ kJ mol^{-1}), thereby increasing the spread of radioactive debris mobilised during an accident involving a sudden vent [22]. A review of the existing database is discussed in Ref. [2]. However, simply measuring the in-vessel dust inventory was demonstrated to be a challenge in existing machines and there are still large uncertainties associated with production mechanisms and rates and extrapolation from present machines to the next generation of tokamaks. Research into dust production mechanisms and rates and their biological interactions has just begun [23,24]. Flaking and break-up of films, resulting from re-deposition of sputtered and vaporised material, are expected to be the primary sources of dust in a device like ITER. Droplet ejection during arc discharges between the plasma and the wall surface may also play an important role, but investigations are needed to better quantify the effects. This dust is likely to be localised in the divertor or at the bottom of the vessel. The effect on dust generation from mixed-materials is also unknown.

3. Highlights on recent advances in plasma–material interaction studies

A brief mention is made below to some of the most recent results in PMI research which are relevant to the design of a next-step machine and have enabled a significant advance of the knowledge base.

3.1. Erosion/re-deposition of wall materials

(a) The release of wall material into the plasma, both by physical and chemical means, is now better understood. In divertor tokamaks, the first-wall is an area of net-erosion whilst both net-erosion and deposition occur in the divertor [2].

(b) In several tokamaks the net-erosion/deposition and associated D retention is asymmetric with respect to the inner and outer strike-points. This, for example, is the case in JET, DIII-D, ASDEX Upgrade [25] and Alcator C-Mod [26]. The outer strike-point is generally a region of net-erosion, whereas net-deposition is seen on the inner divertor. Plasma-edge modelling indicates that this could partly be due to higher average electron temperatures in the outer divertor which results in higher ion impact energies and higher sputtering yields. However, JET data suggest that there is a contribution from anomalous drift in the SOL from outboard to inboard. This drift has been experimentally observed and needs further investigation [27].

(c) Because of the use of carbon near the strike points of ITER, and the necessity to operate the next-step machine in detached plasma conditions, chemical sputtering of carbon-based materials has recently been the target of vigorous investigation. Experiments have identified the overall trends of chemical sputtering yield with a wide variety of plasma and surface parameters: surface temperature, incident species and energy, and incident flux [2]. However, the experimental observation of an apparent reduction in chemical erosion yield with increasing incident flux requires further investigation. In particular, the extents to which parameters other than flux (e.g., energy, re-deposition, photon efficiency in spectroscopic measurements, etc.) affect the observed erosion rates need to be better determined. This is of particular importance to ITER-class divertors that will have an order of magnitude higher flux than current experiments.

(d) The complex role of molecular neutral and radical states and their transport/interactions with the plasma is now coming to light, posing a substantial challenge to PMI modelling. An additional complication is the recent realisation that material mixing (e.g., first-wall metal sputtering depositing on a carbon divertor plate) will be important in future devices that use more than one kind of PFM. Ongoing experimental and theoretical work on issues such as chemical erosion and material mixing are needed to make confident extrapolation of PMI parameters (and their consequences) to the next-step.

(e) The ability to diagnose PMI in tokamaks has also recently improved. Particularly useful has been the installation of dedicated probes for material studies in large Tokamaks (e.g, DiMES on DIII-D, ASDEX-Upgrade, TEXTOR). These are important as they provide unique benchmark tests for PMI models. The results from these diagnostics, in combination/comparison with more traditional long-term sample/probe results, have helped to illuminate key PMI issues (e.g., high-net-erosion near divertor strike-points), pointed to strengths and weaknesses in PMI modelling, and generally raised the profile of PMI issues within the fusion community. Ongoing refinement and/or development of remote time-dependent PMI diagnostic techniques (e.g., spectroscopy, surface interferometry) will continue to be critical in advancing the understanding of PMI physics and chemistry.

(f) Dedicated in situ experiments in tokamaks have been performed in some cases and could distinguish gross-erosion measured spectroscopically from net-erosion as measured from time resolving surface probes exposed to divertor conditions (e.g., as in ASDEX-Upgrade and DIII-D) [2]. However, there are still some uncertainties with interpretation of the results, especially for high-density, low-temperature detached plasma regimes where neutrals play an important role and conventional plasma diagnostics like Langmuir probes and spectroscopy do not work very well. In this case the recycling of neutral atoms, radicals and molecules, including impurities between the surface and the plasma edge, needs closer scrutiny.

(g) Re-deposited layers in some tokamaks with carbon-based targets behave differently with respect to sputtering of the original material. An increase in sputtering yield would result in a more rapid transfer of material from the erosion area into regions of net-deposition, where trapping of H, D, T may occur. In JET, this effect contributed to the heavy (flaking) deposition on the inner louvres. In long-pulse machines the increased sputtering leads to increased target erosion and T retention. However, oxidation rates are also increased, which may assist methods of local removal of the films. See Ref. [2] for further details.

(h) High-Z metals have been successfully used as PFMs (as example, see the review in [28]). Operation with relatively clean core plasma has been demonstrated in Alcator C-Mod with an all-molybdenum wall [29] and in ASDEX-Upgrade with tungsten divertor plates [30]. Sputtering of high-Z materials by deuterium is strongly reduced due to threshold effects, and impurity ion sputtering dominates the erosion rate. It is now realised that the proper selection and location of the PFCs, combined with modern plasma control capabilities, allow us to take advantage of the low-sputtering yields of these materials. The use of high-Z materials in ITER to replace carbon in areas exposed to off-normal thermal transient events (e.g., disruptions) requires development of reliable disruption mitigation techniques, and R&D in this area needs to continue with high priority.

3.2. Tritium retention and control of the inventory

(a) Operational experience in today's Tokamaks show that deuterium and tritium are retained in large quantities with carbon PFCs, mainly due to co-deposition with eroded carbon, e.g., 40–50% in large limiter machines (TFTR, JET pre-1992) and ~15% in JET post-1992 (divertor operation with an integral cryopump). These levels of retention would seriously restrict operations of a next-step device.

(b) In divertor tokamaks, co-deposition occurs mainly in the divertor. This is true even when the divertor material is not carbon, as with the JET Mk-I beryllium divertor [31] or the ASDEX-Upgrade tungsten divertor [30], as other carbon PFCs provide a source of carbon for co-deposition.

(c) Intense co-deposition of carbon and deuterium is found in many tokamaks in regions which are shaded from ion flux but are near carbon surfaces receiving high-ion flux [2]. Since ions cannot reach these shaded surfaces, this carbon deposition must be due to neutral carbon atoms or molecules resulting from dissociation of hydrocarbons. JET has shown that deposition on cold surfaces shaded from the plasma can produce deposits with D/C of ~0.7, and these deposits retained much of the T trapped after the D–T campaign (DTE1).

(d) Deuterium retention on the wall of the main plasma chamber is at the level expected from implantation by energetic charge-exchange neutrals from the plasma. This main chamber wall inventory does not greatly contribute to long-term deuterium inventory because the thickness of the implanted layer is small (<0.1 μm). However, the quantity of deuterium retained in the main chamber walls is much larger than the quantity of D in the plasma and the dynamic variations of this wall inventory dramatically affect fuelling of individual discharges.

(e) Tritium retention in beryllium is expected to be less serious than previously anticipated. Recent implantation experiments at low-energies, high-fluxes and high-particle fluences in beryllium [32] showed that after reaching fluences of about 10^{22} atom m^{-2}, further implantation results in a negligible increase in tritium inventory. Although details of mechanisms responsible for this effect are only partially understood, the creation of surface-connected porosity that provides a rapid return path back to the plasma seems to be important at a high-implantation flux. This process has significant implications in a next-step device lined with Be because it limits the inventory and permeation rate in surfaces exposed to high-particle fluxes.

(f) The methods proven effective for removing tritium from C involve oxidation of the co-deposited layer or physical removal, which are expensive to implement and may produce collateral damage. From the tritium recovery experience recently gained from JET and TFTR and with deuterium in TEXTOR [2,3], it can be concluded that the use of hydrogen, He/O and/or reactive gas or glow discharge cleaning plus atmospheric flushing is also marginally effective in removing significant quantities of the accumulated in-torus tritium inventory, and thus achieving the required regulatory limits. In JET and TFTR the tritium removal took place over several weeks, i.e., considerably longer than the total cumulative duration of high-power DT discharges. By comparison, the tritium removal rate in ITER will need to be orders of magnitude faster when considering the reactor's operational schedule.

3.3. Modelling and projections to reactor conditions

Remarkable advances in computational power have facilitated the development of detailed models which have proved to be vital tools in the design of a next-step reactor. An additional advantage of these codes is that more reliable plasma solutions are now available to erosion models for a next-step tokamak, along with more detailed understanding of the edge plasma in current experiments.

(a) Erosion/re-deposition modelling for ITER shows that the peak net-erosion rate due to chemical sputtering at the target is of the order of 5–15 nm s^{-1}.

(b) Tritium retention in the torus of carbon-containing reactors will be dominated by co-deposition in carbon, and the predictions for the 1998 ITER design are $\sim$2–20 g-T/1000 s pulse [8].

(c) Understanding the physics of material response during off-normal heating has improved and now complex modelling tools are available. However, extrapolation to reactor conditions remains somewhat uncertain, and the modelling of the effect of vapour shielding and the stability of the melt-layer must be further developed and tested against experimental results to better understand the complex interactions at work.

4. Further research and development (R&D) needs

Important results have been obtained in the area of plasma–wall interactions during the last decade from operation of fusion facilities and experiments in laboratories around the world. These, together with advances in modelling, have greatly expanded our knowledge on plasma–wall interaction processes in a fusion environment, for conditions of direct relevance to the design and operation of a next-step fusion experiment. However, there are still several areas of uncertainty and a variety of challenging issues remain to be urgently resolved. The topics of highest priority are summarised in Table 1. This requires an urgent coordinated research and development (R&D) effort, involving extensive participation by all parts of the fusion community.

Further development and optimisation of critical components, e.g., the divertor, based on information provided by future dedicated operational experience in existing tokamaks, will be useful to control and to mitigate PMI processes. Some considerations are being given to solutions that could mitigate co-deposition in the critical areas of the divertor (e.g., by ensuring that regions of probable deposition are kept 'hot' during operation, leading to reduced tritium retention, or by means of 'cold catchers' which could be periodically heated to recover the tritium). Prudently, new design solutions without using carbon are also being explored. The primary alternative material for high-heat flux regions is tungsten. However, the primary shortcomings of W that should be addressed in the coming years are the lack of operational experience and the dearth of experimental data regarding formation of melt-layers (and their properties) during disruptions. Other goals in this area are the minimisations of thermal quench frequency (disruption control) and the reductions of the thermal quench energy deposition through dissipative methods. Operation at higher plasma edge temperature and lower density may lead to further problems that need to be explored.

5. Summary

The experience of today's tokamaks, and the development of sophisticated models, have provided a bridge to designing the ITER device and predicting and optimising its performance. The ITER design has augmented the perception of the PMI problems and highlighted the fundamental need to deepen understanding in certain areas. The long-pulse duration will be the most significant change from present machines. Erosion of the divertor will be severe and force periodic replacement, by remote handling, of the PFCs. Tritium will be co-deposited together with eroded carbon, and

Table 1
Research and development areas of highest priority

(A) *Modelling of tokamak data:* • Enhance the modelling activity of plasma-wall interaction effects (e.g., the erosion/re-deposition patterns and rates; the amount of D(T) retained in co-deposits; the mechanisms and parameters controlling the formation of films/flakes and the retention of D(T) for divertor configurations tested in present-day tokamaks) and in particular to develop a quantitative understanding of mechanisms and develop models that can be validated by comparing the predictions with that experienced over the wide range of experimental tokamak conditions. • Provide new edge and wall measurement capabilities (see point C) to be able to tie in better with models; conduct edge interpretative work in tokamaks as a basis for model development and application.
(B) *Tritium removal from co-deposited layers*: • Develop methods for tritium removal from plasma-facing surfaces with minimum impact on machine availability. • Laboratory experiments are necessary to quantify and understand underlying mechanisms; tests in tokamaks are also essential to answer the remaining outstanding questions concerning the removal efficiency of the proposed techniques. • Continue studies of tokamak co-deposited films including analysis of film stoichiometry, microstructure and impurity concentrations. • Conduct tokamak studies of recovery of plasma performance subsequent to cleaning, and assess the collateral damage, which may result from oxygen exposure. • Investigate the effects resulting from mixing of materials.
(C) *Wall diagnostics*: • Develop and implement with dedicated machine time in situ time-dependent diagnostics necessary for understanding PMIs (by installing, for example, film thickness diagnostics, and a means of measuring erosion/deposition and retention under different operational conditions – start-up/shutdown, disruptions, attached/detached divertor, high-power operations, etc.). The complex and varied discharge history in tokamaks often makes archaeological studies of limited utility to test the models. • Instrumentation and access similar to that available for the DiMES probe in DIII-D are very useful to provide measurements at locations where erosion/deposition occurs, and where tritium and dust/debris are expected to accumulate.
(D) '*Composite*' *wall materials experiments*: • Conduct tests in tokamaks with impurities and wall materials to provide a realistic test-bed which would closely mirror whichever situation is proposed for the next-step device (e.g., the beryllium first-wall and carbon and tungsten divertor proposed for ITER). Such experiments would help answer questions including the magnitudes of erosion and tritium co-deposition, dust formation in the vessel, the ease of tritium removal from mixed-materials, as well as operational aspects (e.g., of using beryllium on the first-wall).
(E) *Experiments in bench-top facilities*: • Experiments under well-controlled and diagnosed conditions simulating the tokamak plasma edge, supported by modelling, are needed to investigate: (1) chemical erosion of C-based materials in plasma simulator and tokamak experiments to determine erosion yields and their dependence on plasma parameters for ranges of conditions (i.e., high-fluxes, low-temperature, high-fluences) where data are missing or scattered; (2) stability and enhanced erosion of promptly redeposited materials; (3) mixed-materials effects; (4) T retained in *n*-irradiated Be; (5) T retention and permeation engineering tests of PFC duplex structures.
(F) *Development and optimisation of critical first-wall components*: • Develop and optimise critical components, based on information provided by future dedicated operation experience in existing tokamaks (see text). • Explore new design solutions without using carbon. The primary alternative material for high-heat flux regions is tungsten. However, for this, efforts to reduce transients and mitigate disruptions in existing tokamaks need to continue with high priority.
(G) *Disruption mitigation and avoidance*: • Develop low-Z mitigation techniques (i.e., massive gas-puff, liquid jet injector, etc.). • Integrate detection and mitigation systems into the control system of an existing tokamak to test and demonstrate reliability.

tritium retention will limit the safe operation of D–T-fuelled tokamaks.

The wealth of data from existing tokamaks and simulators has facilitated great strides in our understanding of the processes involved in plasma–wall interactions. However, there are many unresolved issues relating to a next-step machine. Progress in understanding PMIs in the complex environment of a tokamak has been handicapped by the difficulties of real-time diagnostics. The R&D program needs to address the physics of the erosion mechanisms and the transportation and re-deposition of eroded material. More wall diagnostics with tokamak operation time dedicated to wall experiments with modelling of the results are essential to enable progress on wall physics. Specific PMI instrumentation and diagnostics have received relatively little attention to date. The complex and varied discharge history makes it important to go beyond 'archaeological' studies and test the models with time-resolved data from controlled experiments. Instrumentation and careful time-resolved measurements will be needed in a next-step device to quantify the effects and to better understand the underlying causes. The increased use of instrumentation will allow the database generation to proceed and should provide the component designers with important data.

Confrontation with some of the issues of PMIs may be postponed (or not be so evident) if the plasma duration is of order of tens of seconds – a few particle confinement times. However, only a self-consistent solution to issues of plasma confinement, stability, power and particle exhaust, wall integrity and tritium retention in a long-pulse machine will enable fusion power to become a practical reality.

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