Introduction and overview

Chapter 1 of the special issue: on the path to tokamak burning plasma operation

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Abstract

The International Tokamak Physics Activity (ITPA) has developed a comprehensive overview of the results of its coordinated R&D activities in fusion physics and diagnostics

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implemented during the past two decades with the aim of developing an improved physics basis for the operation of tokamak burning plasma experiments. Here, an introduction is presented to key concepts in the physics of burning plasmas in tokamaks, together with an overview of the scope of the principal areas of physics R&D pursued by the ITPA Topical Physics Groups in which the critical issues, major areas of progress and most significant recent results are highlighted. This paper introduces the following collection of articles, which encompass detailed presentations of the progress achieved by the Topical Groups in preparing the physics basis for the operation of tokamak burning plasma experiments expected to come into operation in the 2030s.

Keywords: Fusion energy, magnetic confinement, burning plasma, tokamak, ITPA

Glossary of Acronyms

Term	Definition
Facilities	
C-Mod	Alcator C-Mod, MIT, Cambridge, USA
ASDEX	Axially Symmetric Divertor Experiment, IPP, Garching, Germany
AUG	Axially Symmetric Divertor Experiment-Upgrade, IPP, Garching, Germany
COMPASS	COMPact ASSembly, IPP-CR, Prague, Czech Republic
DIII-D	DIII-D, GA, San Diego, USA
EAST	Experimental Advanced Superconducting Tokamak, ASIPP, Hefei, China
JET	Joint European Torus, EUROfusion, Culham, UK
JET-ILW	JET with ITER-like wall (Be/W)
JT-60U	Japan Torus-60 Upgrade, QST, Naka, Japan
JT-60SA	Japan Torus-60 Super Advanced, QST, Naka, Japan
KSTAR	Korea Superconducting Tokamak Advanced Research, KFE, Daejeon, South Korea
LHD	Large Helical Device, NIFS, Toki, Japan
MAST/ MAST-U	MegAmpere Spherical Tokamak/-Upgrade, UKAEA, Culham, UK
NSTX	National Spherical Tokamak eXperiment, PPPL, Princeton, USA
TCV	Tokamak à Configuration Variable at EPFL-SPC, Lausanne, Switzerland
TFTR	Tokamak Fusion Test Reactor, PPPL, Princeton, USA
WEST	Tungsten(W) Environment in Steady-state Tokamak, CEA-IRFM, Cadarache, France
Scientific	
AE	Alfvén eigenmode
BAAE	beta-induced Alfvén-acoustic eigenmode
BAE	beta-induced AE
CAE	compressional AE
EAE	ellipticity-induced AE
EGAM	EP-induced geodesic acoustic mode
GAE	global AE
LFAM	low frequency Alfvén mode
NAE	non-circular triangularity-induced AE
RSAE	reversed shear AE
TAE	toroidicity-induced AE
AI	artificial intelligence
BES	beam emission spectroscopy
CQ	(disruption) current quench
CTS	collective Thomson scattering
CXRS	charge exchange recombination spectroscopy
(D)EFC	(dynamic) error field correction
DMS	disruption mitigation system
dpa	displacements per atom (under irradiation)
DT	deuterium-tritium (fuel, reaction)
ECCD	electron cyclotron current drive
ECE	electron cyclotron emission
ECRH	electron cyclotron resonance heating
EDA	enhanced-D_α (H-mode)
EF	error field
EHO	edge harmonic oscillation
ELM	edge localized mode
em	electromagnetic
EP	energetic particle
EPM	energetic particle mode
ETB	edge transport barrier
ETG	electron temperature gradient (turbulence)
FIDA	fast-ion D_α spectroscopy
FILD	fast-ion loss detector
FLR	finite Larmor radius
FPP	fusion pilot plant
GAM	geodesic acoustic mode
GRS	gamma-ray spectroscopy
H&CD	heating and current drive
HFS	high field side (relative to plasma centre)

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HPC	high-performance computing
ICCD	ion cyclotron current drive
ICE	ion cyclotron emission
ICRF	ion cyclotron radiofrequency (heating)
(I)NPA	(imaging) neutral particle analyzer
IPB	ITER Physics Basis
IRP	ITER Research Plan
ITB	internal transport barrier
ITG	ion temperature gradient (turbulence)
ITPA	International Tokamak Physics Activity
KBM	kinetic ballooning mode
LIBS	laser-induced breakdown spectroscopy
LIDS	laser-induced desorption spectroscopy
LIF	laser-induced fluorescence
MGI	massive gas injection
MHD	magnetohydrodynamic
ML	machine learning
MRE	modified Rutherford equation
MSE	motional Stark effect
NBCD	neutral beam current drive
NBI	neutral beam injection
NES	neutron emission spectroscopy
(N)TM	(neoclassical) tearing mode
NTV	neoclassical toroidal viscosity
PBM	peeling ballooning mode
PFC	plasma-facing component
PFM	plasma-facing material
PIPB	Progress in the ITER Physics Basis
PWI	plasma-wall interactions
QCE	quasi-continuous exhaust (regime)
QCM	quasi-coherent mode
QH	quiescent H(-mode)
RE	runaway electron
RMP	resonant magnetic perturbation
RWM	(ideal plasma) resistive wall mode
RWTM	resistive wall tearing mode
SoC	system-on-chip
SOL	scrape-off layer
SPI	shattered pellet injection
TBM	test blanket module (for tritium breeding)
TEM	trapped electron mode (turbulence)
TIP	toroidal interferometer and polarimeter
TQ	(disruption) thermal quench
TG	(ITPA) Topical (Physics) Group
VDE	vertical displacement event
VNS	volumetric neutron source
VS	vertical stabilization
XPR	X-point radiator (plasma regime)

1. Introduction

In recent decades the central motivation for research on tokamak plasmas has been to develop the necessary understanding of magnetically confined plasma behaviour to facilitate the transition to routine plasma operation with DT fuel at the earliest possible date. This transition will enable the demonstration and exploration of burning plasmas with significant fusion power production and fusion power gain while addressing the challenges of handling the intense heat, particle and neutron fluxes associated with burning plasma operation and supporting the development of essential capabilities for maintaining long-pulse operation at reactor-scale performance. Impetus for this focus in the magnetic fusion R&D program is provided by the increasing emphasis on the strategic orientation of fusion energy as a major energy resource contributing to a 'net zero' energy mix in the second half of the 21st century: the substantial progress on ITER construction (e.g., [1]), R&D and construction activities for further burning plasma devices (e.g., [2]), the development of conceptual designs for DEMO-class devices (e.g., [3, 4, 5, 6, 7]) and proposals for FPPs (e.g., [8, 9, 10, 11, 12 13]) are indicative of the commitment of the international fusion research community to this overall goal.

The ITPA, which operates under the auspices of the ITER Agreement, but is directly sponsored by the fusion programs of the ITER Members, has, since its establishment in 2001, been centrally involved in this research. The ITPA TGs have worked closely with the experimental facilities and fusion theory and simulation activities across the international fusion community to address the key issues in fusion plasma physics which need to be resolved to support the design of burning plasma devices and the planning for their operation.

Following on from the ITER Physics Basis published by the ITER Physics Expert Groups [14], which informed the design of the ITER device [15], the ITPA issued an initial report on its progress in this research in 2007 [16], focused primarily on strengthening the physics design basis for ITER. This report, published under the title 'Progress in the ITER Physics Basis', provided an essential resource for the evaluation of the physics basis for the updated ITER design which emerged from the ITER Design Review in 2008 [17].

Since publication of the PIPB, the ITPA has continued to address key challenges in preparing for routine operation of burning plasmas, implementing wide-ranging collaborations to deepen the physics understanding of magnetically confined fusion plasmas. The scope of the R&D collaborations has expanded to encompass the establishment of the physics basis for the application of emerging concepts in areas such as operational scenarios, advanced feedback control techniques, and diagnostic instrumentation and components. An important feature of the ITPA's research activities during this period has been the implementation of a program of joint international experiments, developed in collaboration with the teams responsible for the operation of the magnetic fusion facilities, which has allowed common experimental studies on specific topics to be undertaken on multiple devices. This has proved

to be a particularly valuable instrument for studies of fusion plasma physics, yielding greater insight into the commonalities and differences of plasma scenario characteristics and physics processes among many of the tokamaks within the international fusion program.

The progress on R&D described in this special issue includes topics of relevance to ITER, as well as to tokamak burning plasmas more generally. In many cases, the underlying physics R&D activities were motivated by specific features of the ITER 2016 Baseline developed around the staged approach (see [18] and [19] for details), but the scope of the implemented R&D studies has wider implications. The new ITER 2024 Baseline currently under elaboration proposes modifications to several of these features (e.g., first wall material, H&CD mix, etc.), as introduced in [1]. These proposals are, to a significant extent, based on ITPA R&D studies discussed in [20], but they will, nevertheless, give rise to additional issues requiring further R&D beyond that described in subsequent chapters.

During the period covered by the reviews within this special issue, there have been major developments in tokamak facilities and their ancillary systems which have substantially enhanced the experimental capabilities available for the study of fusion-relevant plasmas. Such developments include:

- The commissioning of several fully superconducting devices, allowing plasma pulse lengths beyond several hundred seconds and supporting quasi-stationary plasma operation;
- An expanded range of devices with all-metal walls, including the JET-ILW (e.g., [21]) developed specifically to address physics and operational issues associated with the beryllium first wall/tungsten divertor target configuration foreseen within the ITER 2016 Baseline;
- Major expansion in the scope and sophistication of advanced real-time feedback control capabilities for numerous aspects of scenario control; entirely new elements include the use of RMPs for the suppression of ELMs; in the long-term program to detect, avoid and mitigate disruptions, the increasing exploitation of ML techniques for the integration and analysis of diagnostic measurements has shown considerable promise for the application to burning plasmas;
- Impressive progress in the continuing development of H&CD, fuelling and diagnostic systems, which have enhanced capabilities for plasma scenario control and allowed more detailed studies of physics processes at higher time and space resolution.

These experimental developments have been complemented by impressive progress in fusion plasma theory and in simulation capabilities, building on the increasing computational power provided by HPC systems. Combining the improvements in experimental techniques, particularly the availability of parameter measurements across the plasma and into the SOL with high time and space resolution, with a range of first-principles based simulation and analysis capabilities has allowed more detailed analysis of physics processes in all regions of the plasma and has yielded both improved understanding and an enhanced predictive capability towards the burning plasma regime.

The collection of reviews encompassed by this special issue has been prepared by the ITPA TGs to survey the progress made in the understanding of tokamak plasma physics in the fusion-relevant regime since publication of the PIPB. The reviews build heavily on the collaborative studies implemented by each TG, but also incorporate the results of the wider R&D program in experimental, theoretical and simulation investigations of tokamak fusion plasmas undertaken by the international fusion research community. The papers report not only on the progress made towards the establishment of the physics basis for the operation of burning plasma devices and the achievement of significant fusion gain, but also identify the outstanding challenges to be addressed to support the preparations for operation of such devices and reduce risk in their experimental programs. The scope of the progress reported in the following papers is indeed impressive, including:

- Significant advances in theoretical understanding and numerical simulation capabilities across areas such as plasma transport, MHD stability, edge and divertor physics, and EP physics;
- Major emphasis on integrated modelling of burning plasma scenarios, supporting both physics understanding and experimental operation;
- Development and exploration of a range of plasma scenarios with high confinement, robust MHD stability, acceptable control of power and particle exhaust and able to sustain quasi-stationary operation, including in the presence of all-metal PFCs;
- Advances in plasma control methodology allowing an integrated approach to the active management of plasma equilibria, MHD instabilities, power and particle transport, etc., in long-pulse plasma operation.

The achievement of high fusion power in DT operation in JET under quasi-stationary conditions [22] was a highlight of recent magnetic fusion research, confirming the progress achieved in integrating numerous elements of the advances made in the optimization of tokamak plasma operation during the preceding decades.

Following a brief discussion of the key physics challenges to be addressed in making the transition from the current generation of tokamak devices to a burning plasma facility (section 2), subsequent sections of this paper focus on providing a concise summary of the key results obtained within each of the ITPA TG's areas of research during the past ~20 years, with cross-references to the relevant sections of the

corresponding TG paper in the special issue. Outstanding questions which must be resolved to strengthen the physics basis for burning plasma operation are also highlighted. Since each of the special issue papers provides extensive references to the original research around which the TG reviews have been developed, citations to this research are not, for the most part, included in these sections. The sequence of research areas summarized in the topical sections is:

3. Transport and confinement physics [23]
4. Pedestal and edge physics [24]
5. MHD, disruptions and control physics [25]
6. Divertor and scrape-off layer physics [26]
7. Integrated operation scenarios [27]
8. Energetic particle physics [28]
9. Diagnostics [29]

Section 10 then provides a brief overview of the implications of the progress (and outstanding challenges) reported in the topical papers for the future development of fusion energy.

It should also be highlighted that the papers making up this special issue include an overview of each topical area, developed following a pedagogical approach. The aim is to provide a self-consistent character to the document and to contribute to knowledge management in the field of magnetic fusion physics, encompassing applications beyond ITER and, in some respects, beyond tokamaks. A key goal in the preparation of the special issue, therefore, has been the provision of a comprehensive document of value to all stakeholders in the wider fusion community, including those involved in private initiatives, and, in particular, to the emerging cadres of scientists and engineers that will carry the trans-generational fusion challenge forward.

2. Burning plasmas

Although there is no formal definition of the 'burning plasma' regime, a widely accepted working definition encompasses plasmas which are predominantly self-heated by the α-particles produced by DT reactions. Since the α-particle emerges with 20% of the DT reaction energy, the threshold for access to the regime corresponds to a value for the fusion gain, Q, the ratio of the DT fusion power to the external heating power to the plasma, of at least 5. While the initial generation of burning plasma devices might operate around this value (e.g., ITER targets $Q \geq 10$ in its baseline inductive scenario and $Q \geq 5$ in fully non-inductive operation), economically relevant electrical output from fusion power plants will likely require the sustainment of higher values of fusion gain, perhaps $Q \sim 30$, a region of operation often referred to as 'controlled ignition' – the earlier goal of achieving 'ignition', with $Q \sim \infty$, is no longer considered relevant or necessary for a tokamak fusion power plant, either because fully non-inductive operation is favoured or because external heating is required for plasma scenario control.

A key aspect of burning plasmas, therefore, is the presence of a significant population of high energy α-particles, with a birth energy of 3.5 MeV, which not only heats the plasma through collisions with the thermal electrons and ions, but which can have significant interactions with phenomena at a range of scales, from the microscopic scales typical of turbulent fluctuations to the macroscopic scales typical of global MHD instabilities. These interactions can modify both the thermal plasma behaviour (transport processes, MHD stability, etc.) and that of the α-particle population (e.g., by radial redistribution, which changes the α-heating profile). Moreover, the potential for nonlinear behaviour in the plasma due to, e.g., the dependence of the α-heating profile on the temperature and density profiles of the thermal plasma, introduces a significant novel aspect which distinguishes the burning plasma regime from the plasmas routinely studied in current tokamak experiments (note: DT fusion power, $P_{fus} \propto n_i^2 T_i^2$, in a first approximation, where n_i and T_i are the ion density and temperature respectively). This relationship might also generate a need for active 'burn control' to ensure that the fusion power output is stabilized at the desired value. The resultant interactions between the α-particles and these various aspects of plasma behaviour are often considered to be the central issue in burning plasma physics and, indeed, they are likely to introduce numerous novel physics processes which must be understood, and possibly controlled, to optimize fusion performance in the burning plasma regime.

Nevertheless, there is an additional range of physics issues implicit in the production and sustainment of this regime, and these must also be successfully addressed to strengthen confidence that the physics basis developed from current experiments, complemented by theory and simulation activities, guarantees robust and reliable operation of burning plasmas with high fusion gain. These include:

- The development of the required plasma scenarios which allow long-pulse or steady-state operation of burning plasmas, and the reliable implementation of the more complex plasma control schemes to sustain these scenarios.
- The scaling of plasma transport processes (and hence global confinement) to the required parameter range to support the targeted fusion gain – a key consideration derived from the dominant self-heating via α-particles is that external influences on plasma profiles, in particular of temperature and rotation, exerted in present experiments via auxiliary heating systems, is reduced or eliminated, and there is a degree of 'self-organization' of plasma profiles. Additional influences on plasma transport in the burning plasma regime, including access to improved (relative to L-mode) confinement regimes, derive from the fuel isotope mixture and from the predominantly electron heating processes which produce plasmas with $T_e > T_i$, in contrast to much of the current

experimental database, largely assembled from plasmas with $T_i \geq T_e$.

- Developing the necessary methodology for the control of power and particle exhaust (including the exhaust of thermalized α-particles, i.e., helium 'ash') in both stationary operation and during plasma transients (in particular, ELMs). This is likely to require active control of divertor plasma 'detachment' [30] by exploiting edge and divertor radiation enhanced by a low concentration of 'seeded' impurities. Associated issues which must also be addressed include efficient fuelling of DT plasmas, limiting erosion of PFCs, and minimizing both dust production and in-vessel tritium retention.
- Ensuring MHD stability close to operational limits (to achieve cost-effective fusion performance), in particular, to limit the impact of disruptions at high technical parameters (magnetic field, plasma current) and high fusion power. Sources of nuclear radiation, such as radioactive tritium and in-vessel structures activated by neutrons, which can 'seed' RE growth following disruptions, present additional challenges in the burning plasma environment. As in other areas, self-heating of the plasma complicates control of MHD stability, since a significant 'bootstrap' (i.e., pressure-driven) current fraction [31] determined by the (at least partially) 'self-organized' pressure profile will contribute to the current profile and current profile gradient, and hence safety factor (q) profile, influencing local MHD stability at low-order rational q-surfaces.

Associated with these challenges is the need to provide a comprehensive measurement capability to provide adequate monitoring, control and understanding of the burning plasma, implying the development of 'hardened' diagnostic systems which can survive in (quasi-) continuous operation in a harsh environment involving high heat, particle and neutron fluxes. Equally, developing robust control schemes that are compatible with the restricted diagnostic access and limited actuator flexibility of burning plasma devices will be essential.

Providing an improved physics (and, in the case of diagnostics and PFCs, technical) basis for developing responses to the challenges summarized in the preceding paragraphs, thereby ensuring a viable physics methodology for the demonstration of reliable burning plasma operation, has been the focus of the research programs implemented by the ITPA TGs since their inception. The progress made is the subject of the subsequent papers in this special issue.

The expected behaviour in a burning plasma depends, of course, on the specific operational regime chosen as the core of the burning plasma scenario. In recent decades tokamak experiments have revealed numerous plasma regimes, distinguished by a variety of access conditions and/or physics characteristics. Several of these regimes are predicted to exhibit superior fusion performance to the standard 'L-mode', or low confinement, regime which is universally observed in tokamaks, at least at low values of auxiliary heating power. The first regime discovered with significantly improved energy confinement relative to that of L-mode was the (ELMing) H-mode [32], which has been extensively studied since its discovery in the original ASDEX tokamak. H-mode remains a 'reference' improved confinement regime providing the baseline scenario around which many burning plasma concepts have been developed, in particular providing the design basis for ITER. As outlined in section 3 below, the strong dependence of the DT fusion reactivity on the confinement improvement factor (or 'enhancement'), H, allows either a significant increase in Q at fixed device scale, or a reduction in size (and hence cost) at a specified value of Q. This motivated a major effort in the IPB [33] to determine a quantitative characterization of the access conditions for the H-mode (at that time in terms of a 'power threshold', P_{th}) and for the scaling of the energy confinement time, τ_E, in terms of global plasma parameters (in the absence of a reliable first-principles based predictive capability). Details of these scalings, their subsequent refinement and a summary of the progress towards a first-principles based understanding of plasma transport processes is presented in section 3.

Due to the H-mode's 'reference' status, much of the physics research pursued by the ITPA has been focused on developing a more detailed understanding of key physics processes in ELMing H-mode plasmas and establishing an improved predictive capability for the design and operation of burning plasma experiments. Nevertheless, several limitations in the extrapolability of the standard ELMing H-mode plasmas to burning plasmas have motivated the development and study by the magnetic fusion community of alternative plasma regimes with the potential to overcome these perceived limitations. For example, the 'hybrid' or 'improved H-mode' regime exhibits a higher confinement enhancement (at the same plasma current) in existing experiments, which could allow operation at lower plasma current than in H-mode (see, e.g., [34]), providing greater margin against the most serious consequences of major disruptions. Regimes such as the QH-mode [35] and I-mode [36] offer the potential for achieving H-mode quality confinement, but without ELM perturbations at the plasma edge, obviating the need to mitigate or suppress such instabilities in burning plasma experiments, where they can generate intolerable levels of PFC erosion and plasma impurity contamination. More recently, similar considerations have stimulated more detailed studies of scenarios exploiting plasma equilibria with negative triangularity (see, e.g., [37]).

To achieve the ultimate goal of fully non-inductive steady-state operation of fusion reactors, widely considered essential for the economic exploitation of fusion energy, so-called 'advanced' plasma scenarios have been developed (see, e.g., [38, 39]). In some cases, such regimes exploit an 'internal

transport barrier' (a region of reduced plasma transport), located in the plasma core, in addition to the H-mode's 'edge transport barrier'. Advanced plasma scenarios do not share a single set of characteristics, e.g., there are variations in plasma current profile and in the strength and localization of an ITB, but typically they exhibit a better confinement enhancement factor than H-mode (when referenced against the standard H-mode confinement scaling) and can achieve fully non-inductive plasma operation by a combination of bootstrap current and auxiliary current drive.

There is a range of macroscopic behaviour across these various regimes: some share certain characteristics of the H-mode, e.g., ELMs, but differ significantly in other characteristics, e.g., current profile. While all exhibit some characteristic that may be advantageous at the burning plasma scale, e.g., enhanced confinement relative to H-mode, there are also alternative challenges, e.g., maintaining enhanced confinement and global MHD stability may be more challenging in advanced scenarios relying on subtleties of the current profile shape. Equally, certain challenges associated with the burning plasma regime are essentially the same across all regimes, e.g., achieving satisfactory power and particle exhaust in quasi-stationary conditions to ensure, *inter alia*, structural integrity of PFCs, acceptable PFC erosion rates and tolerable levels of plasma impurity contamination (including helium). Overall, these alternative regimes do not currently have such an extensive physics basis for extrapolation to the burning plasma regime as the ELMing H-mode, but ITPA research has gradually expanded to encompass studies of the physics processes determining key aspects of plasma behaviour observed in alternative regimes, and the TG papers within this special issue discuss progress made to date.

A key consideration, which assumes increasing significance as the fusion program approaches operation of burning plasma devices, is that the numerous elements of burning plasma physics which are (to a large extent) being studied in focussed R&D activities will need to be integrated into a comprehensive solution supporting (quasi-) continuous operation at high fusion power and gain: optimized operation of a burning plasma will require integration of an optimized core solution (with respect to confinement, MHD stability, etc.) with an optimized plasma pedestal (including suppressed ELMs or naturally ELM-free) and an optimized SOL and divertor solution satisfying all of the requirements related to power and particle exhaust. Development of such an integrated, and fully self-consistent, scenario will, of course, be a key role of burning plasma experiments, but the preparatory R&D underway will need to test, and confirm, the compatibility of solutions to specific challenges outlined above with this overarching goal. Since there will likely be interactions, or trade-offs, among the various solutions developed, such as core performance versus divertor power handling, it is critical that present experiments explore and optimize these trade-offs in relevant physics regimes to retire, or at least reduce, risks for future devices.

3. Transport and confinement physics

Transport and confinement in a burning plasma are important for two main reasons. First, fusion power production, which is a major programmatic goal of burning plasmas, is very sensitive to the confinement time. This can be illustrated by considering the impact of varying the *H*-factor in the IPB98(y,2) confinement time scaling for tokamaks [40]:

$$\tau_{\mathrm{E,th}}^{IPB09(y,2)}(s) = 0.0562\, I_{\mathrm{p}}^{0.93}\, B_{\mathrm{t}}^{0.15}\, \bar{n}_{\mathrm{e}}^{0.41} \times P^{-0.69}\, R^{1.97}\, M^{0.19}\, \kappa_{\mathrm{a}}^{0.78}\, \varepsilon^{0.58} \qquad (1)$$

where I_p is the plasma current in MA, B_t the toroidal field in T, $\overline{n}_e$ the line averaged electron density in units of 10^{19} m^{-3}, P the loss power from the plasma in MW, R the plasma major radius in m, M the average isotope mass, κ_a the plasma elongation at the plasma separatrix (i.e., at the last closed magnetic flux surface, with a the plasma minor radius) and ε the inverse plasma aspect ratio ($\varepsilon = a/R$). Based on Eq. 1, $H_{98y2} = \tau_{E,th}/\tau_{\mathrm{E,th}}^{IPB98(y,2)}$ is defined as the 'confinement enhancement factor', with $\tau_{E,th}$ the thermal energy confinement time measured in a specific experiment or obtained from a scaling derived for an alternative plasma regime. By definition, $H_{98y2} = 1$ for standard ELMing H-mode plasmas, while empirically $H_{98y2} \sim 0.5$ for L-mode plasmas.

For plasma with ion temperatures, T_i, of 10-20 keV, the plasma reactivity scales approximately as T_i^2. In regimes of limited α-heating, the fusion power scales approximately as H^2, since the fusion power scales as the square of the stored energy. In regimes approaching ignition, where the role of the external heating power is minimal, the fusion power scales as $H^{5.3}$. Thus, a 10% decrease in the H-factor can result in an approximately 40% decrease in the fusion power. Especially for FPPs or DEMO devices, this has significant implications for the achievement of high Q, the ratio of fusion power production to auxiliary heating power, and for reaching steady-state operation.

Second, understanding turbulent transport has been a major scientific topic for predicting transport and confinement of plasmas considering the complex interplay between heat, particle and momentum transport and that different instabilities have multiple scale lengths, which can interact. Since the previous report in the PIPB [41], major progress has occurred due to improved diagnostics, simulation tools, and plasma control techniques. The increased understanding can guide the development and discovery of new operational regimes with improved confinement. A more detailed presentation of the issues and results discussed in the following can be found in [23], which provides detailed references to the original research.

3.1 Phenomenological approach

Historically, the predicted performance of ITER discharges relied heavily on scaling laws derived from databases composed of input from multiple devices. Since then, the databases have been extended to new operating regimes and with higher quality data. Thus, it is appropriate to review the changes in the requirements for transition from the L-mode to H-mode, and confinement scaling laws.

3.1.1 Power threshold scaling for enhanced confinement regimes. As discussed in the IPB [33], the achievement of higher confinement than L-mode is required to enhance the fusion power production in ITER and the achievement of a burning plasma. Thus, the requirements for achieving an H-mode, which is the baseline scenario for ITER, are important. In burning plasma devices, prior to the onset of α-heating, the auxiliary power determines if a transition to H-mode will occur and at what density. The present ITPA scaling for the L-H power threshold is:

$$P_{\mathrm{th}}\,(MW) = 0.0488\, B_{\mathrm{t}}^{0.803}\, \bar{n}_{\mathrm{e},20}^{0.717} S^{0.941} \qquad (2)$$

where $\overline{n}_{e,20}$ is the line averaged electron density in units of 10^{20} m^{-3} and S is the plasma surface area in m^2.

The power threshold scaling in Eq. (2) is based on deuterium discharges in which the power threshold is increasing with density. Below a minimum density, $n_{th;min}$, the power threshold increases with decreasing density. C-Mod and JET, operating with its ITER-like wall, individually identified a positive, nearly linear, scaling of $n_{th;min}$ with toroidal field. A tendency of $n_{th;min}$ to increase with plasma current has been observed on JET-ILW and AUG. More recently the variation of the minimum L-H power threshold with density on AUG has been attributed to a critical power flow through the ion channel and an empirical scaling has been proposed. Several other explanations have also been proposed to explain the transition from the low density to high density branches.

There is substantial scatter in the power threshold data, which reflects, in part, that other variables are important in determining the threshold power for the transition to H-mode. Only a few examples are given here, but a more extensive discussion is given in [23] section 7. Experiments in AUG, JET, and C-Mod with all-metal walls and divertors have shown a modest decrease in the power threshold (~20% at high density) compared with experiments with carbon walls. Experiments also indicate that divertor geometry can affect the power threshold. Experiments on JET, MAST and TCV have shown that the power threshold decreases if the X-point is closer to the vacuum vessel wall. Another example is that in the double null configuration the power threshold is reduced. This has motivated further ITPA experiments that reveal a complex dependence on plasma shaping and divertor geometry. In addition, the application of RMPs to suppress ELMs has also been observed to increase the power threshold requirements.

Historically, the increased power threshold in hydrogen discharges compared with deuterium suggested that the power threshold scales as the reciprocal of the atomic mass, M. More recent experiments with different heating techniques, as well as the study of helium discharges, has concluded that the $1/M$ scaling is a rule of thumb with considerable variation. Since burning plasma facilities will begin operation with a phase with low fusion power production, this may be an operational consideration.

While the focus of research has been on the transition to H-mode, the development of the I-mode, as discussed in [23] section 6, has been studied, including the transition from L-mode to I-mode and from I-mode to H-mode.

3.1.2 Energy confinement time scaling in H-mode. The combination of a large increase in data over a wider parameter regime, also from new facilities, and coupled with improved diagnostics and analysis techniques motivated revisiting the energy confinement time scaling reported in the IPB. The results of this re-assessment are reviewed in [23] section 8.

For consistency with the predictions by IPB98(y,2), the following plasma parameters were used for confinement prediction in the baseline inductive $Q = 10$ scenario in ITER: $I_p = 15$ MA, $B_t = 5.3$ T, $\overline{n}_e = 10.3\times10^{19}$ m^{-3}, $R_{geo} = 6.2$ m, $\delta = 0.48$, $\kappa_a = 1.7$, $\varepsilon = 0.32$ and $M_{eff} = 2.5$. The power through the last closed flux surface was 87 MW based on 40 MW auxiliary heating power and 400 MW fusion power and, by considering an estimate for the power radiated from the core, the energy confinement time was predicted to be 3.62 s. The new database considers various fits to a subset of the database as well as different statistical analyses. The scaling regression that is viewed as most relevant to ITER is given by:

$$\begin{aligned}\tau_{\mathrm{E,th}} = {} & (0.067\,^{+0.059}_{-0.032})\, I_{\mathrm{p}}^{1.29\pm0.16}\, B_{\mathrm{t}}^{-0.13\pm0.17}\, \bar{n}_{\mathrm{e}}^{0.147\pm0.097} \\ & \times P_{\mathrm{l,th}}^{-0.644\pm0.061}\, R_{\mathrm{geo}}^{1.19\pm0.27} \\ & \times (1+\delta)^{0.56\pm0.36}\, \kappa_{\mathrm{a}}^{0.67\pm0.63} M_{eff}^{0.30\pm0.16} \end{aligned} \qquad (3)$$

where $P_{l,th}$ is the thermal loss power from the plasma in MW, R_{geo} the geometric centre of the plasma in m, δ the plasma triangularity at the plasma separatrix and M_{eff} the average isotope mass.

Except for the inclusion of triangularity, this has the same engineering variables as used in the IPB98(y,2) scaling shown in (1). In the new analysis, the dependence on ε is found to be almost zero and is therefore no longer included in the scaling. This results in an energy confinement time of 2.9±0.46 s using the weighted least squares fit. In this estimate the same power through the last closed flux surface is assumed; however, that corresponds to increased auxiliary heating power since the

α-heating power is reduced due to the reduced confinement time.

While most of the exponents in the more recent regression analysis are similar to those in IPB98(y,2), one significant difference is in the value of exponent for tokamak size, R_{geo}, which decreased from 1.97 to 1.194. This is an area still under investigation by the ITPA group.

This discussion has focused on scaling using engineering variables. The ITPA group has also performed scaling studies using physics variables and observed a Bohm-like dependence. Nevertheless, the authors note that the uncertainty in the exponents is large.

While scaling analysis has some well-known limitations, this has been a valuable update to the previous results and identified important new lines of inquiry, including the role of rotation, data from hybrid discharges, as well as the impact of wall materials and coatings. In particular, the database includes results from devices with a variety of wall conditions and application of wall coatings. This more recent database suggests that identifying regimes of operation with further enhancements in confinement compared with these projections may be necessary to meet the ITER objectives.

3.2 Understanding confinement through modelling collisions and turbulent processes

While the use of scaling relationships has been useful in scoping studies, there are some fundamental limitations regarding the extrapolation outside of the regime in which the fit is performed and issues associated with the neglect of other engineering parameters such as the divertor shape, wall material and wall conditioning. This has motivated the development of physics-based models. There has been significant progress in theoretical simulations of turbulent processes and also in the inclusion of collisions. The gyrokinetic framework has been the focus of simulations though there has also been progress in the gyrofluid nonlinear framework, hybrid, gyrofluid-gyrokinetic schemes, and projecting velocity space onto a Laguerre-Hermite basis. The nonlinear gyrokinetic simulations have evolved significantly since the publication of the PIPB. This is due, in part, to new algorithms and vastly increased HPC facilities that have enabled studies that were not previously possible. Furthermore, new physics has been incorporated into simulations, including electron dynamics, collisions, higher-β simulations, multi-scale which includes both ion and electron scale lengths, and flux-tube, as well as global, simulations. Here $\beta = 2\mu_0 \langle p \rangle / B_t^2$, with $\langle p \rangle$ the volume averaged plasma pressure, is the ratio of plasma pressure to magnetic field pressure.

The nonlinear models have been compared with turbulence measurements and turbulent particle, momentum, and heat fluxes, validating the applicability of the models and identifying regimes where further work is required. Some open questions remain, such as the electron temperature fluctuation level and the study of a wider range of parameters closer to that of a burning plasma. While the progress has been impressive, additional physics has been identified that needs to be incorporated into these simulations, especially in simulating 3D magnetic fields and the interaction of EPs with turbulence. This will be discussed further in the context of the different transport channels.

The need to simulate the time dependence of the plasma profile in near real-time motivates the development of reduced models. The reduced models are widely used for analysis, predictions and time dependent simulations. The most widely used approach to reduced-order tokamak turbulence models invokes the 'quasilinear' approximation, which requires the solution of the linear instability spectrum. Nonlinear physics is captured by analytic saturation rules. A large effort has been made to compare full gyrokinetic calculations with the nonlinear fit used in the quasilinear models, as discussed in [23] section 9. These saturation rules are a crucial part of the model. Comparisons with experimental data, which identify shortcomings, motivate the inclusion of additional physics in the nonlinear models or extension of the range explored. Further nonlinear runs are then performed to update the saturation rules. The nonlinear simulations described here rely on the δf model with flux tube simulations to reduce the computational requirements to develop the saturation rules. The local treatment should be applicable to the core of the plasma but may be marginal in the critical edge and pedestal region.

Recent experimental results for various transport channels will be discussed in the context of turbulent and collisional transport. While in the discussion below different transport channels are discussed separately, they are often strongly coupled due to off-diagonal terms and, of course, the underlying physics.

3.2.1 Particle transport. Particle transport has a direct impact on fusion power production since the fusion power density is proportional to the product of the deuterium and tritium ion density. Thus, peaked density profiles are beneficial. Moreover, the bootstrap current, which is important in tokamaks operating in steady-state, is driven in part by the density gradient. More generally, impurity transport can have an impact on the power balance and depends on particle density gradients. As discussed in [23] section 2, particle transport has been studied in the context of diffusion and particle pinch. Both ion and electron particle transport coefficients were obtained with improved diagnostics and perturbation techniques. Analytical models and gyrokinetic simulations have both identified particle transport mechanisms in the presence of the ITG mode and the TEM.

Fast isotope mixing and fast impurity transport in ITG regimes is expected based on analytical calculations and

nonlinear gyrokinetic simulations, where it was demonstrated that the resonance condition leads to ion particle transport coefficients substantially larger than electron particle transport coefficients. Significantly, the existence of large ion particle transport coefficients implies that ion density profiles are uncorrelated to the corresponding ion source. The opposite situation is encountered in the TEM regime, where it is found that the ion particle turbulent coefficients are smaller than the corresponding electron coefficients.

An anomalous particle pinch had been measured in several tokamaks at low collisionality, an observation which is understood in the framework of the ITG/TEM transport theory. Multi-variable regression analysis using a combined density peaking scaling database, which included data from JET, AUG, C-Mod and JT-60U, identified the effective collisionality, ν_{eff}, as the most important parameter statistically to characterize density profile peaking. The correlation between ν_{eff} and n_e/n_G, with n_G the Greenwald density, could be broken by exploiting C-Mod data, a result which strengthened the case for ν_{eff} as the controlling parameter. A comparison of the data to nonlinear gyrokinetic simulations showed good agreement on the density peaking. Predictions for ITER and DEMO indicate significant density peaking, which was not considered in earlier ITER predictions and can improve performance.

Recent experimental work has focused on isotope effects. Local measurements of the particle transport coefficients, comparing, for example, deuterium and tritium, have also highlighted that the isotope scaling of particle transport can differ from that for heat transport. In the heat transport channel, a strong favourable scaling for tritium is seen compared with that for deuterium, but the particle transport channel exhibits no, or a very small, isotope scaling. Experimentally, particle transport in the core does not seem to depend on isotope in L-mode and H-mode plasmas, consistent with GENE simulations. However, quasilinear turbulence calculations using TGLF-SAT2 indicate that the agreement with experiments in tritium in the region $0.5 < \rho < 0.8$ is not as good as that in deuterium L-mode plasmas.

3.2.2 Impurity transport. A predictive understanding of impurity transport is needed to calculate the fuel dilution in a burning plasma, for example, by helium resulting from α-ash, and the power radiated, for example, by high charge states in the core and lower charge states in the edge and divertor region. Both turbulent and neoclassical transport can be important, but their impact can be different for different charge states and mass, as discussed more extensively in [23] section 3.

Early helium experiments on TFTR and DIII-D showed that helium transport was faster than predicted by neoclassical theory, with the helium particle diffusivity being of the same order as the ion heat diffusivity. In conventional aspect ratio tokamaks, turbulent impurity transport dominates over neoclassical transport over a broad range in minor radius. In contrast, in spherical tokamak discharges, which are characterized by strong rotation, so that ITG turbulence is stabilized, and in which TEM is unstable only outside mid-radius, neoclassical transport can play a dominant role in this extended radial region for light impurities. JT-60U discharges also observed helium and carbon transport faster than neoclassical, even in the presence of ITBs when the ion heat diffusivity was consistent with neoclassical predictions. In contrast both JT-60U and JET, as well as other devices, observed the accumulation of impurities with higher charge, broadly consistent with neoclassical theory.

Significant theoretical progress has been made in understanding impurity transport due to turbulence. The transition from TEM- to ion ITG-driven turbulence results in the ratio of the turbulent impurity diffusion coefficient to the turbulent heat conductivities having a strong non-monotonic dependence on the electron to ion heat flux ratio. The turbulent impurity transport relative to the heat conductivity is largest when the ion and electron heat fluxes are comparable. Furthermore, off-diagonal components also exhibit the property that their direction can change sign depending on the type of turbulence. These trends are supported by gyrokinetic simulations.

Databases of intrinsic C and B density profiles were assembled from JET C-Wall and AUG H-mode plasmas and analyzed to identify the principal regions of parameter space where large disagreements between theory and C and B experimental profiles occurred. The disagreement in JET is found to be particularly large at high collisionality, while the disagreement in AUG is found to be particularly large at high values of the ion temperature logarithmic gradient and rotation gradients, which correlate with NBI heating powers. Comparison of data from the two devices showed that AUG cases with higher ion temperature and rotation gradients were close to the JET parameter space investigated and confirmed the overprediction of impurity density peaking found.

In C-Mod separate diffusive and convective transport coefficients of Ca have been experimentally inferred, yielding good agreement with both quasilinear gyrofluid and nonlinear gyrokinetic simulations. Good agreement has been found generally, in particular, for the diffusion coefficient of all examined cases and for conditions with inward impurity convection. Nevertheless, in an ICRF-heated EDA H-mode plasma, clear experimental evidence of outward impurity convection and a hollow Ca density profile is obtained, a result which cannot be reproduced by theoretical predictions.

Experimental results on JET exhibit a significant variation in impurity density peaking: the helium profile is as peaked as the electron density profile, while profiles of C and N are flat to slightly hollow around mid-radius. Theoretical results are characterized by a more limited variation of the density

peaking. In addition, JET turbulent transport models were not able to reproduce the observed hollowness of the intrinsic C density profiles. Recent research has identified additional mechanisms which can contribution to the production of hollow impurity density profiles in connection with the role of energetic beam ions, the polarization drift and the impact of an externally applied torque.

Neoclassical transport often produces the dominant convection of heavy highly charged impurities. This is due, in part, to the neoclassical convection term scaling as the square of the ion charge, while the turbulent transport convection is constant with charge at high charge. In addition, the poloidal inhomogeneity of the impurity density due to plasma rotation strongly increases neoclassical transport. In many experiments at high collisionality, with highly charged impurities which typically are in the Pfirsch-Schlüter regime, processes driven by toroidal rotation can favour the central accumulation of impurities: the rotation strongly increases the neoclassical transport and can also reduce the relative strength of the temperature screening. Recent theoretical predictions and experimental results at JET-ILW have shown that when the impurity collisionality is sufficiently low, reaching conditions in which the impact of the plasma toroidal rotation increases the relative strength of the temperature screening, more favourable conditions for avoiding impurity accumulation occur.

Impurity accumulation unfavourably impacts present tokamak plasmas in the most common high collisionality regime of heavy impurities. Under such conditions, its mitigation critically relies on modification of the plasma profiles, to reduce or reverse inward convection, typically by reducing the ion density gradient and by increasing the ion temperature gradient, and/or via the relative increase of turbulent transport by central wave heating, and/or by reducing plasma rotation and/or the poloidal asymmetry. These elements provide a clear rationale for the interest in understanding and predicting the magnitude of the turbulent diffusion, even in regimes which typically are dominated by neoclassical transport.

3.2.3 Ion and electron thermal transport. Though ion and electron thermal transport were the focus of experimental and theoretical investigation in the IPB [33], as discussed in [23] section 4, new experimental results have challenged the models and resulted in the inclusion of new effects and new understanding. These new results have provided very constraining experimental evidence for theory validation, in particular, for the striking observation that in some regimes, ion profile stiffness (which quantifies the rate of reaction of the temperature gradient to a variation in the heat flux) decreases with increasing power. Understanding ion and electron profile stiffness is important due to the relationship with the pedestal temperature. In regimes, where the stiffness is large, the core temperature is very dependent on pedestal temperatures.

During the past ~10 years, intense collaborative activities have focused on two specific topics: the effects of em stabilization and fast ions on ion heat transport and the role of electron-scale turbulence and multi-scale interactions on electron heat transport. Gyrokinetic theory has been extended to include fully em nonlinear gyrokinetic equations for tokamak edge turbulence, improved collisional operators and $\boldsymbol{E}\times\boldsymbol{B}$ treatment, global simulations, flux-driven simulations within a full-f approach, interplay with neoclassical transport and with EPs, and multi-scale interactions at real mass ratios. The increase in available computer power has allowed an accompanying increase in the numerical resolution routinely used. Progress made in the development of theory-based approaches to the modelling of core thermal transport is reviewed in [23] section 9.

Several linear stabilization mechanisms for ITG turbulence are known. Main ion dilution by fast or impurity particles and Shafranov shift stabilization are well known basic electrostatic mechanisms. Recently, a resonant linear electrostatic mechanism has been identified involving wave-particle resonance between fast ions and the ITG mode when the fast ion magnetic drift frequency approaches the linear ITG frequency margins. Nonlinear em effects associated with coupling of the ITG to shear Alfvén modes lead to ITG stabilization with increasing β, even in the absence of fast particles, but the effect is significantly enhanced by the presence of fast particles, which increase the drive of the shear Alfvén modes. In gyrokinetic simulations with GENE, inclusion of both fast particles and nonlinear em effects was key to matching the JET experiment. The simulations predict that with increasing β there is an accompanying increase in the zonal coupling efficiency. The inclusion of fast ions and em effects can, in some experiments, have a very large impact. This is an area of continued investigation, including the effect on electron heat transport. Since the validity of the local limit is questionable when considering MeV-scale fast ions, highly demanding global flux-driven simulations may ultimately be necessary, both to validate current local gradient-driven simulations and to assess whether these fast-ion related ITG stabilization mechanisms remain and are relevant in burning plasmas.

Since the publication of the PIPB, the new data has driven updates to the quasilinear models, and particularly the saturation models. While the updated quasilinear models reproduce many aspects of experiments, they tend to fail when applied to advanced scenarios, significantly under-predicting the ion temperature and density peaking in the inner half-radius, where the missing nonlinear physics is important due to high-β and a broad region of low magnetic shear. This highlights the development of a theory-based em stabilization

model for quasilinear transport models as a high priority research topic for the near future.

While the core electron heat transport has been historically interpreted as mostly due to ITG or TEM micro-instabilities, comparison of experimental data with simulations indicated that ETG modes could also play a key role in some cases. GYRO simulations which include both ion- and electron-scale instabilities in a multi-scale nonlinear simulation have been performed. These simulations matched the ion and electron heat fluxes, as well as the electron profile stiffness, in C-Mod experiments. Multi-scale simulations have shown a very complex interplay between ITG, TEM and ETG instabilities, with the development of zonal flows. For example, GYRO simulations of the ITER baseline case showed that the ETG instability can even stabilize the ITG. This has motivated further experiments on AUG, JET and TCV and comparisons with quasilinear and nonlinear simulations. The picture that emerges is that ETGs could impact the electron heat flux when $T_e \sim T_i$ and R/L_{Te} $(= -(R/T_e)(\mathrm{d}T_e/\mathrm{d}r)$, the normalized logarithmic electron temperature gradient) is sufficiently high, though the role of impurities is still under investigation.

Because multi-scale simulations require very extensive computational resources, they are not routinely performed. This has motivated the development of heuristic models to determine when ETG instabilities are important. As discussed in [23], this is an active area of research and further comparisons between the quasilinear and nonlinear models are planned.

3.2.4 Rotation and momentum transport. Within the transport and confinement fusion community, historically, the heat and particle transport channels received significantly more attention than the momentum transport channel, as the plasma temperature and density enter directly into the produced fusion power. However, with the increased understanding of the transport in these channels, came increased awareness of their interconnectivity. Indeed, it is now well recognized that plasma rotation directly influences plasma turbulence, as well as confinement transitions and impurity transport. Moreover, the plasma rotation can play a critical role in the stabilization of deleterious MHD instabilities, for example, helping to prevent 'locked modes' and avoid the associated disruptions. The latter is of particular importance for larger devices, such as ITER, which operate at higher plasma current and magnetic fields than present day devices.

With this recognition came the associated drive to develop momentum transport theory to provide a predictive capability for the rotation profile in future devices, and to improve the accuracy of the predictions of the other transport channels. Progress made in the development of the relevant theory and in the detailed comparison with experimental observations is reviewed in [23] section 5. Early work on the subject first considered the turbulence momentum diffusivity, and concluded it was expected to be of the same order as the ion heat diffusivity, making the ratio of these two quantities, known as the Prandtl number (*Pr*), of order 1. This turned out to be a robust prediction, subsequently reproduced by increasingly sophisticated turbulence simulations. This prediction was also worrying with respect to future devices, as it would lead to very low rotation values.

However, comparison to early experimental measurements showed a seeming discrepancy, as early experimental values of *Pr* were robustly less than unity. This discrepancy was resolved with the realization that the momentum transport channel cannot be described purely by diffusive transport. As is the case with particle transport, there exists a non-negligible convective component, better known as the Coriolis momentum pinch. And, in addition, the momentum transport channel features a third transport component known as the residual stress, which corresponds to the sum of all terms not proportional to either the rotation or its gradient. These additional momentum fluxes have been robustly demonstrated in the experiment and shown to be non-negligible for the accurate determination of the rotation profile, a result which is also expected to be true for future devices. In addition, the inward direction of the momentum convection in combination with the strong intrinsic rotation observed at the plasma edge in present-day tokamaks provides two mechanisms that can counteract the diffusive transport, resulting in increased rotation levels and potentially peaked rotation in future devices.

The last fifteen years have seen strong development in momentum transport theory, which now features a robust physical understanding of the origins of the observed momentum convection and has also identified the key players in residual stress generation. Modelling capabilities have also advanced significantly, with full-*f* simulations now elucidating the intrinsic rotation generation mechanisms at the plasma edge, and with quasilinear gyrokinetic models now able to provide reasonable predictions for the momentum diffusion and convection in the plasma core. Early attempts at experimental validation of the predicted diffusion and convection typically did not achieve quantitative agreement. However, recent work has shown that this is likely to be a result of the neglect of residual stress and the time dependence of the transport coefficients in those early experimental analyses. When these elements are self-consistently included, quite good agreement is achieved for the momentum diffusion and convection. Local, quasilinear models, however, do not contain the physics needed to calculate the residual stress. Rather, global nonlinear simulations are required for this purpose and have, in fact, succeeded in reproducing experimentally measured intrinsic rotation gradients measured in the core of tokamak plasmas.

The largest unknown with respect to momentum transport modelling of H-mode plasmas is the rotation at the pedestal

top, which, in present-day integrated modelling, is typically taken from the experiment, as no pedestal momentum transport model exists. In fact, pedestal transport models with respect to the heat and particle channels are also missing. In the case of momentum transport, measurements from multiple devices have demonstrated the presence of a strong, co-current intrinsic rotation in the edge pedestal region. The origins of this rotation, however, are not clear, as multiple mechanisms are present in this region and able to contribute. Multiple inter-device studies have been performed to estimate this quantity and have resulted in a wide range of predictions for ITER, spanning an order of magnitude. How this intrinsic rotation will scale to future devices, therefore, is highly uncertain and remains one of the primary challenges facing the momentum transport community.

3.2.5 Effect of 3D fields on transport. Non-axisymmetric 3D magnetic field configurations have been extensively studied in stellarators and reversed field pinches. There is a complex interplay between the plasma's response to the perturbation and its impact on transport processes in both the core and the edge region. For tokamaks, research on 3D fields has been motivated, in part, by the application of RMPs to suppress ELMs. Many of the theoretical concepts originally developed for other magnetic field configurations have therefore been extended to tokamaks. The discussion below will highlight the role of 3D fields on turbulence and neoclassical transport in tokamaks, a subject which is reviewed in considerable detail in [23] section 6.

The physical picture of how 3D fields influence plasma transport has evolved from an initial simplified approach involving a stochastic boundary, where electron heat and particle diffusivity are enhanced by additional radial terms, to a rather more complex picture involving an interplay between the penetration of the external fields, governed by the plasma parameters, and their effects on both neoclassical and turbulent transport. Experimental results showed, moreover, that RMPs induce NTV torque and damping in the plasma core or change the transport regime from ITG to TEM at the plasma boundary.

The application of RMPs to generate ELM suppression, or mitigation, is often associated with a decrease in the pedestal plasma parameters, particularly noticeable in the plasma density (the so-called 'pump-out'). As noted above, conditions in the edge region can have significant impact on the core temperature profiles, as well as on the toroidal velocity. In addition, they can have an impact on the L- to H-mode transition. Because the application of 3D fields is a very complex phenomenon, this has motivated extensive experiments and theoretical modelling.

Neoclassical transport due to 3D fields can affect the edge conditions and can contribute to pump-out. This behaviour is now understood to be triggered by the same physics that produces the toroidal NTV torque. The enhanced particle transport (pump-out) often observed in the presence of RMPs, alters the pedestal pressure gradient and is one of the ingredients that pushes the plasma into a stable peeling-ballooning regime. A strengthening of the pump-out is observed at low collisionality and near the offset of the plasma rotation, regimes which are relevant for ITER. While NEO and M3D-C[1] have reproduced the pump-out in ELM-suppressed DIII-D discharges using RMPs, the modelled pump-out does not always match the experimental values.

While NTV often damps rotation, it can also increase the rotation magnitude in low-torque regimes. This has been observed on DIII-D using non-resonant $n = 3$ perturbations to observe an acceleration in the counter-rotation direction, resulting in improved confinement in QH-discharges. Toroidal field ripple-induced NTV torque is predicted to be smaller than the NBI and intrinsic torque drives responsible for rotation in ITER. The toroidal field ripple due to TBMs may damp rotation, however, though it may be possible to correct that effect by applying $n = 1$ fields.

Within the ideal MHD description of plasmas, RMPs induce shielding currents on rational surfaces to preserve the magnetic topology. Such shielding currents produce a braking torque, referred to as an electromagnetic torque. The steady-state rotation frequency is then determined by the balance between the viscous and em braking torques. This has also been studied in the context of locked mode formation. More direct experimental evidence of the interplay between magnetic perturbations, plasma rotation, and high-fidelity modelling finds support through extended MHD computations, which have successfully reproduced the parametric space of ELM suppression via RMPs in the DIII-D and KSTAR tokamaks.

Tokamak experiments have indicated that the power threshold for the L-H transition increases with increasing resonant magnetic field strength derived from plasma response models. The impact on the power threshold is reduced, however, in cases where the applied field is mostly non-resonant. Generally, any weakening of the $\boldsymbol{E}\times\boldsymbol{B}$ shear at the plasma edge results in an increased power threshold for the L-H transition. Measurements in both DIII-D and AUG revealed a flattening of the edge $\boldsymbol{E}\times\boldsymbol{B}$ rotation, with a reversal occurring at high enough magnetic field strength. The reversal of the $\boldsymbol{E}\times\boldsymbol{B}$ shear in AUG was linked to the increased power threshold, while an increase in power threshold was already observed in DIII-D prior to the change in sign in $\boldsymbol{E}\times\boldsymbol{B}$ shear.

Effects associated with the application of 3D fields entail a synergistic approach, as, e.g., changes in the neoclassical NTV torque induced by external fields can cause changes in the radial electric field which then directly affects turbulent transport vortices via the $\boldsymbol{E}\times\boldsymbol{B}$ shear. The plasma boundary is also significantly affected by external perturbations, as there is both experimental and theoretical evidence for a heterogeneous structure of the magnetic field lines

intersecting the PFCs. Though the complex interplay between transport and macroscopic effects is not fully understood, enormous progress has been made.

4. Pedestal and edge physics

The understanding of physics processes in the plasma pedestal has made huge progress since publication of the PIPB, which reviewed the emerging physics of the H-mode edge pedestal within a chapter on confinement and transport [41]. Here, progress towards the physics of the pedestal and ELM behaviour is documented in a dedicated publication [24].

The H-mode edge pedestal sets the boundary condition for small-scale, transport-generating instabilities in the plasmas core, largely determining the core profiles and fusion performance in cases with 'stiff' core transport, where heat diffusivity increases strongly with heat flux. It also interacts with the SOL and therefore influences conditions for power exhaust and impurity control. Through these connections the edge pedestal defines many aspects of the operation space of a tokamak reactor. The optimal conditions for the core (high temperature) and boundary (high density, low temperature) are often in opposition and thus achieving both through the edge pedestal interface is a key challenge for burning plasmas. Last but not least, the H-mode pedestal governs edge stability, particularly the occurrence of ELMs, a self-healing plasma edge instability that leads to intermittent bursts of particle and energy loss. In a fusion reactor these transient events can severely limit the lifetime of, or even the over-all integrity of, the first wall material.

Progress in the extensive worldwide research activities on pedestal and edge physics is presented in [24]. Major advances in predicting the radial profiles of pedestal quantities (density, temperature, pressure, etc.), primarily in the type-I ELM regime, are discussed in [24] section 2, dealing with pedestal structure and stability. The following section characterizes the transient behaviour of the ELMs in this regime. Since the publication of the PIPB it has become clearer that the heat loads from unmitigated type-I ELMs are not tolerable in a burning plasma device or fusion reactor (see [26]). Therefore, subsequent sections cover research aimed at active suppression or mitigation of ELMs, as is planned, e.g., on ITER, and on regimes which are naturally free of large ELMs. There has been a strong increase in the number of candidate regimes for addressing the transient heat loads issue, and in the understanding of the physics mechanisms controlling such regimes.

4.1 Pedestal structure and stability

The understanding of, and ability to predict, the height of the H-mode pedestal has greatly advanced, with efforts focusing primarily on the type-I ELM regime. While MHD limits of the gradient were broadly understood at the time of the PIPB publication, there was large uncertainty in the scaling of pedestal width. Experimental measurements of widths in different parameters were compared with potentially relevant physical scales, such as ion gyroradius and neutral penetration length, with widely varying results, including some that were quite unfavourable. This was noted as a major outstanding issue in the PIPB [41]: the resultant large range in pedestal height was a dominant source of uncertainty in model predictions of burning plasma fusion performance.

Through a combination of advances in high resolution pedestal diagnostics and modelling, the pedestal height and width are now understood to be strongly coupled. Many models predict PBM stability limits and, in most cases, these both agree with experiments and reproduce trends with plasma shaping in the type-I ELM regime on multiple tokamaks. Peeling modes tend to dominate at lower density/collisionality with high bootstrap current and ballooning modes dominating at higher density. Transport physics in the pedestal also plays a role in setting the pedestal width and thus, for pedestal gradients limited by pedestal stability physics, also the pedestal height. In the EPED model (see [24] section 2.3.5) the pedestal gradients are postulated to be limited by KBMs, giving a pedestal width proportional to $\beta_p^{1/2}$, where $\beta_p = 2\mu_0 \langle p \rangle / \langle B_p(a) \rangle^2$, is the volume averaged plasma pressure normalized to the poloidal field averaged around the plasma separatrix, $\langle B_p(a) \rangle$. EPED model predictions have been applied to tokamaks with a wide range of parameters, in most cases giving good agreement with scalings of the pedestal width, and thereby predicting pedestal height to within about 20% (see [24] section 2.3.5). In C-Mod this was extended to pedestal pressures approaching those of ITER (see [24] section 2.9). Together these results from ITPA R&D studies have greatly increased confidence in predictions of this key boundary condition.

The role and prediction of pedestal density vs pressure profiles is an active area of research. While the EPED model, for ease of predicting future devices, uses an assumed input density profile, it has been found both experimentally and in extended models that the relative shift between the density and temperature profiles, linked to separatrix density, can impact pedestal pressure height. Fuelling, impurity seeding and wall material all influence the pedestal profiles; in particular, experiments in both AUG and JET showed significant changes when transitioning from C to W PFCs (see [24] section 2.4). Increased pedestal physics understanding and predictive capability, ideally including coupling to SOL and divertor boundary models, will be critical for extrapolating pedestal performance predictions to metal-walled burning plasma devices such as ITER, FPPs, DEMOs and tokamak fusion power reactors.

Pedestal turbulence and transport between ELMs is another active area where great progress has been made but more research is still needed. Several tokamaks have documented changing fluctuations during the ELM cycle, linked to

different rates of rise in different parameters (n_e, T_e, T_i) between ELMs. Turbulence models are being extended into the pedestal region for comparison, although the overlapping and variable scale lengths of gradients, fluctuations and gyroradii make this challenging. If transport from other modes exceeds the KBM limit, this can lead to lower pedestal gradients and, somewhat counter-intuitively, potentially wider pedestals and higher pedestal pressure. Improved understanding and predictive capability of pedestal transport and its effect may be particularly important in extending models to regimes with either active mitigation of ELMs or intrinsically ELM-free regimes. This may also explain some cases, notably on JET, where ELMing discharges are found not to lie on the expected peeling-ballooning boundary (see [24] section 2.4).

4.2 ELM characterization and modelling

ELMs have been observed and studied since the discovery of the H-mode (see [24] section 3.1). Early attempts to find a purely empirical scaling of ELM losses have shown limited success. More recently, modelling capabilities have increased significantly, so that after publication of the PIPB the focus has turned to detailed, time-dependent MHD simulations to understand the rich phenomenology of ELMs. It has now become possible to simulate the full ELM cycle (pedestal evolution before an ELM, triggering of an ELM, ELM crash and recovery) with the aim of reproducing the experimentally observed features: pedestal profile evolution, divertor heat load footprint, magnetic mode spectra (see [24] section 3.4). It remains as a goal, not yet fully reached, to develop a predictive capability for ELM losses and divertor heat load in future reactor plasmas.

4.3 ELM control with actuators (3D fields, pellet injection, plasma position oscillations)

Predictions of heat losses due to ELMs in future large tokamak plasmas are still somewhat uncertain but, in all likelihood, ELMs of large amplitude cannot be tolerated in a burning plasma device. Therefore, much emphasis has been placed on researching techniques to either mitigate ELM losses or to avoid ELMs altogether. They can be grouped into methods that rely on actuators that affect the ELM instability directly, and plasma parameter regimes with enhanced small-scale fluctuations that induce enhanced radial transport in the edge gradient region and thereby act to keep the edge pedestal below ELM stability limits.

4.3.1 ELM control with 3D fields. After the surprising finding in DIII-D that small non-axisymmetric perturbations to the magnetic field of a tokamak H-mode plasma can completely suppress ELMs (See [24] section 4.1.1.1), extensive research has been devoted to reproducing and exploring the applicable parameter range for this technique and to understanding the underlying physics mechanisms. Since publication of the PIPB, all conventional aspect ratio tokamaks equipped with in-vessel saddle coils for magnetic perturbations (AUG, DIII-D, EAST, KSTAR) have reproduced and studied the phenomenon (see [24] section 4.1.1.2). In contrast, in tight aspect ratio devices (MAST, MAST-U, NSTX) and in JET (with only ex-vessel EFC coils), a reduction of ELM losses has been achieved but not full suppression (see [24] section 4.1.1.2). Unfortunately, purely empirical extrapolation of the applicability range to larger plasmas is hampered by the fact that suitable sets of in-vessel saddle coils are available only in the above-mentioned tokamaks of about the same size and not, for example, in JET. Therefore, large emphasis has been placed on better understanding of the physics of ELM suppression with the goal of developing models with predictive capabilities for burning plasmas. In support of this research a rich experimental database has been obtained, with identification of several critical parameters for access to ELM suppression: plasma rotation, ion species, pedestal collisionality/density, heating scheme, plasma shaping, and edge safety factor (see [24] section 4.1.3). Full ELM suppression has so far been achieved only in plasmas up to a critical pedestal plasma density (see [24] section 4.1.3.2). There is ambiguity in present day experimental data as to whether this is an actual limitation of the plasma density or a related quantity such as pedestal collisionality. Current experiments on RMP ELM suppression can match the ITER pedestal collisionality, but not simultaneously the ITER pedestal density and cannot, therefore, predict access on a purely empirical basis.

A clear experimental result is the importance of the perturbation field being resonant with the background field in the pedestal region, i.e. a resistive (tearing) plasma response as opposed to non-resonant, pure kink-like perturbation (see [24] section 4.1.2.6). The situation is somewhat complicated by plasma rotation-induced helical currents at resonant rational surfaces which effectively suppress (shield) the RMP, unless the driving electron fluid plasma rotation vanishes, or kinetic effects produce the resistive response. Experimental results on AUG and DIII-D suggest that during ELM suppression these conditions are met at resonant surfaces near the top of the edge pedestal (see [24] section 4.1.1.1). Observations of the HFS plasma response (DIII-D) and direct imaging of the magnetic flux perturbation (AUG) confirm the existence of a magnetic island near the pedestal top (see [24] section 4.1.2.4).

The externally applied magnetic perturbation at magnetic surfaces in the steep gradient region is usually well shielded, so the resonant response at the pedestal top is produced by poloidal mode coupling with stable ideal kink-modes due to poloidal plasma shaping. In experiments, the strongest effect on ELMs and the lowest threshold for ELM suppression is found if the external perturbation vacuum field is aligned with

these kink-modes, which are amplified by the edge pressure gradient and edge bootstrap current (see [24] section 4.2.1.4). In experiments in DIII-D, MAST and AUG this appears as a small but significant deviation of the optimum perturbation from the resonant field pitch (see [24] section 4.1.2.6). The physics of the plasma response to the external magnetic field is studied in detail with a variety of resistive MHD and kinetic models: MARS-F, JOREK, BOUT++, NEO-2/KiLCA, M3D-C[1] (see [24] section 4.1.2).

A picture of ELM suppression is now emerging in which additional radial transport in the gradient region and/or near a rational surface reduces both the pressure gradient and the radial extent of the gradient region, with the effect to stabilize the edge pedestal against ELMs (see [24] section 4.1.2.4). Enhanced radial transport of particles (the pump-out effect noted previously) is indeed observed. The precise nature of the transport enhancement is still under investigation: static magnetic islands induced by the RMP, neoclassical gyro-centre drifts and turbulent transport can all contribute, and their relative importance is yet unknown.

A potential issue for burning plasma devices with applied 3D RMP fields for ELM control may be a performance penalty imposed due to the modification of the H-mode edge pedestal by the pump-out effect. Also, the H-mode power threshold increases if the 3D field perturbation is already applied during the H-mode transition, in time to suppress the first large ELM. Finally, there is usually significant torque from the applied fields to reduce the plasma rotation, which can affect plasma scenarios that rely on the intrinsic torque of a heated fusion plasma or moderate momentum input from auxiliary heating systems. These potentially detrimental effects are studied intensively in current experiments with the aim to understand and mitigate their impact on fusion performance (see [24] sections 4.1.1.2, 4.1.2.8, 4.1.2.9, 4.1.6).

4.3.2 ELM pacing by pellets and plasma position oscillations. A completely different route to ELM mitigation is based on observations that intentionally triggered ELMs, at frequencies higher than the natural ELM frequency, often reduce energy loss per ELM and consequently reduce heat flux to the material wall. It is found that both injection of cryogenic pellets and fast plasma position excursions can trigger small ELMs. These induce additional pedestal transport that can be controlled such that spontaneous destabilization of large ELMs is avoided (see [24] sections 4.2 and 4.3). These techniques imply a degradation of the H-mode edge pedestal, and a main research goal is to minimize the performance reduction while keeping heat losses from triggered ELMs at a tolerable level.

Frozen deuterium pellets are typically injected with small pellet mass and small velocity to deposit their fuel in the shallow pedestal region. The local plasma density increase around the ablation region causes a temperature drop compared to the rest of the flux surface. This produces heat transport along the magnetic field towards the ablation zone, which is faster than the particles streaming away from it. As a result, the local plasma pressure increases and triggers an ELM. In MHD simulations (JOREK) this trigger mechanism is reproduced (see [24] section 4.2.2). The large local plasma density leads to reduced heat losses of triggered ELMs as compared to those produced by the spontaneous ELMs that would occur without pellet injection. Interestingly, results differ for tokamaks with C walls and metal walls, as exemplified by AUG and JET: in both devices, when using C PFCs, ELMs could be triggered immediately after a previous ELM, but could be triggered only after a certain 'dead time' with metal PFCs (and lower C impurity content) (see [24] section 4.2.1.2). This behaviour was attributed to differences between the inter-ELM recovery of the edge pedestal in cases with and without significant C impurity content; but definitive conclusions have not yet been reached.

Fast plasma position excursions produce a transient of the poloidal magnetic flux (and hence, loop voltage) which increases the edge current density so that the plasma edge becomes unstable against peeling modes, in consequence triggering an ELM. This method has been studied in various experiments, and with detailed MHD modelling: DINA/ MISHKA, JINTRAC/CREATE, JOREK/STARWALL (see [24] section 4.3.3). Since fast plasma position transients can exert significant forces on the tokamak coils and support structures, it is important to understand, and to be able to extrapolate, the minimum displacement required to trigger ELMs.

Note that future reactors are predicted to need a significant reduction in natural ELM size, by a factor of ~100, based on analysis of target material damage limits. This implies a need for a substantial rise in the ratio of triggered ELM frequencies/natural ELM frequency for either the pellet or plasma position excursion techniques, representing a considerable challenge for these techniques. This will be particularly the case if the ELM amplitude does not continue to decrease inversely with the normalized frequency increase at high frequencies and if large ELM triggering pellets are required at low collisionality.

4.4 Regimes without large ELMs

The study of regimes with high confinement but which are naturally free of large type-I ELMs has received significantly increased worldwide attention in recent years. This reflects the recognition that while active ELM mitigation is planned for ITER, there will be serious engineering challenges in implementing similar techniques on DEMO and future fusion reactors. As noted above, there is still uncertainty about whether ELM suppression, or sufficient mitigation, can be achieved for all plasma parameters of interest. At the time of the PIPB, there was brief discussion of 'small ELM' regimes

and of only two quiescent regimes: the EDA regime, then observed only on C-Mod, and the QH-mode regime, at the time observed on DIII-D, but only at low density and with counter-NBI (see [24] sections 5.4 and 5.3 respectively). The extended discussion devoted to ELM-free and small ELM regimes in this issue reflects both a significant expansion in the number of regimes observed experimentally and in the range of tokamaks actively studying them.

The I-mode regime differs from H-mode regimes in that it features development of an edge temperature pedestal but not a significant change in density profiles from L-mode; this is not just an H-mode with enhanced transport. The phenomenon was noticed transiently in early AUG and C-Mod discharges with ion-∇B drift away from the X-point, a condition for almost all I-modes (see [24] section 5.2), and studied for insight into transition physics. More recently, I-mode has been recognized as potentially attractive for burning plasmas due to lack of both ELMs and impurity accumulation. The stationary regime has been studied most extensively on C-Mod, where it was found that conditions for access and sustainment scale favourably with magnetic field: the L-I power threshold scales much more weakly with B_t than the H-mode threshold, leading to a wider I-mode power range at high B_t (see [24] section 5.2 and [23] section 7.4). AUG and EAST observe similar trends, also with density. I-mode has also been observed on DIII-D and KSTAR, together representing a wide variety of heating methods and wall materials, and indicating robustness. Detailed studies of changes in turbulence and transport at the L-I transition have been reported. In most cases there is a reduction in low-wavenumber fluctuations in some frequency ranges and a higher frequency 'weakly coherent mode' appears. Models for I-mode physics are emerging, but there is not yet consensus on the key mechanisms or the role of various fluctuations; this is an active topic of research. Another outstanding challenge, common to several quiescent regimes, is integration with a detached divertor solution. Work is in progress to develop multi-device scalings of I-mode access and energy confinement.

The 'standard' QH-mode was extended in the 2000's to AUG, JT-60U, JET and EAST (see [24] section 5.3.1); at that time each of these tokamaks had C PFCs. Experiments and analysis at DIII-D have greatly extended the range of conditions for the regime and improved its understanding. Pedestals are consistently found to lie close to the peeling branch of the peeling-ballooning boundary, with a coherent, low-n edge harmonic oscillation (EHO) apparently providing sufficient transport to prevent ELM crashes. Importantly, QH-mode has been achieved with both co- and counter-I_p NBI, showing that a critical magnitude of rotation shear is needed, independent of its sign. Access with net zero NBI torque, combined with NTV torque from external coils, has been demonstrated. Energy confinement is typically very high, well above the IPB98(y,2) scaling, and pedestal density has been increased to 60% of the density limit. A variant of the regime, the 'wide pedestal QH', mode has been found on DIII-D, in which the coherent EHO transitions to more broadband MHD (see [24] section 5.3.1). This regime, accessed at zero net NBI torque, features a broader, lower gradient pedestal and has achieved even higher pedestal pressures and confinement. Great progress has been made in modelling of both regimes; physics understanding of the underlying physics of QH-mode is now mature compared to other ELM-free regimes. A key issue, for both regimes, is that, due to the need for low collisionality and high SOL temperatures, impurity sources tend to be high, leading to high core plasma impurity levels, Z_{eff}. QH-mode has recently been accessed on metal-walled devices (AUG, EAST) but it has been more challenging to sustain robustly (see [24] section 5.3.1); this requires further research for burning plasmas. Similarly, in current devices, it is difficult to achieve high separatrix density and divertor detachment while maintaining low collisionality.

The EDA H-mode regime also features stationary H-mode pedestals, but at higher density and collisionality than QH-mode. The regime was the most common mode of operation on the C-Mod tokamak and extended to most of its operating parameters, up to $B_t \sim 8$ T. Record tokamak plasma pressures were achieved in this regime at $B_t = 5.7$ T (see [24] section 5.4.3). Key advantages of the regime are high particle transport, avoiding impurity accumulation, and ease of integration with seeding and detached divertor operation; reduction in heat flux to ITER requirements has been demonstrated while maintaining $H_{98y2} \geq 1$. EDA H-mode has been extended to AUG, where it is accessed most readily with ECH (see [24] section 5.4.1). In EDA the pedestals are away from the peeling-ballooning boundary, with additional transport apparently provided by a QCM. Highly detailed probe measurements of this mode were made on C-Mod. Most measurements localize it near the outer pedestal or separatrix, though there have been some discrepancies between devices and diagnostics (see [24] section 5.4.2). There is also not yet consensus among models on its physical mechanism. A key question for extrapolation of the regime to burning plasmas is the location for which high collisionality must be maintained. If, as AUG experiments suggest, it is the separatrix (see [24] section 5.4.5), then it may be possible to combine a high effective collisionality at the plasma separatrix, ν^*_{sep}, needed for divertor solutions, with the low-ν^* pedestal needed for core performance. These are hard to separate on smaller devices, and research on larger tokamaks is needed. Also of interest is the connection to the 'quasi-continuous exhaust' regime, discussed below, which appears closely related, since discharges often evolve from EDA to QCE at higher power and pressure.

In addition to the normally quiescent regimes above, several H-mode regimes exist which have periodic

instabilities which are much smaller in amplitude than type-I ELMs and thus produce smaller heat pulses. Type-II ELMs are facilitated by strong shaping, closeness to double null and higher density. In recent years this regime has received increased attention, particularly at AUG (see [24] section 5.5.1.4). Experiments comparing pellet to gas fuelling showed that it is the pedestal foot, not the pedestal top, which must have high collisionality, and analysis shows the foot region to be ideal ballooning unstable. Transport to the SOL occurs through a series of filaments, which broaden the SOL. This regime is highly promising for core-edge integration and has been renamed the QCE regime. It has also been observed on TCV (see [24] section 5.5.1.5). As for EDA H-mode, a key question for extrapolation of the QCE regime to burning plasmas will be the degree to which low collisionality at the pedestal top can be maintained with the required high collisionality at the pedestal foot. Analysis of experiments in the final JET campaign (in progress) will be important in this regard.

Grassy ELMs, in contrast to the EDA and QCE regimes, are accessed at low collisionality and relatively high q_{95}: >4 on JT-60U and >5.3 on DIII-D and EAST (see [24] sections 5.5.1.1, 5.5.1.2 and 5.5.1.3). Grassy ELMs have high frequency, typically 500-1000 Hz, and are observed at higher β_p. This ELM regime is thus well suited to steady-state advanced tokamak scenarios, as reported on multiple tokamaks.

JET, with the ITER-like wall, has reported a new small ELM regime observed at high current and power and with low gas fuelling. This was named 'Baseline Small ELMs' to highlight that key plasma parameters (q_{95}, β_N, H_{98y2}, ν^*) are typical of the ITER baseline scenario (see [24] section 5.5.2.6). Pedestal gradients are strong in temperature and relatively weak in density, and they are found to be peeling-ballooning stable. Further research is needed to understand the physics and extrapolability of this regime.

Each of the above regimes has been the subject of considerable modelling, and understanding is improving. For all small ELM regimes, and even for transients which sometimes occur in nominally ELM-free regimes, an important question is how the size of heat pulses will scale to burning plasmas and whether they will be tolerable. Prediction of the pedestal top conditions, which may differ from those in type-I ELMs, is an outstanding and important problem for both quiescent and small-ELM regimes.

Discharges with negative triangularity are also described in this section. These differ from other regimes in that there is no substantial ETB per se, and thus no ELMs. However global confinement is often increased close to H-mode levels, largely due to decreases in core transport. In some cases, a modest increase in edge temperature gradient, reminiscent of I-mode pedestals, is seen. This operating regime was first studied on TCV and relatively recently on DIII-D and AUG (see [24] section 5.6). It appears well suited for core-edge integration, and more research is needed and planned.

5. MHD, disruptions and control physics

Magnetohydrodynamic stability of the plasma is one of the critical requirements for the robust and reliable operation of a magnetic confinement fusion device, as unstable global MHD modes can cause a rapid deterioration in plasma confinement and terminate plasma operation. Abrupt plasma termination due to disruptions can lead to excessive thermal and em loads on the tokamak, causing damage to the first wall components, thereby reducing the operational lifetime of the device. Disruptions can also produce other secondary effects like the generation of high energy REs that can seriously damage PFCs. The study of MHD instabilities, gaining better understanding of disruption phenomena and devising means of avoiding, predicting, controlling and mitigating disruptions, has therefore been a major focus of magnetic fusion research over the past few decades. Significant progress has been achieved in this area since previous reviews published as part of the IPB [42] and PIPB [43] review papers. This is the product of better understanding of some of the primary MHD instabilities contributing directly or indirectly to disruptions, the availability of a more extensive experimental database, advances in modern statistical tools, including ML-based techniques to characterize and analyze disruption phenomena, and improvements in plasma control techniques. However, our understanding and ability to control/mitigate disruptions are still not at a stage that can be considered acceptable for the robust operation of burning plasma and reactor-grade devices and this remains an active area of R&D today. This section is based on the material in [25] (and references therein), which provides a comprehensive overview of the progress made in the last ~20 years and the outstanding physics issues and technological challenges that still need to be resolved.

5.1 Disruptions

As discussed in [25] section 3, a plasma disruption is characterized by two main phases: the thermal quench, during which most of the thermal energy of the plasma is offloaded onto the first wall, and the current quench, where the plasma current decays due to the high post-TQ resistivity of the plasma. During this phase there can also be an unstable VDE, which can bring the plasma into contact with the first wall. There can be serious deleterious consequences of disruptions in the form of substantial em and thermal/particle loads on major elements of the device.

5.1.1 Electromagnetic loads. Electromagnetic loads arise from induced currents in the surrounding structures due both to the current decay and to uncontrolled vertical motion of the plasma during a VDE, as well as from poloidal plasma 'halo' currents generated by a 'short-circuit' between the vertically

displaced plasma and first wall components (see [25] section 3.1.1). Electromagnetic loads have been experimentally assessed on several tokamaks including JET, AUG, COMPASS and EAST, and extensively modelled by various MHD codes such as M3D, M3D-C[1], NIMROD and JOREK. An important experimental finding has been the observation of lateral (sideways) forces of unprecedented magnitude on the JET vessel as a result of an asymmetrical disruption event, which is not generally observed in other devices. Since this can have serious repercussions for ITER, there has been a significant effort towards gaining a proper understanding of the underlying causes of an asymmetric disruption and of the associated sideways forces. A likely cause that has received much simulation/modelling attention, is the growth of a kink-like mode (an $m = n = 1$ mode) during the disruption evolution. Other factors that could lead to such forces are the existence of so-called Hiro currents and toroidally asymmetric currents in the plasma SOL. There is, as yet, no general consensus about the proper physical modelling or understanding of the exact dynamics of the asymmetric disruptions.

5.1.2 Heat and particle loads. During a TQ, the plasma thermal energy directly flows onto the PFCs by conduction and convection, which subsequently leads to an influx of impurities from the walls, leading to a radiative collapse of the plasma thermal energy. During a CQ, the plasma magnetic energy partly drives large eddy currents in the vacuum vessel and also gives rise to a toroidal electric field within the plasma which, in combination with kinetic processes, drives REs. The latter can then produce very localized and intense heat loads on the first wall. The total amount of deposited heat, the time scales over which this occurs and the material properties of the PFCs are key factors that influence the deleterious impact of disruptions, as detailed in [25] section 3.1.2. The role of the wall material was brought out clearly in JET experiments. With a C wall, the C impurities released by a TQ generated sufficient radiation to balance the ohmic power generated in the CQ phase thereby reducing the heat flux during the CQ phase. When JET shifted from C to a Be wall, the latter released lower-Z impurities which radiated much less and did not compensate the ohmic power, leading to hotter post-disruption plasmas and resulting in a much slower CQ. The subsequent VDE, initiated at high plasma current, generated significant heat fluxes to the PFCs, causing localized PFC melting. Later, when W PFCs were installed on the divertor targets, the radiation fraction increased from about 30% to 50% of the ohmic input power, with a corresponding much faster CQ. Extensive experimental studies on surface melting have been carried out in several tokamaks, including JET, WEST, EAST, T-10, AUG and DIII-D. The melting of the PFCs may itself trigger a disruption and lead to increased disruptivity in subsequent pulses, as observed in C-mod.

REs generated during a disruption are a further important factor that can lead to significant PFC damage due to their small impact area, fast time scale of deposition and deep penetration. Key physics aspects of RE generation and their interaction with the thermal plasma are outlined in section 8.1 (and discussed in detail in [28] section 12). The combined influence of em forces, thermal loads and high energy impacts from REs can generate substantial loads on the tokamak vessel and in-vessel components, with the potential for damage to components which would prevent reliable operation of the device. Hence considerable attention is being paid to devise means of avoiding and predicting the occurrence of disruptions and, in the event of an occurrence, implementing control and mitigation measures, as discussed in section 5.1.5.

5.1.3 Disruption avoidance. Disruption avoidance strategies depend strongly on the disruption path that defines the disruption trigger and its subsequent evolution of the plasma towards disruption (see [25] section 3.2). Appropriate sensors and actuators are employed to detect early disruption signatures and, where possible, suppress their subsequent development to prevent a disruption. The growth of low-(m, n) resistive MHD modes, such as (N)TMs and RWMs, are a common feature of many disruption scenarios. Their early detection and stabilization have therefore been the subject of many experimental and modelling studies over the past few decades. However, the prediction of (N)TMs remains elusive, while their onset β-threshold is observed to fall as reactor values of ρ^* (the ion Larmor radius normalized to the minor radius) are approached. Nevertheless, feedback control schemes for (N)TMs based on early detection of 'seed' islands and the use of ECCD to control their growth have been shown to be very effective on many tokamaks. The modified Rutherford equation has proven to be a useful physics model to describe the island evolution of (N)TMs and to provide a reliable basis for real-time control. A complementary 'pre-emption' scheme consists of anticipating the onset of an unstable (N)TM and taking pre-emptive measures, such as injecting ECCD at the most probable location of the mode trigger before the emergence of an island there, or even radially sweeping the deposition around the relevant mode rational surface. The efficacy of such methods, that normally require less power than would be needed to stabilize a growing mode, has been demonstrated on DIII-D, TCV, JT-60U etc., but needs further development. For avoidance of (N)TMs, one needs to understand and identify the underlying seeding mechanisms and suitably modify them (e.g., by altering the q-profile) to prevent the occurrence of the mode – a somewhat challenging task when extrapolated to reactor conditions.

In the case of RWM-induced disruptions, several active feedback systems – consisting of a combination of internal and external coils – have demonstrated effective control of RWMs in dedicated experiments in DIII-D, NSTX and other devices.

The basic strategy is to counteract the measured RWM mode perturbation (detected by magnetic sensors) by an applied field with a certain spectrum and phase. Avoidance of an RWM-induced disruption involves an optimization of the conditions that enhance the stabilizing mechanisms of an RWM, such as toroidal plasma rotation and drift-kinetic damping effects. Other disruption prevention schemes involve the early detection and control of vertical plasma position, since the failure of such control can lead to a VDE, or tracking events that can push the plasma beyond the known operational boundaries (such as the H-mode or L-mode density limits) and modifying plasma parameters to prevent such a trajectory. Prevention of impurity-induced disruptions, as demonstrated on JET and AUG, involves the early detection of temperature hollowing through bolometric measurements of core radiation from high-Z impurities and taking appropriate preventive measures, such as increasing the deuterium fuelling flux to reduce overall W influx and/or central heating using ECRH or ICRF.

5.1.4 Disruption prediction. Developing a reliable method for predicting disruptions to be able to take appropriate remedial actions, either to prevent their occurrence or to mitigate their deleterious consequences, has been a high priority research topic since the early 1990s. As detailed in [25] section 3.3, there are two major approaches that have been adopted: (i) a physics-based approach that relies on the timely identification of the various MHD instabilities and other physical triggers responsible for initiating a disruption; and (ii) a mathematical approach that relies on statistical tools to identify plasma states preceding a disruption from a database of past experimental disruption events. While great progress has been achieved in developing physics-based criteria for the onset of disruptions, such as the critical island amplitude, the distance of the $q = 2$ surface from the edge, dependence of time-to-disruption from plasma and tokamak properties, the RWM stability criterion etc., they are not yet adequate to provide a reliable disruption prediction model. Given the extremely complex nature of disruptions there has been an increasing focus on employing statistical methods to uncover correlations in experimental data that are difficult to identify from known physics-based criteria. The statistical approach has made great progress in recent years thanks to the rapid strides made in the development of ML tools and techniques based on AI. An integrated approach that combines the best of both these approaches offers much hope for the future development of a reliable disruption prediction model and marks the most significant progress made in this area over the past decade and a half.

5.1.5 Disruption mitigation and control. In parallel with the ongoing research on disruption prediction and avoidance, there has been a steady development in disruption mitigation research aimed at distributing the thermal and magnetic energy released during disruptions to protect devices against damage to the vacuum vessel or to in-vessel components (see [25] section 3.4). Following early recognition of the physical and technological limitations of MGI, attention has shifted to particle delivery via solid and shattered pellet injection (this last referred to as SPI) that can reduce the thermal load of a TQ event by replacing the conducted heat by uniform radiation. A key challenge for the technology R&D supporting the development of the pellet injection capability for burning plasma devices will be to achieve the parameters required for adequate pellet penetration and TQ mitigation, including necessary injection velocities.

The relative success of such a mitigation scheme is measured by two metrics: the radiation fraction and the radiation peaking. The objective is to maximize the radiation fraction towards future reactor requirements (e.g., >90% for ITER) while minimizing the peaking factor (through spatially distributed injection) to prevent localized heating and melting of PFCs. Such a goal is being pursued through many experimental studies on existing tokamaks as well as advanced simulation codes like KPRAD that tracks the full dynamics of SPI assimilation. The injected materials also help in mitigating the consequences of the CQ by helping to maintain an acceptable plasma current decay rate and CQ duration. From a range of modelling analysis, it now seems that when the CQ time is significantly less than the resistive wall time (τ_w) of the vessel, asymmetric forces should be significantly reduced and well within tokamak design limits. Thus, the CQ duration needs to lie inside an acceptable operational window, where the lower limit is defined by the maximum tolerable amplitude of eddy currents and em loads on the in-vessel components, while the upper limit arises from the minimization of halo currents and reduction of thermal loads on the PFCs, as well as of the (axisymmetric and sideways) vessel forces. For ITER operating at the full plasma current of 15 MA, this range is 50-150 ms.

As noted in section 5.1.2, following a TQ the resulting high toroidal electric field generally gives rise to REs, which can reach relativistic energies. Experimental and simulation studies aimed at mitigation measures to prevent, control or deconfine REs form an important component of disruption mitigation research. Among the measures being considered are injection of mixtures of deuterium with higher-Z impurities that can lead to avoidance of REs, possible enhancement in the transport of REs through their interaction with background unstable plasma waves, use of EFs to deconfine REs through use of passive asymmetric coils, active mitigation of REs using RMPs, etc. Since some traditional disruption mitigation mechanisms, while reducing heat and em loads, can generate background plasma conditions producing a stronger RE drive, the exploitation of applied or induced MHD to dissipate REs may be a promising option. Among the various collisional techniques, the SPI technique

has proved to be the most successful and technologically feasible. SPI systems are now being tested in several tokamaks in support of the design of the SPI-based ITER DMS.

5.1.6 Disruption modelling. Further support to the ITER DMS design is being provided by intense numerical modelling efforts to generate physical insights and quantitative predictions that can help in optimizing the design/operation of the system (see [25] section 3.5). A full 3D modelling of a disruption involving phenomena with multiple temporal and spatial scales is an extremely challenging task that is constrained by the limitations of present day nonlinear MHD codes. For example, to carry out a simulation of a tearing mode for a realistic Lundquist number (e.g., $S > 10^6$), representing large and hot plasmas such as JET and ITER, one needs an extremely fine spatial resolution to resolve the current layers present in MHD simulations. The state-of-the-art codes today can resolve plasmas with $S = 10^7$ in the centre, whereas ITER is projected to have central values of $S > 10^9$. Likewise, to provide a complete temporal evolution of a disruption or disruption mitigation process can require more than a hundred Alfvén times – a capability that present codes do not possess. Nevertheless, much progress has been achieved by codes such as JOREK, NIMROD, and M3D-C[1] that use high-order finite elements and implicit time stepping and whose capabilities are expected to rise as more powerful computing resources become available in the near future. Many challenges remain but there is optimism that, as the capabilities of the codes improve, it might be possible to undertake modelling of the disruptive limits in the state space of the basic discharge parameters using realistic transport parameters and sources.

5.2 MHD stability

Much of the progress made in characterizing disruptions and controlling/mitigating them, as described above, can be attributed to our improved understanding of major MHD instabilities that trigger and contribute to a disruption. A comprehensive review of the advances made in the experimental and theoretical/numerical studies carried out for the principal MHD instabilities that plague a tokamak plasma is presented in [25] section 2.

5.2.1 Sawtooth instability. The sawtooth instability (see [25] section 2.1) is the most important central core instability in tokamaks and one that has been extensively studied over the past several decades. It is therefore one of the best understood of the various MHD instabilities. Several theoretical models, starting with the earliest Kadomtsev reconnection-based model and its refinements, such as the Porcelli model, have successfully delineated the main features of the instability. These have been further augmented by numerical simulation studies that have provided important details regarding its nonlinear evolution, the time scales of the reconnection process, factors influencing the sawtooth period, their impact on impurity and fast particle redistribution, etc. There are still fundamental features of the instability that are not yet well understood, such as a definitive physical explanation of the sawtooth crash process, the triggering process of the instability and the evolution of the q-profile during and after the crash. These and other issues are still the subject of active theoretical model studies, numerical simulations and basic experimental investigations. The sawtooth instability *per se* is not considered to threaten plasma confinement and its presence has beneficial effects, such as the prevention of impurity accumulation in the core region. Nevertheless, it is important to control this instability, since its uncontrolled growth can lead to deleterious effects like seeding of an (N)TM that can set up a chain of events ending in a disruption. Fortunately, control of the sawtooth instability is well established in present-day tokamaks, most efficiently by using ECCD, and this can be implemented on larger devices like ITER/DEMO.

Considerable attention is now focused on investigating the complex (N)TM seeding process using a variety of diagnostics, including ECE, that enable a detailed insight into the temporal dynamics of the process. This has helped in identifying several of the important factors that play a role in seeding an (N)TM, such as the amplitude of the ideal (1, 1) mode, the sawtooth crash size, the relative position between the resonant surfaces as indicated by the value of q_{95}, the magnitude and direction of plasma rotation and its gradient. At a given value of β_N ($= \beta/(I_p/aB_t)$), the 'normalized-β', the sawtooth period is an important critical factor in determining the triggering of an (N)TM. For ITER, a natural sawtooth period of 20–50 s is predicted by transport modelling. At the target operating pressure for the ITER ELMing H-mode scenario (with $\beta_N = 1.8$), scaling predictions based on measurements in present tokamaks give the critical sawtooth period to be around 70 s, which is higher than the predicted natural sawtooth period. In principle, therefore, ITER should be free of (N)TM triggering. However, this is not guaranteed, since an (N)TM can be seeded even by a sawtooth of shorter period, as observed in some present experiments.

Thus, a provision for sawtooth control needs to be established in ITER. ECRH and ECCD have been found to be effective in influencing the sawtooth period (τ_{ST}) and can be used to control the sawtooth period in ITER and prevent disruptions. Dedicated simulations using codes such as SCENIC and HAGIS also show that ICRF heating can be used to reduce the sawtooth period in ITER to below 50 s, which can allow plasmas to operate with β_N above 2 without triggering (N)TMs. ICRF heating can also be modulated to pace sawteeth at around or just above the natural sawtooth period. This would allow the timing of the crash to be known in advance, which can be exploited for efficient (N)TM

avoidance. The sawtooth period can also be controlled by heating power modulation, even in cases where there is a significant lengthening of the sawtooth period due to EPs. These and other experimental and simulation findings have helped to define and develop several sawtooth control strategies that have been well tested on present day tokamaks and give confidence that sawteeth can be effectively controlled on ITER.

5.2.2 Neoclassical tearing modes. NTMs are closely related to the classical tearing modes in being current-driven and involving the slow growth of magnetic islands around the low-(m, n) rational surfaces. The current source in this case is the bootstrap current: in the presence of a small island, the very high parallel thermal conductivity that connects the inner and outer flux surfaces of the island causes the pressure profile to flatten locally, which reduces the bootstrap current within the island and increases the instability drive. This corresponds to a negative current perturbation within the island that produces a magnetic field perturbation which strengthens the island magnetic field and increases the size of the island. Unlike the classical tearing mode, an NTM instability requires the presence of a finite-size (i.e., 'seed') island and is therefore a subcritical instability. In tokamaks capable of producing high temperature plasmas, where the plasma is collisionless, NTMs limit the attainable β to values well below that predicted by ideal MHD. Their control and suppression are therefore crucial for the successful operation of long-pulse plasmas and reactor-grade devices like ITER.

The threshold size of the seed island required for growth of the neoclassical instability and the details of the nonlinear evolution of the mode to a saturated final state are influenced by additional physics such as Δ' (the stability index of the corresponding classical tearing mode), the thermal conductivity of the plasma, plasma inertial effects, kinetic contributions, etc. There have been considerable new physics developments on this topic since the IPB and subsequent review articles that have furthered our basic understanding of the instability, as well as enhanced the ability to control them (see [25] section 2.2). These include the role of plasma rotation shear, the inclusion of self-consistent neoclassical and transport effects, kinetic effects on the seed island formation, and novel mechanisms of seed island generation such as turbulence seeded islands. These have been incorporated in improved 2D and 3D modelling of the instability that have been benchmarked against extensive experimental data, enabling reliable prediction and early detection of the instability. Based on such improved understanding, a reduced physics-based model, the MRE, has been further refined to aid real time control of the instability (as discussed in [25] section 2.2.2.3).

The physics of seed island formation is one of the critical issues in the study of NTMs. Recent investigations have focused on two main areas, namely, determination of the polarization current contribution in terms of collisionality, island size and rotation frequency, and the mechanism of turbulence generated islands. Inclusion of kinetic effects such as finite banana width, finite orbit width and FLR tends to reduce the polarization current contribution to threshold physics. Turbulence-seeded magnetic islands have been primarily studied with 2D/3D reduced MHD and em gyrokinetic simulations, and these nonlinear phenomena are predicted to occur at island sizes at, or below, the diagnostic spatial resolution. Experimentally, therefore, they might be observed as 'triggerless' NTMs.

An additional important effect, particularly in burning plasmas, is that associated with fast particles (see also [28] section 5.1). 3D resistive MHD codes have been improved with the inclusion of fast particle contributions and self-consistent neoclassical effects, such as the modified bootstrap current, two fluid effects, and H&CD sources. The rotation profile within the plasma is a further significant aspect that is important not only with respect to turbulence and polarization current, but also on account of the effect of NTV torque on the 3D magnetic islands. Experimental evidence from several tokamaks shows that equilibrium shear flows also have a strong influence on NTM threshold and saturation levels. Under appropriate conditions, sheared flows appear to have a benign influence on NTMs, possibly offering a further approach to control or avoidance of the instability.

Although there have been significant advances in the physics understanding of (N)TMs since the time of the PIPB, there continues to be sustained experimental and theoretical efforts towards elucidating the physics at small island size and predicting the onset of (N)TMs. In this respect, the MRE remains the most successful physics model, with tests in numerous tokamaks in recent years and its reliable use for real time control. To expand the scope of MRE-based analysis, enabling studies of both stable and unstable (N)TMs, a new formulation of $\Delta'(w)$ has been implemented which combines the evolution at zero and small island width derived from the classical tearing mode dependence with the that at large w, where the neoclassical drive starts to dominate. Although the effects of (N)TMs can be observed, measured and controlled only when the island has grown to a size at which the bootstrap drive starts to dominate, exact models are not required to allow control of (N)TMs: a model need only be 'good enough' to enable the implementation of feedback control, with adaptive models to compensate for uncertainties in the measurements and model. Experience has confirmed that the MRE is very reliable in reproducing and explaining the main physical effects and in determining the strategy for (N)TM pre-emption, avoidance and control. Despite the many successes on present experiments, development of a detailed strategy for (N)TM control in burning plasmas is an outstanding issue: it remains the goal of the wide-ranging studies aimed at

furthering understanding of such fundamental physics issues as the role of turbulence and the interaction between magnetic reconnection and micro-turbulence, including (gyro-) kinetic effects from electrons, ions and fast particles.

5.2.3 Resistive wall modes. RWMs are a class of instabilities that arise from the presence of an external conducting wall and can be either ideal or resistive in nature. Accordingly, they are identified as an 'ideal plasma RWM' (or RWM for short) and a 'resistive wall tearing mode'. Their global eigenfunctions closely resemble kink/ballooning eigenmodes with modifications arising from the mode coupling with the external conducting structure. This coupling restricts the mode growth rate and its rotation frequency to $O(1/\tau_w)$.

As discussed in [25] section 2.3, the RWM is considered the most severe of these instabilities and has therefore been extensively studied to elucidate its physics underpinnings and to devise means of controlling it. A marked characteristic of the RWM, in sharp contrast to a global kink ballooning mode, is that it becomes less stable when the plasma is brought closer to the conducting wall. Early theoretical models based on the MHD descriptions, therefore, sought to passively stabilize the mode through rotation and coupling to higher frequency damped modes, such as shear Alfvén or ion acoustic waves that were Landau damped or experienced sound wave continuum damping. Such a dissipation mechanism was shown to produce RWM stabilization at, or above, a critical plasma rotation frequency. Subsequent experiments validated some of these theoretical results but also revealed significant differences from earlier experimental observations. In particular, there was increasing evidence from experiments in DIII-D and NSTX that the value of the RWM critical plasma rotation frequency published previously was not the result of RWM stabilization physics. These discrepancies between theoretical modelling and experimental findings were suitably addressed by kinetic extensions to the ideal MHD dispersion relation, including effects due to trapped and circulating ions, trapped electrons and Alfvén damping.

The development of this kinetic MHD model constitutes one of the major advances in the field, as it exhibits the highest qualitative and quantitative success, both in reproducing RWM stability behaviour and in defining stability boundaries in the state space of relevant plasma parameters, e.g., β_N, plasma rotation profile and magnitude, pressure profile peaking, plasma internal inductance and plasma collisionality. The kinetic-MHD model established that RWM stabilization is due to the combined effect of plasma rotational inertia and dissipation. The model helped in refining (and lowering) the estimates for the critical rotational frequency, and also in defining its upper and lower bounds as a function of the distance between the plasma boundary and the conducting wall. These results were further complemented by experimental observations of alterations in plasma rotation due to NTV induced by non-resonant applied magnetic fields that made it possible to obtain nearly zero plasma rotation across the entire plasma profile, but without the occurrence of MHD mode locking.

The kinetic MHD model has significant positive implications for ITER, which is expected to operate at relatively low plasma rotation and collisionality and where the kinetic stabilizing terms could prove beneficial. Extensive simulation studies of RWM stability under ITER conditions, conducted using codes such as MARS and MISK in which kinetic effects have been added, are helping to delineate the stabilizing contributions of various kinetic effects. These simulations also reveal that, while results derived from the kinetic-MHD model are far more optimistic in relation to RWM passive stabilization than earlier MHD dissipation models, active stabilization techniques will, nevertheless, be required in ITER and in future devices to ensure control of the RWM during plasma state transitions. Active RWM control in tokamaks has greatly benefited from analogous experience on various reverse field pinch experiments and has been successfully implemented on NSTX to achieve very high β plasmas, at near zero rotation and without mode locking. Such success on NSTX and other devices provides great confidence for the successful application of active RWM control to ITER plasmas.

5.2.4 Error field physics and corrections. Error fields, consisting of non-axisymmetric magnetic perturbations, can naturally exist in a tokamak due to imperfections, construction misalignments and asymmetries in the current feeds to the primary poloidal or toroidal coils. Eddy currents and mechanical forces during plasma operation, as well as the presence of ferritic material in the tokamak structures, are further potential sources of EFs. They can arise, moreover, as sideband contributions of externally imposed fields such as RMPs for ELM control. As discussed in [25] section 2.4, their presence can have significant deleterious effects on tokamak performance, primarily due to braking of plasma rotation, leading to locked modes and plasma disruptions. A proper identification and measurement of EFs and their active compensation through EFC coils therefore remain a major concern in both present experiments and for future devices like ITER.

This was well recognized and reported in the PIPB [43], where it was stated that EFs as small as $\delta B_r/B_0 \sim 10^{-4}$ could lead to a major disruption. Since then, this high sensitivity to a small EF has motivated a considerable range of experimental and theoretical research, allowing a deeper understanding of the underlying physics and the development of better EFC algorithms. One of the major advances has emerged from the realization that the plasma response to the EF perturbation plays an important role and needs to be taken into account in

determining the EFC criteria. This has led to a substantial improvement over the earlier 'vacuum 3-mode criteria' that were based on the superposition of EF spectral components in vacuum, neglecting the plasma response. The deformation of the plasma caused by the EF, also known as resonant field amplification, is a further important factor that must be incorporated to arrive at a physically valid metric for an EFC criterion. Field penetration leads to large nonlinear islands which are considered to be the origin of locked modes.

Thus, development of a reliable EFC system depends critically on a proper prediction and assessment of field penetration. The primary goal of EFC is the avoidance of locked modes, while also minimizing rotational damping through NTV. Rotational damping can destabilize a range of instabilities at various scales, potentially leading to degradation of particle, momentum and energy confinement through both thermal and energetic particle transport channels. This was confirmed from an experimental mock-up of an ITER TBM on DIII-D that generated a local magnetic ripple, $\delta B/B_0 > 3\%$. These experiments also revealed the role of the $n = 1$ component in causing the major degradation and the importance of compensating for this component to mitigate such degradation. The experimental conclusions have also been validated in codes like IPEC and IPEC-PENT. Advances in precision metrology are expected to ensure that the static EF environment in ITER will be well characterized prior to initial plasma operation. Based on the progress on EF physics made in the last two decades and a greater appreciation of the importance of the 3D plasma response, an advanced EFC strategy and design is being put in place. The aim will be both to minimize construction errors and to provide sufficient EF compensation to avoid the disruptive locked mode while imposing dynamic EFC to compensate for residual effects such as rotational damping.

5.3 Plasma magnetic control in ITER

As the first reactor-scale magnetic confinement device, ITER faces a unique combination of control challenges that must meet competing demands of high-performance fusion gain, reliable fault handling capability and (very nearly) disruption-free operation in a constrained parametric space. The design requirements of the ITER plasma control system can therefore be considered representative of those applicable to the coming generation of burning plasma tokamak experiments. The ITER plasma magnetic control system has been designed with these goals in mind and development continues towards meeting these demanding challenges (see [25] section 4). It has made significant advances since the PIPB by addressing issues identified in the Design Review and has also accomplished key integrated design tasks. These include a comprehensive analysis of the vertical position control using both in-vessel VS coils and external poloidal field control coils, with an assessment of the maximum attainable control limits in vertical displacement, Z_p, and maximum rms value of low frequency noise in the measurement of the plasma vertical velocity used in the VS feedback loop, $max\langle dZ_p/dt\rangle$. The introduction of in-vessel VS coils, allowing the addition of the so-called VS3 loop for control of the plasma vertical position, has considerably increased the capability of the vertical control system to deal with larger values of Z_p and to accommodate higher levels of noise. A comprehensive study of the level of low frequency plasma noise expected in the ITER dZ_p/dt diagnostic signal and results from the present ITER-relevant superconducting long-pulse devices suggest that an rms noise target of $0.2\ ms^{-1}$ in the real-time estimate of dZ_p/dt is potentially achievable and is a desirable operational target for ITER.

Extensive simulations of potential ITER plasma scenarios using the DINA code (assuming a Be first wall, as foreseen in the 2016 ITER baseline) indicate that the fiducial 15 MA scenario is eminently achievable within the operational limits of maximum voltage, current, and power step amplitudes in superconducting coils. On-going studies include simulations of plasma magnetic control before and during disruptions in 15 MA and 7.5 MA H-mode plasmas (with the formation of REs) using feedback control of the plasma vertical position by the VS in-vessel coils to determine the limitations of the control system in handling emergency shutdown scenarios.

6. Divertor and scrape-off layer physics

Mastering the intense interactions between the plasma and the vessel walls is a key issue for efficient and reliable operation of next step fusion devices. Indeed, the plasma facing components will be subject to unprecedented heat and particle loads compared to present day devices (see, e.g., [44]). PFCs will have to ensure adequate heat exhaust capability while avoiding core plasma contamination from material eroded from the PFC surface; the divertor should also provide exhaust of helium ash from the DT fusion reactions. In addition, PFCs should have sufficient lifetime to allow high duty operation of next step devices over time periods of several years. Predicting heat and particle loads to PFCs and the associated PWI requires understanding of processes that govern plasma properties in the edge boundary plasma (i.e., the SOL and near-pedestal region of the confined plasma)[1]. PWI include several processes, such as material erosion, transport in the plasma and redeposition at the PFC surfaces, fuel retention in the vessel walls, dust production and PFC

[1] The reader is referred to Fig. 1 of [26] for an illustration of the configuration of the edge plasma, SOL and PFCs (divertor and main chamber).

damage. These processes govern PFC lifetime, have safety implications (as the radioactive tritium in-vessel inventory and dust production must be limited for safety reasons) and can impact core plasma performance.

Significant progress has been made in this area since the previous review in the PIPB [45], based on experimental data from fusion devices but also from high heat flux facilities and linear devices simulating plasma edge conditions. The R&D activities have also benefited from further development of sophisticated plasma edge and PWI simulation tools. In addition, improved diagnostic capabilities, such as fast turbulence measurements, spectroscopic imaging techniques and laser-based diagnostics for fuel retention, allowed new light to be shed on the complex and often localized and non-axisymmetric physics processes in the SOL. Within this special issue, [26] covers the progress in SOL and PWI physics, both from the experimental and modelling points of view, in the following areas:

- SOL physics and divertor transport (section 2);
- Managing steady-state and transient heat loads (sections 3, 4 and 5);
- Material migration, dust production and fuel retention/recovery (sections 6 and 7);
- PFC damage during nominal operation and off-normal events (sections 8 and 9).

The following discussion presents a brief overview of the context of R&D on SOL physics and PWI during the period since publication of the PIPB (section 6.1) and summarizes the main findings (section 6.2) as well as the remaining open issues in this area (section 6.3). A more detailed presentation can be found in [26], which provides detailed references to the original research.

6.1 R&D in plasma edge physics and plasma-wall interactions: background

The update of the ITER divertor material strategy in 2013 caused a major evolution in the focus of the R&D program in the divertor and SOL physics area relative to that represented in the PIPB: the ITER project originally foresaw a divertor configuration consisting of a combination of C and W PFMs for the non-nuclear phase and a divertor using full-W PFMs for the nuclear phase. However, in the frame of the ITER cost-containment policy, supported by the experience acquired up to that time from operation in tokamaks using a divertor with full-W PFMs, it was decided to implement only a single divertor for the entire operational phases up to the achievement of the $Q \geq 10$ milestone in the nuclear phase. The initial divertor configuration was therefore eliminated and replaced by a divertor with full-W PFMs, with the aim of retaining this divertor until a DT fusion gain of $Q \geq 10$ was achieved, while using Be PFMs on the main chamber first wall (see, e.g., [18, 19]). The use of a divertor with full-W PFMs in the nuclear phase, i.e., when operating with DT fuel, was motivated by the significant fuel retention linked to the use of C PFMs. While this is not an issue for current devices operating only with hydrogen or deuterium, the use of C PFMs was not deemed compatible with next step devices operating with DT fuel mixtures, since maintaining an in-vessel tritium inventory below the regulatory safety limit would have required an unacceptably high fraction of operational downtime for fuel recovery.

The results of the R&D program reviewed in this special issue reflect this evolution, focusing on the impact of replacing C by W in the divertor. Indeed, W is considered as the most promising material for DEMO- and FPP-class devices, since it is one of the most resilient materials under high heat fluxes and neutron impact. In addition, it is not prone to hydrogenic fuel retention compared to low-Z materials such as C or Be. Tungsten is, however, a less 'forgiving' material than C in tokamak operation. Carbon is a good radiator in the edge plasma, where it contributes to taming heat loads on PFCs, and can be tolerated in the plasma core up the percent level without degrading core plasma performance. In contrast, W radiates deleteriously in the plasma core and must be strictly controlled to avoid degrading performance (the maximum allowable W concentration is around several $\times 10^{-5}$ in the plasma core of ITER for $Q \geq 10$ operation). In addition, while C sublimates under excessive heat loads, W melts, which, in the most severe cases, can result in not being able to operate with the plasma strike points at the damaged location. The additional risks linked to starting operation with a full-W divertor were therefore scrutinized, in particular, in terms of the lifetime of the divertor W-PFMs, in relation to the achievement of radiating regimes in plasmas with essentially no C impurity content, with respect to the control of W erosion and W core contamination, as well as to confirm the expected benefits in terms of fuel retention.

Major upgrades in the fusion facilities also followed, with a significant number of devices transitioning from C to all-metal PFMs during the period reviewed, in particular[2]:

- AUG pioneering operation with full-W PFMs, with a progressive change from C to W from 1999 to 2007 [46], following initial tests in 1996 [47];
- JET operating with a Be first wall/W divertor combination (so-called JET-ILW) from 2011 onwards [21];
- Tore Supra changing from a C limiter configuration to a full-W divertor configuration in 2016 [48] with a full actively-cooled ITER-grade divertor in 2021 [49];

[2] The reader is referred to Fig. 53 of [26] for an illustration of key devices operating with all-metal PFMs.

- EAST and KSTAR operating with actively-cooled divertors utilizing W PFMs since 2014 [50] and 2023 [51] respectively.

This has allowed major progress in experimental characterization of operation in devices with all-metal PFCs, in particular, in the field of SOL physics and PWI, and has provided confidence that reliable operation can be sustained in ITER with a full-W divertor. A change from C to W PFMs is being considered in several further devices, such as the JT-60SA (Japan) and DIII-D (USA) tokamaks, and the W7-X stellarator (Germany). Tungsten is also the reference PFM for several large-scale devices under design or construction, such as DTT (Italy) [52], SPARC (USA) [2] and STEP (UK) [13].

The ITER design is at present being redefined, with the current combination of a Be first wall and W divertor making way for a configuration with full-W PFMs, as part of its new baseline concept. This will bring the ITER material choice into line with that expected to be favoured in DEMO-class devices. This will reduce the emphasis on PWI studies related to the use of Be PFMs in the future, as has already occurred for C. Nevertheless, results of R&D obtained using Be are reviewed in [26], since the Be/W combination was an element of the ITER baseline configuration over the period covered by the review. In particular, the JET-ILW with a Be wall provided many key results for plasma operation with a fully metallic first wall and divertor.

6.2 Progress achieved in plasma edge physics and plasma-wall interactions

Progress has been achieved during the review period in a wide range of topics from near-SOL and far-SOL transport to PWI issues, including tritium retention, in which significant advances have been made. Several are highlighted in the following discussion in view of their potential impact on ITER and on the integration of burning plasma performance with acceptable requirements for PFCs and PWI-related issues. A more extensive overview of progress made in this field is presented in [26] together with the associated references. In the following, section 6.2.1 outlines progress in understanding plasma edge transport, section 6.2.2 reviews steady-state and transient heat loads on PFCs, as well as potential associated PFC damage, while section 6.2.3 focuses on approaches to the dissipation of steady-state heat loads in highly radiative regimes, section 6.2.4 highlights progress in the investigation of PWI processes, and section 6.2.5 briefly summarizes some of the most salient progress in modelling the edge plasma and PWI.

6.2.1 Understanding plasma edge transport. Establishing a more fundamental understanding of transport processes in the boundary plasma and SOL has been a significant aspect of research studies in this area in recent decades (see [26] sections 2 and 4). An important outcome is that a better picture of turbulence in the edge region has been obtained, showing that turbulence might originate from a region inside the last closed flux surface, at the pedestal bottom near the separatrix. This is likely due to the strong gradients in plasma density and temperature profiles in this region. Different behaviours were identified in the near-SOL versus the far-SOL, where, in the latter case, transport is clearly not diffusive and is dominated by macroscopic filament propagation.

The exploration of far-SOL physics has continued, with a broadening of the SOL density profile (also called density shoulder) evident when operating at high density/gas fuelling in current devices. This could potentially lead to high heat/particle loads on the ITER first wall, as well as to an enhancement of the associated erosion of Be, but extrapolation to ITER conditions requires further studies.

The filamentary nature of ELMs has been further elucidated and the size and velocity of ELM filaments, as well as the resulting power fluxes on the first wall have been characterized in much greater detail. Models for ELM filament propagation and wall power deposition have been developed and extrapolations to ITER evaluated based on these models with empirical inputs. These extrapolations indicate that the impact of ELM wall loads in ITER are dominated by wall erosion due to ions arriving with high energy at the wall, rather than by the ELM-related averaged power fluxes. It is expected, therefore, that H-mode scenarios that provide acceptable transient loads for the divertor (small ELM/no-ELM, ELM suppressed scenarios …) will also lead to acceptable fluxes to the first wall.

The role of drifts and flows, which can influence the divertor plasma solution, has been further investigated, both experimentally and through simulations. Experimental results do not yet offer a unified picture, with a significant effect of drifts evidenced in H-mode in DIII-D, while studies in AUG concluded that drifts had a small impact. It was shown, however, that including drifts in plasma edge simulations generally improves agreement with experimental data for current devices, in particular with respect to the matching of in/out divertor asymmetries. For larger-scale devices, such as ITER, which will run at higher power in semi-detached regimes, present simulations indicate that the impact of drifts and in/out divertor asymmetries could be less pronounced, but this requires further confirmation.

6.2.2 Characterization of steady-state and transient heat loads and associated damage on PFCs. Progress has been achieved in understanding and estimating the steady-state heat flux pattern on PFCs, in particular on the divertor, which is the most heavily loaded component (see [26] section 3). Results obtained in H-mode are of particular interest to ITER and to future burning plasma devices more generally.

The SOL heat flux width is a key parameter for deriving the divertor heat load pattern. An experimental multi-machine scaling for the SOL heat flux width was established for inter-ELM attached H-mode plasma conditions. Experimental data from current devices are consistent with a heuristic drift model. Direct extrapolation of the scaling to ITER parameters predicts a very narrow SOL width ($\lesssim$1 mm), resulting in highly peaked and severe plasma heat loads on the divertor. Reducing such loads would require operation with very high separatrix densities (see the discussion on dissipative divertor operation at section 6.2.3 below) and the compatibility of such a regime with the sustainment of high confinement in H-mode is an open issue. However, certain edge turbulence codes applied to ITER conditions indicate that the SOL width could be significantly larger (several mm), due to the dominance of a different turbulence regime (electron dominated turbulence) from that prevalent in current devices (blobby ion-dominated transport). These predictions need further consolidation and experimental results confirming their physics basis, which are just starting to be available.

With respect to limiter heat loads, studies during the plasma start-up phase found evidence for a narrow near-SOL width on the HFS limiter configurations. This result, which was not available at the time of the PIPB, was a key input for optimizing the ITER HFS first wall panel design for plasma start-up.

In addition to steady-state heat loads, PFCs must also withstand transient heat loads produced by various core plasma phenomena. As noted above in section 4, the transient loads due to ELMs are expected to be challenging for PFC integrity in ITER and burning plasma devices. ELMs expel short pulses of high heat and particle fluxes onto PFCs with an expected frequency of several Hz in ITER (see [26] sections 4 and 5 for further details).

A multi-machine ELM heat load scaling was established and compared to PFC thermo-mechanical limits, showing that an efficient ELM control system (mitigation or suppression) is required for ITER. Indeed, as evidenced from PFC testing in high heat flux facilities, PFC damage from transients (including surface roughening and crack formation) appears even under moderate ELM-like loads when a large number of transients is applied. This represents major progress since the PIPB, since only ELM energy scalings (and not absolute heat loads) could be established at that time. The newly developed ELM heat load scaling also showed that mitigation of type-I ELMs (e.g., by increasing ELM frequency) had a very limited effect on the peak ELM load at the divertor, which suggests ELM-suppressed or small ELM/no-ELM H-mode scenarios as the most viable option for H-mode scenarios in burning plasma devices.

ELM control methods have therefore been explored extensively in recent years, in particular using RMPs following the inclusion of RMP coils in the ITER baseline configuration (see, e.g. [18, 19]). The associated impact on quasi-stationary divertor heat loads was also studied, showing splitting of the divertor target heat load patterns consistent with the magnetic perturbation applied. This poses specific challenges for access to dissipative divertor regimes across the divertor target. Application of RMPs will therefore require fine tuning in ITER for efficient ELM control while keeping manageable divertor heat loads, and this optimization remains an open field of research.

Other potential transient heat loads on PFCs originate from disruptions or VDEs, where, as outlined in section 5.1.2, plasma confinement is lost rapidly and a large fraction of the plasma stored energy is dissipated on the first wall on very short time scales. These events are benign in devices with little stored energy, but scale up with device size. They can cause severe PFC degradation and damage in large-scale fusion devices, requiring the development of efficient disruption avoidance and mitigation schemes. These key issues are not reviewed in [26] but are discussed in [25] section 3.

As discussed in [26] sections 8 and 9, a thorough evaluation of PFC damage under ITER-relevant combined heat and plasma loads was performed using high heat flux test facilities, linear plasma devices and tokamaks. The output from these studies has been used for defining targets for heat load mitigation required, in particular for ELMs.

Earlier PFC damage studies were extended to higher numbers of cycles for both steady-state and ELM-like heat loads, and to address the combination of plasma and heat loads, allowing further investigation of synergistic effects. First studies of W PFC behaviour operated above the W recrystallization temperature (which can lead to a degradation of the W thermo-mechanical properties) have also been performed although this topic remains the subject of ongoing research.

Code packages used to predict PFC melting and subsequent melt layer motion have been validated against tokamak experiments under a variety of relevant regimes (excessive steady-state or ELM-like loads), leading to increased confidence in the accuracy of their predictions to ITER. On the other hand, efforts on the complex modelling of RE impact on PFCs (discussed in section 5.1.2) and its validation have only recently been launched, but with encouraging results for Be PFCs.

6.2.3 Development of dissipative divertor regimes. Significant progress has also been achieved in exploring partially detached divertor regimes, using impurity seeding to radiate the heat in the plasma edge and spread the divertor power loads, reducing them to values manageable by metallic PFCs. Operation with a partially detached divertor is the reference regime for the ITER full power nominal scenario

with $Q \geq 10$. The seeding impurity mix[3] to be used in ITER should ideally maintain detachment, allow robust detachment control schemes, buffer transients, minimize fuel dilution and radiation in the core to avoid core performance degradation (in particular, maintaining sufficient power loss across the separatrix to remain above the H-mode power threshold) while minimizing PFC erosion and fuel retention. Section 3 of [26] discusses the significant progress achieved in this area, which is summarized below.

Earlier work involving mainly nitrogen (N_2) seeding has been extended to neon (Ne) and argon (Ar) seeding, which are foreseen as divertor radiators for ITER (N_2 seeding is not an option favoured for ITER, to avoid the formation of tritiated ammonia, which is challenging to handle in the gas exhaust system and in the tritium processing plant). Experiments and simulations have shown a favourable device size scaling of Ne as a radiator for detached operation with high input power associated to the relative scale lengths of impurity and hydrogen ionization. While predictions show that Ne should provide suitable conditions for radiative divertor operation in ITER, the optimal impurity mix under ITER conditions is still under investigation.

A regime denoted 'X-point radiator', involving radiation concentrated in a small region around the X-point and exhibiting similarities to a stable MARFE (a toroidally symmetric region of highly radiating cold edge plasma), has been demonstrated in several devices. The XPR has been shown to be an ELM-free regime under certain conditions and is controllable over long timescales. Although not the reference regime foreseen at present in ITER, the XPR could be of interest, since it provides a simultaneous solution to the stationary and transient (ELM) heat exhaust problem, assuming that it also satisfies other requirements, such as providing an adequate helium exhaust rate.

A better understanding of atomic and molecular processes involved in radiating regimes (with, for example, improved cross-sections for plasma-neutral interactions at low plasma temperature) has contributed to improving plasma edge modelling. Both simulations and experimental results indicate that operation at high density high neutral divertor pressure is beneficial for plasma exhaust, while the compatibility of scenarios at low separatrix density with acceptable divertor heat loading could be challenging.

First successful attempts for real-time control of divertor plasma detachment have been performed, using extrinsic impurity injection as an actuator to ensure manageable divertor steady-state heat loads. This is an essential requirement for reliable operation of ITER and long-pulse burning plasma devices.

[3] The reader is referred to Fig. 7 of [26] for a good illustration of the radiating capabilities of the various impurities considered.

6.2.4 Progress in understanding PWI processes. As reviewed in [26] sections 6, 7 and 8, studies of plasma-wall interactions over the past ~20 years have, to a significant extent, been focused on the use of W and Be as PFMs as foreseen in ITER. A particularly important result which has emerged from these investigations is the demonstration of significantly lower fuel retention in devices with all-metal PFMs than is observed in devices using C. These studies have also yielded progress in predicting fuel retention/recovery in Be deposits, which are expected to be the main driver for fuel retention in a Be/W configuration. It should be noted that the fuel retention in a full-W configuration, as is now proposed for ITER, is expected to be significantly lower than in the Be/W configuration initially foreseen. However, specific wall-conditioning techniques will likely be required in a full-W configuration, e.g., boronization, in which a thin B coating is applied on plasma-facing surfaces. The impact of residual B on fuel retention is currently under investigation, since the deposited B layer might behave similarly to a Be first wall in relation to erosion and subsequent codeposition of tritium.

Progress has also been made in understanding local (e.g., W prompt redeposition) and global material migration, showing that, in contrast to C, W and Be do not exhibit long range transport to remote areas of the vessel (i.e., areas not directly exposed to plasma), due to the different erosion/deposition processes involved.

Characterization of helium-W interactions, in particular of W fuzz formation, has been an important element of PWI R&D activities. These studies have shown that W fuzz is predicted to occur in a very narrow region of the ITER divertor under a restricted range of plasma parameters (helium fluence, helium incident energy, wall temperature) where conditions for fuzz formation are met. Compared to processes observed in linear devices, a complex interplay between W erosion, particularly from impurities and ELMs, and W fuzz formation has been observed under tokamak conditions. This could lead to a saturation of the W fuzz layer thickness, should it appear.

The physics basis for the detailed shaping of the ITER divertor has also been established within the scope of these studies (the ITER final choice is a toroidal bevel, protecting poloidal leading edges of divertor components) and has been demonstrated in tokamak experiments. A variety of relevant phenomena were observed experimentally, including optical hot spots, which are predicted to occur in ITER at the crossing of toroidal gaps between divertor components. As noted above, in section 6.2.2, the physics basis for the detailed shaping of first wall panels, in particular, for the start-up phase of the discharge, was also established.

6.2.5 Progress in modelling the boundary plasma and PWI. The significant progress made in modelling this complex area of fusion plasma physics over the review period should also be highlighted. This is due to a combination of more powerful HPC, allowing larger scale simulations and/or the advance from axisymmetric 2D geometry to a full 3D simulation which includes local effects, and to the implementation of new and more accurate physics models. A few examples are listed below (note that the list of codes given below is not exhaustive):

- Fluid plasma edge modelling code packages now include SOL drifts on a more routine basis. In addition, some code packages can be run:
  - with grids extended to the walls to model more realistically the far-SOL physics and the wall loads (SOLPS-ITER, SOLEDGE, …);
  - in 3D to handle RMP or stellarator cases (EMC3-Eirene, …);
  - using a refined molecular and atomic database for processes involved in detachment physics;
  - to tackle ITER- and even DEMO-sized simulations, although this is still extremely demanding in computational time.
- First-principle based 2D/3D edge turbulence codes are now becoming available (TOKAM3X, GRILLIX, GBS, BOUT++, …);
- More sophisticated codes for material migration are now available in 2D and 3D (WallDYN, ERO2.0, DIVIMP, Guitr, …);
- Dedicated codes are also being developed to model the impact of PWI on PFMs on the micro- or macro-scale. This includes, for example, tools to model material evolution under plasma exposure, such as the Xolotl code package, fuel retention in materials (FESTIM, MIMHS, TMAP …), W evolution under helium exposure (ballistic implantation of helium into the W bulk through the growing porous fuzz layer), crack formation (TREX), or melting (MEMOS-U and MEMENTO).

6.3 Remaining R&D issues related to plasma exhaust

Although substantial progress has been achieved towards developing the experimental capability to ensure manageable plasma exhaust in next step and ITER-grade fusion devices, several open issues remain to be resolved by future R&D, particularly for the more challenging conditions of DEMO-class devices. Some of these issues have also become of interest to ITER with the recent evolution to a full-W configuration.

Short term high priority R&D areas for the ITER full-W configuration include:

- Optimization of a boronization system for vessel conditioning purposes and the associated impact of B on fuel retention/removal. This system was not planned for the Be/W configuration of ITER initially foreseen, as Be is an efficient oxygen getter, but is deemed necessary for efficient operation of ITER with full-W PFCs.
- Investigation of the start-up phase of the discharge on W limiters, including the requirements on boronization in that context.
- The impact of W sources from the main chamber first wall (including the impact associated with specific heating and current drive systems, such as ICRF-related W sources) and the compatibility in terms of core W contamination.

The last two items are common to ITER and DEMO, while the first is presumably relevant only for ITER.

More generally, high priority open issues in the field of plasma exhaust requiring further R&D studies include:

- Further consolidation of the SOL width under detached H-mode conditions and ITER-relevant conditions for turbulence, to resolve the discrepancy between the current empirical scaling and the prediction of turbulence modelling.
- Refined predictions for the far-SOL physics and the associated main chamber wall loads, since the divertor loads are much better characterized than the main chamber loads at present.
- Continuation of studies to determine the best impurity mix for seeding in ITER (Ar/Ne) and to ensure detachment control in reactor-like conditions in terms of the available sensors/actuators.
- In addressing DEMO-like conditions, core radiation (using, e.g., krypton or xenon) should be investigated in addition to the more usual exploitation of plasma edge radiation, to cope with the increased edge power flow. Given the magnitude of this issue for DEMO, highly radiative divertor regimes (e.g., XPR), alternative divertor configurations (e.g., using magnetic configurations with increased connection lengths, such as super-X or snowflake divertors), and alternative PFMs (e.g., advanced W alloys or liquid metals) are also being investigated.
- Extension of PWI studies to ITER-relevant high duty cycle conditions to consolidate the basis for estimating the lifetime of the divertor and other PFCs: high particle fluence (deuterium/helium/impurities), large numbers of cycles under high steady-state heat loads and ELMs. Requirements are even more stringent in DEMO-class devices, where PFC erosion rates must be limited to significantly lower levels.
- Finally, while the impact of neutrons on PFMs is expected to be limited for ITER (~1 displacement per atom (dpa) at end of life), this is a key issue for DEMO (several tens of dpa) which can impact many aspects of plasma-material interactions, including tritium retention

and schemes for its removal. This is a key area for developing future fusion power plants which requires specific facilities for material testing under relevant fusion neutron fluence (such as the IFMIF-DONES facility [53] currently starting construction). Other concepts are also under discussion at present, such as a VNS, based on a dedicated medium-sized fusion facility with external heating which drives a thermonuclear DT plasma to provide relevant neutron fluences for testing key components, such as the tritium breeding blankets.

7. Integrated operation scenarios

The mission of the Integrated Operation Scenarios (IOS) TG is to develop controlled and predictable operating scenarios for present and future fusion devices. Integration, which builds on the knowledge base of and collaborates with the other six ITPA TGs, is a critically important aspect of this work. This is essential when planning for operation in a new device, which will not recognize divisions within the plasma such as core/edge, transport/stability, etc.

Although R&D activities in this group have targeted successful operation of ITER through all phases of the IRP [18, 19], the knowledge gained is largely generic and can apply more broadly to a wide range of future devices. Activities within the IOS TG have leveraged joint Integrated Modelling efforts and Joint Experiments to address critical aspects of future experiments in preparation for burning plasma operation.

The TG's efforts, which are reviewed in detail in [27], have focused on most aspects of integrated tokamak operation, including robust and reproducible plasma initiation, development and validation of operating scenarios, actuators and scenario control. The emphasis of all work within the TG has moved increasingly towards the development and validation of predictive models in an integrated framework to address the challenging task of preparing to operate devices that are well outside the experimentally established operating space.

Moving toward next-generation devices like ITER, even the initial phase of a discharge presents new challenges, which are discussed in [27] section 2, where progress made in their resolution is detailed. These devices will have large volumes, thick metallic vacuum vessel walls, and superconducting coils. With these restrictions, the achievable electric field for plasma breakdown and burn-through will be very low. Operation with a very low prefill density could ease the breakdown, but at the expense of increased risk for RE generation. As the prefill density is increased, the danger of RE generation is reduced, but breakdown and burn-through become even more difficult, necessitating the addition of external heating power. Electron cyclotron heating preionization, planned for ITER, will ease these challenges, but the injected power must be limited to avoid damage to in-vessel components due to low absorption in the still diffuse plasma. In the light of the variety of restrictions, the operating window is small. This has been addressed by investing considerable experimental effort in developing and demonstrating optimized techniques for reliable plasma initiation, burn-through and ramp-up to flattop.

Ensuring controlled termination of burning plasmas to minimize the risk of off-normal events during the burn termination and current ramp-down presents its own challenges. A survey of termination scenarios was undertaken to document conditions for exit from H-mode (see [27] section 8.4). It was found that all current experiments lie within the same window of normalized parameters, while ITER simulations fall outside of it, mostly because of the (likely) underestimation of radiated power in the simulations and the absence of validated models for density predictions. The activity indicated a need for continuing research on the plasma current ramp-down phase to inform the design of controllable termination scenarios for ITER. Experiments to support this area of research need to mimic the exit from H-mode in the presence of α-particles and the challenges of controlling large radial shifts at the H-mode exit, due to the large drop in β_p.

Several different classes of operating scenarios have been considered for the current flattop. Low-activation scenarios were developed with ITER's first research phase (Pre-Fusion Power Operation I) in mind, where, within the IRP developed for the 2016 ITER Baseline [18, 19], L- and H-mode operation was planned around hydrogen and/or helium plasmas with low heating power, with the aim of commissioning many of ITER's systems and operational capabilities. The recently developed 2024 ITER Baseline and associated IRP (see [1, 20]) foresees that the H-mode mission will now be carried out in deuterium plasmas with a considerable increase in heating power, so that much of this research may no longer be relevant to ITER operation. However, as discussed in [27] section 3, work in this area has yielded considerable understanding of important physics such as isotope effects and the L-H threshold.

Looking forward to the production of fusion power, a range of scenarios has been developed and remains under study, and progress in the major candidate scenarios for burning plasma operation are reviewed in [27] sections 4, 5 and 6. These include the 'ITER baseline scenario' (see [27] section 4), with $q_{95} = 3$ and an anticipated fusion gain of $Q \geq 10$. Experiments and simulations aim at addressing critical issues for the demonstration of ITER-like plasmas in present day experiments and their extrapolation to ITER. This includes, but is not limited to, access to H-mode, including operation close to the transition threshold, operation in ELMing regimes and burning plasma control. Long pulse operation in advanced regimes, with higher q_{95} (typically hybrid and steady-state operation – see [27] sections 5 and 6 respectively) have largely benefited from experiments on the

Asian super-conducting tokamaks, with major contributions from KSTAR, addressing both hybrid and steady-state operation, and from EAST, focusing on steady-state operation. Research on EAST and KSTAR also contributes via dedicated similarity experiments in collaboration with DIII-D. EAST achieved a reproducible, fully non-inductive H-mode duration of about 400 s using only radiofrequency wave heating. KSTAR achieved sustained operation at $\beta_N \sim 3$, and, more recently, developed a new scenario with an ITB regulated by fast ions (FIRE mode), sustained for up to 50 s.

In light of the recent decision to move to full-W PFCs on ITER, joint experiments on devices with all-metal PFCs are especially relevant for the preparation of ITER operation (see, in particular, [27] section 4.2.2). Experiments in AUG indicate that the dominant source of high-Z impurity does not come from divertor tiles, but from the main chamber, and this sets the boundary conditions for plasma performance. Density peaking and impurity accumulation reduce the confinement, which can be recovered with boronization. However, large type-I ELMs appear difficult to mitigate at $q_{95} \sim 3$, therefore motivating alternative scenarios with larger q_{95} to maintain $\beta_N \sim 2.0$ and $H_{98y2} \geq 1.0$. These options are now considered in the proposed new IRP, where the condition of $q_{95} \sim 3$ is relaxed in favor of operating at an edge safety factor value that can achieve the target confinement and fusion gain factor. Discharges with high-Z material also deviate from predictions of the reference empirical scaling of the energy confinement time for H-mode (IPB98(y,2)) and the corresponding confinement enhancement factor, H_{98y2}, which were developed primarily under C-PFC conditions.

For over 20 years, further studies of hybrid scenarios have been pursued across a wide range of plasma operation windows with high performance of $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$, i.e., above the ITER baseline scenario under stationary conditions for periods longer than $5\,\tau_E$. A variety of experimental approaches has demonstrated access to the hybrid regime in ITER-relevant conditions.

Early heating (i.e., typically during the current ramp-up) is routinely used to access hybrid and steady-state regimes, and this approach is applied to simulations of advanced regimes in ITER. While differences exist between simulations – mostly due to differences in individual components in the integrated frameworks used – scenario modelling agrees on the feasibility of achieving $Q > 5$ in (long-pulse stationary) hybrid scenarios with the available H&CD mix on ITER. However, the physics mechanisms for maintaining the q-profile above or close to unity are not yet well understood. Further improvement in understanding is needed before accurate projections can be made to ITER, since this is closely linked to the confinement and stability of hybrid scenarios. Even if the mechanisms are identified, the feasibility of fully exploiting the hybrid scenario in ITER remains uncertain due to differences in dimensionless plasma parameters between the values typical of current experiments and those expected in ITER plasmas. Moreover, the mechanism underlying the observed confinement enhancement must be further clarified. The outstanding issues in extrapolating the application of the hybrid scenario to ITER are discussed in greater detail in [27] section 5.5.

There are two principal options to satisfy the ITER steady-state goals: (i) scenarios with strongly reversed magnetic shear and high bootstrap current; and (ii) scenarios with weakly reversed or low magnetic shear. Experiments have demonstrated significant progress towards the sustainment of high normalized fusion performance in steady-state. However, simultaneous achievement of high performance and steady-state has proven to be very challenging. Furthermore, discharges with ITER-similar plasma shape tend to achieve a somewhat lower performance than discharges with a more strongly shaped equilibrium. In some cases, the very high plasma confinement enhancement achieved in current experiments may be due to turbulence stabilization effects that are not expected to play a significant role in ITER. Limitations in the applicability of current experiments and simulations to ITER are reviewed in [27] sections 6.3 and 6.4.

R&D activities coordinated by the IOS TG are contributing to H&CD research with dedicated benchmarking activities on physics models, as well as dedicated experiments, as discussed in [27] section 7. Examples include lower hybrid wave absorption and current drive, where experiments on EAST have provided insight on the synergy between two frequency sources and between lower hybrid and electron cyclotron waves. Another example is the coupling of ion cyclotron waves, recently expanded with new dedicated experiments on WEST with full-W PFCs. This activity is benefiting from HPC capabilities, which are enabling high fidelity simulations of the radiofrequency coupling, with detailed 3D structures of the antenna.

While some of the activities in support of ITER were targeting the previous IRP, most of the results remain relevant after the proposed re-baselining. For example, integrated modelling for the low-activation phase mostly relies on ECRH and ICRF at 2.65 T (i.e., at 50% of the nominal ITER toroidal field value).

During recent years, the R&D activities of the IOS TG have shifted from scenario design, focusing, e.g., on demonstrating that coil limits are satisfied, to an integrated, physics-based approach where boundary conditions such as pedestal transport are relaxed. This enables more realistic scenarios that address physics issues and are evolving towards core-edge integration. New activities dedicated to specific phases of the discharge have been created, such as plasma initiation and termination.

This shift in the approach to scenario design responds to a need to answer specific operational questions and is driving a transition in this TG, where modelling and experiments are

more closely linked to each other in a truly integrated approach. A key aspect of the TG's mission is the development of integrated control solutions supporting the robust realization and sustainment of plasma scenarios. The fusion community has made remarkable progress during the last two decades towards the development of the control solutions that are required for the reliable operation of burning plasma devices, progress which is reviewed in [27] section 8. Perhaps the greatest achievement by the IOS TG and the broader plasma control community has been to recognize the need for a model-based approach to control design. Such an approach facilitates extrapolation of control solutions tested in present devices to ITER and future fusion reactors. A model-based approach, in contrast to the more usual methodology involving tuning of the control system based on experimental trial-and-error, also facilitates the design of the necessary control architecture and actuator-sharing strategy, enables the physical dynamics of the controlled system to be embedded in the control synthesis for both the feedforward and the feedback components of the control solution, allows for testing of the control solutions via simulations prior to implementation and experimentation, and maximizes the readiness of the control algorithms for initial device operation.

The focus on model-based control has stimulated development of a novel class of physics-based predictive codes that emphasize their application to the simulation of plasma control. Codes such as METIS, COTSIM, and RAPTOR have enabled control-oriented simulations and allowed the derivation of the required control-oriented models to tackle control problems beyond equilibrium control, as noted previously. Section 8 of [27] highlights several examples of progress in plasma control applications enabled by such modelling. Present efforts on devices such as DIII-D are focused on emulating the burn dynamics so that the burn controllers developed can be tested before ITER operation.

8. Energetic particle physics

Burning plasmas are predominantly self-heated by energetic α-particles born at 3.5 MeV, making energetic particle physics a key discipline of burning plasma physics. The temperature and density profiles in burning plasmas are necessarily self-organized, which consequently determines the volumetric distribution of the fusion burn. The physics and stability properties of this self-organized system with predominant α-self-heating constitute a new regime of operation to be tested in ITER and future burning plasma devices. EPs are particles with energies significantly higher than the plasma temperature (measured in energy units). Certain key EP parameters in burning plasmas are encompassed by our experimental database from high-performance plasmas in contemporary tokamaks, such as the normalized EP pressure, β_{fast}, and the normalized EP pressure gradient, $R\nabla\beta_{fast}$. However, other key parameters are substantially different in burning plasmas, such as the EP Larmor radius normalized by the plasma minor radius, ρ^*_{fast}, the EP slowing-down time normalized by the energy confinement time, τ_{SD}/τ_E, or the Alfvén Mach number, v_{fast}/v_A, where v_{fast} is the EP velocity and v_A the Alfvén velocity. Other crucial differences are the shape of the EP profile and the nearly isotropic (α-particles) vs. highly anisotropic (NBI, ICRF) EP distribution functions of contemporary experiments. Due to these differences, qualitatively new interactions between EPs and MHD instabilities can be expected in burning plasmas. Therefore, to optimize scenarios in future burning plasma devices for power generation, a thorough understanding of α-particle physics and, more broadly, EP physics is required. Consequently, EPs are studied in considerable detail on current devices where they are generated using NBI and ICRF heating and, in several specific experimental campaigns, in DT experiments in JET and TFTR.

In addition to energetic ions, energetic electrons that can reach relativistic energies (the REs discussed previously in sections 5.1.2 and 5.1.5) can appear spontaneously and exhibit similar interactions with plasma waves to those observed for energetic ions. Like energetic ions, REs in burning plasma devices will have parameter ranges that cannot be experimentally explored in present devices.

The confinement that results from collisional processes of EPs in tokamaks and stellarators is well understood, including 3D effects. Current research focusses on the interactions between EPs and a plethora of instabilities, which can be driven by the thermal bulk plasma or by the EPs themselves. Furthermore, EPs and instabilities can interact synergistically with other instabilities, with turbulence and with externally imposed 3D fields. High-fidelity predictions of such effects are crucial to ensure optimum performance of burning plasmas and are intensely studied in contemporary EP physics research.

Even in present devices with low fusion power gains, i.e., only weakly burning plasmas, our current ability to predict the drive or damping of modes in the plasma, as well as the ensuing EP transport, is challenged when multiple modes are active at the same time and when synergistic interactions between different EP components occur. Beyond the ongoing research on these topics in existing devices, it will be essential to diagnose instability drive and damping of instabilities, together with transport and losses of EPs, in several burning plasma devices, both in their burning fusion plasma operational phases and in their preparatory non-active phases. Experiments in such tokamaks (and stellarators) will establish an experimental basis to validate the simulation and modelling tools available for the study of EP physics in reactor-scale devices. Ultimately, the goal is to use these validated tools to design and optimize fusion reactors while accounting for EP physics.

Interactions of EPs and instabilities can lead to anomalous, i.e., non-collisional, EP transport from the plasma core to the periphery, leading to a loss of core plasma heating and energy, and to high-energy impacts of EPs onto the first wall or divertor. Important instabilities driven by the thermal plasma that interact with EPs are sawteeth, NTMs, RWMs, KBMs and ELMs. EP-driven instabilities encompass a zoo of AEs, which range from high-frequency modes in the ion cyclotron frequency range down to intermediate-frequency modes in the Alfvén frequency range, and at the lower limit, low-frequency modes in the diamagnetic frequency range, along with their kinetically modified branches. Linear stability analysis can accurately describe the onset of these instabilities and predict their mode structure, whereas nonlinear analysis is required to describe their evolution and saturation levels. If the EP pressure is large, so-called 'energetic particle modes' may appear. These often emerge from the EP-driven instabilities in the least damped parts of the Alfvén continuum. EPMs are modes of the EP population itself and have inherent properties characteristic of the EPs, such as EP orbital drift frequencies. Since EPMs can no longer be treated perturbatively, they require a nonlinear physics description, for example, to capture their rapid frequency chirping, as observed in fishbones. Synergistic interactions among modes as well as interactions with turbulence and 3D effects lead to highly complex plasma behaviour.

The complexity of the interactions between EPs and plasma modes has led to several complementary simulation approaches which attempt to address different aspects of the full problem. Gyrokinetic simulations comprise the most fundamental approach based on first principles, but are so computationally expensive that the evolution of the EPs and the instabilities can be simulated only over short time intervals and small volumes. Computationally faster simulation models require extra modelling assumptions while attempting to keep the essential physics of the simulated problem. Such approaches are kinetic-MHD models, gyrofluid models, reduced models, and semi-analytical models. These modelling approaches and codes are currently under development with the aim of validating them on experimental data from contemporary plasma experiments to prepare for burning plasma operation. Nevertheless, ITER will be indispensable for the establishment of the experimental basis for understanding EP physics in burning plasmas and for the experimental validation of modelling predictions in reactor-relevant regimes, an essential step in the development of an improved physics design basis for fusion power plants. This is a major element in the rationale for the construction of ITER and of other future burning plasma experiments.

The following subsections summarize key aspects of the EPs together with their interactions with 3D effects and thermal- and EP-driven instabilities, including the modelling and diagnostic measurements of these effects. A detailed presentation of the key EP physics issues and progress made in addressing them over the past two decades can be found in [28].

8.1 Energetic particle sources

In burning plasmas, most of the plasma heating necessary to sustain the fusion burn comes from the fusion-generated energetic α-particles, whereas, in contemporary plasmas, energetic ions are introduced by NBI and ICRF auxiliary heating. The EPs slow down due to collisions with the thermal plasma until they become thermalized. The balance between EP birth and slowing down results in complex EP distribution functions deviating substantially from the local thermal equilibrium distribution. The inhomogeneous, non-Maxwellian nature of the EP distribution function provides a source of free energy that can drive instabilities.

The α-particles generated in the DT fusion reaction are born isotropically at 3.5 MeV in the centre-of-mass frame, so that in steady-state the α-particle population is described by an almost isotropic slowing-down distribution function. However, drift orbit effects and, in NBI-heated plasmas, the preferential velocity direction of the deuterium ions produced by NBI make the α-particle distribution slightly anisotropic. An additional possible source of anisotropy in the α-particle distribution function is the velocity-space selective anomalous transport caused by various instabilities, e.g., sawteeth or AEs. The MeV-range energies and the nearly isotropic distribution of α-particles make EP physics challenging to replicate in present devices using EPs resulting from NBI or ICRF heating, which generate strongly anisotropic EP distributions at lower energies. The most intense α-particle populations in magnetic fusion experiments to date were generated during the JET DT campaign in the early 2020s by injecting deuterium beams into a tritium-rich plasma [54], although the Q-value was significantly lower than in earlier experiments at JET and TFTR using 50/50 DT mixtures in the late 1990s.

NBI is a reliable auxiliary heating method installed on most contemporary tokamaks. ITER will be initially equipped with two 16.5 MW NBIs at energies of up to 870 keV for hydrogen injection and 1 MeV for deuterium injection, subsequently to be upgraded with a third NBI for later long-pulse and steady-state operation. Such high energies require negative ion beam acceleration, which has been demonstrated on LHD and JT-60U and will be further tested on JT-60SA. NBIs generate a steady source of EPs at their injection energy in a narrow pitch-angle range, leading to highly anisotropic and inhomogeneous EP distribution functions. Tangential NBIs injecting passing particles can drive current non-inductively, a method known as NBCD. The NBIs at ITER will inject EPs close to the trapped-passing boundary.

An efficient fusion burn requires sufficiently large devices operating at high plasma densities. Under the general

assumption that the density profile in burning plasma will be rather flat, a significant fraction of the beam heating power will be deposited at large minor radius, even for MeV-range injection energies. It is therefore difficult to heat the plasma core of reactor-scale devices with NBI. ICRF heating, on the other hand, can heat the plasma core by tuning the ICRF frequency to ensure power absorption by the targeted species on-axis. 1st to 3rd harmonic ICRF heating scenarios are consequently foreseen for ITER. 2nd and 3rd harmonic ICRF heating scenarios are routinely used for plasma heating on several contemporary plasma devices. Since the power coupling improves with increasing Larmor radius (the so-called FLR effect), it is beneficial (but not necessary) to combine 2nd and 3rd harmonic ICRF heating with NBI, which seeds EPs with large Larmor radii that are further accelerated by ICRF heating. A third ICRF heating scenario involves minority ICRF heating, which targets a minority species with a concentration of up to a few percent. Minority heating does not require the FLR effect and has been demonstrated to be robust and efficient in many fusion devices. Recently, a fourth ICRF heating scenario, the three-ion scheme, has been proposed and has been successfully applied in several fusion devices [55]. The three-ion scheme locates the ion-ion resonance of two main ion species at the desired heating location. In the standard three-ion scheme, the targeted species is a third ion species with a charge-to-mass ratio between the two main ion species and is either a naturally present impurity or a species injected for that purpose. The third species has its cyclotron layer near the ion-ion resonance layer of the main species. This setup exploits the large $|E^+|$ component near the cyclotron resonance layer of the third species, which achieves efficient single-pass damping and allows larger minority ion concentration. In an alternative version of the three-ion scheme, the neutral beam-injected ion population can be used as the third species. Their typically large velocity component parallel to the magnetic field causes a Doppler shift, so they satisfy the local resonance condition. Targeting beam-injected ions with the three-ion scheme has been shown to result in efficient heating localized in the plasma core and efficient generation of MeV-range EP tails in the distribution.

Finally, REs are an MeV-range EP population posing a major concern for burning plasma devices based on the tokamak concept [56]. Energetic electron acceleration in the toroidal electric fields in tokamak plasmas can lead to a runaway phenomenon since the collisional drag on electrons decreases with increasing velocity. A runaway can occur due to an avalanche phenomenon from a small seed population of energetic electrons and the large electric fields induced by the current collapse during disruptions, as noted in section 5.1.2. The resulting RE distribution is a highly directed beam reaching relativistic energies, which can cause local melting where the RE beam hits the first wall. ITER must handle this possible impact on the lifetime of PFCs with great caution, and, as discussed in section 5.1.5, several counter-measures are being developed, including SPI, ML-based disruption prediction and avoidance, high-power ECRH and ECCD actuators, and diagnostics for REs (see, e.g., [25] section 3 and [28] section 4.3). RE physics in ITER will occur in new regimes that are not accessible in contemporary tokamaks. Simulations and modelling based on classical physics have been carried out, but the effects of various instabilities driven by REs still need to be self-consistently integrated into predictive models. Since instabilities driven by REs might help mitigate the issue through enhanced scattering, a better understanding of their role will be crucial in the ongoing search for effective RE control. Progress in understanding RE formation during disruptions and in describing their behaviour and interactions with the residual post-disruptive plasma is discussed in [28] section 12.

8.2 Neoclassical energetic particle confinement with 3D perturbations

Effective neoclassical EP confinement (i.e., confinement in toroidal geometry in the presence of collisional transport) is essential for any burning plasma device to ensure efficient core plasma heating and high fusion performance. Additionally, EP losses to the first wall must be avoided due to the wall heating and high-energy impact collisions that can damage the material. Such first wall heating due to any anomalous EP losses is likely to be more critical than the loss of core plasma performance resulting from a lack of core heating by EPs. EPs are more susceptible to enhanced transport from magnetic 3D perturbations than thermal particles due to their large orbit widths and low collisionality.

The neoclassical slowing down and thermalization of EPs in quiescent plasmas is well established (see, e.g., references in [57, 58]). The α-particles in ITER and future burning plasma devices must eventually heat the fuel ions to ensure the fusion burn. However, they are born at energies substantially larger than the critical energy, such that a significant fraction of the α-particle energy heats the electrons, which in turn must heat the fuel ions. The high-energy α-particles in the electron-collision dominated regime lose energy without substantial pitch-angle scattering until they reach the ion-collision dominated regime below the critical energy where pitch-angle scattering becomes significant. The so-called 'α-channelling' ([28] section 10.4) could, in principle, avoid electron heating by coupling the α-particle energy into waves damped on the bulk plasma ions. However, so far, no α-channelling mechanism with sufficient efficiency has been found.

It is more difficult to confine EPs than thermal particles due to their large drift orbit widths, i.e., their large excursions from flux surfaces. The collisional timescale of EPs is much larger than the drift orbital timescale, such that, to a very good approximation, the EP energy is conserved over a full drift orbit. Additionally, the EP magnetic moment is an adiabatic

invariant. In a perfectly toroidally symmetric tokamak, the toroidal canonical angular momentum would be an exact invariant of the EP motion, providing perfect confinement.

However, in reality, tokamaks are constructed with several elements breaking the toroidal symmetry, leading to the loss of the invariance of toroidal canonical angular momentum and hence of the perfect EP confinement. Tokamaks are constructed using a finite number of toroidal field coils leading to a periodic magnetic field ripple. ITER, for example, has 18 toroidal field coils, leading to a magnetic field ripple of more than 1% in the plasma periphery. Since this field ripple would cause an unacceptable level of EP-induced heat loads on local areas of the first wall, ITER will be equipped with ferritic steel inserts designed to smooth the ripple. Ferritic material in (tritium breeding) TBMs will also cause magnetic perturbations, breaking the toroidal symmetry. As noted previously in section 5.2.4, magnetic field perturbations due to TBMs have been investigated using a TBM simulation coil at DIII-D, and the corresponding EP losses have been experimentally characterized (see [28] section 8.2.2). Finally, as discussed in [28] section 8.2.3 and [24] section 4.1, RMP coils deliberately introduce 3D magnetic perturbations to mitigate ELMs and RWMs. The 3D effects are largest at the plasma periphery and consequently have the largest impact on EPs found there. In high-density plasmas, the high probability of charge-exchange reactions with plasma ions results in a significant fraction of the NBI ions being born in the periphery. EPs can also reach the plasma periphery due to their large orbit widths or due to anomalous transport outwards by instabilities such as sawteeth or AEs.

Losses due to 3D effects can be calculated accurately in MHD quiescent plasmas using orbit following codes such as ASCOT. In ITER, 3D losses are expected to be at acceptable levels in MHD quiescent plasmas by design of the device (see [28] section 8). This is also expected be the case in future burning plasma devices. However, 3D effects can synergistically interact with plasma instabilities and turbulence, a challenging research topic that will need to be resolved to ensure tolerable EP loss levels and to optimize the fusion plasma performance. Equilibrium and stability calculations including 3D effects continue to be key in EP research.

As discussed previously in section 3.2.2, when the α-particles have collisionally slowed down they become helium ash, which must be continuously removed from the plasma to avoid diluting the DT fuel, which could jeopardize the sustained fusion burn and, for ITER, the mission goal for the pulse length. Ideally, the α-particles should have good confinement at high energies and poor confinement at thermal energies, which is a challenging task. Potential candidate mechanisms for this energy-selective α-particle transport include injection of radiofrequency waves, 3D fields, and EP-driven or thermal-plasma-driven MHD instabilities such as sawteeth (see, e.g., [28] section 10).

8.3 Interactions of energetic particles and thermal-plasma-driven MHD instabilities

MHD instabilities appearing in the absence of EPs are referred to as thermal-plasma-driven MHD instabilities, and these can also interact strongly with EPs, as discussed in [28] section 5. EPs can change the mode stability, and the instabilities can in turn cause EP transport, as has been observed for sawteeth, kink modes, NTMs, ELMs, KBMs and RWMs.

NTM stability and growth are strongly affected by EPs, and NTMs are known to cause EP transport and losses on several devices. For example, in the recent JET DT experiments, NTMs caused transport of EPs, leading to losses detected by FILD measurements. EPs can stabilize or destabilize NTMs, depending on whether trapped or passing EPs dominate, on whether the NTM frequencies resonate with an EP subpopulation, on the drift-orbit size and on the magnetic shear. At DIII-D, NTMs caused the NBCD efficiency to drop by as much as 80% and the neutron rate by as much as 65%, suggesting anomalous EP transport. For ITER plasmas, (2,1) and (3,2) NTMs have been simulated with the ASCOT code in a magnetic field configuration which included a realistic toroidal field ripple. NTMs increased loss levels uniformly but did not cause any hot spots on the first wall: the wall heating power densities were found to be well within acceptable levels.

The interaction of sawteeth and EPs is well understood and experimentally corroborated. EPs with intermediate energies behave like thermal particles and are transported by sawteeth. EPs with high energies, however, have wide orbits and can decouple from the evolving flux surfaces, and the transport depends on their orbit types. These predicted phase-space selective transport patterns have been experimentally confirmed recently at numerous tokamaks using velocity-space tomography. Just before sawtooth crashes, an internal $n = 1$ kink mode is often observed. Recent modelling of a saturated internal kink mode and a sawtooth at JET found that the losses from the kink mode and from the sawtooth are comparable. EPs can stabilize the kink mode and prolong the sawtooth period. This typically leads to severe sawtooth crashes nicknamed ‘monster’ or ‘giant’ sawtooth crashes, which can trigger NTMs and lead to reduced plasma confinement. Such ‘monster’ sawteeth are expected in burning plasmas due to the stabilizing effects of the α-particle population. Instabilities that remove fast ions from the core, such as tornado modes (core-localized TAEs), are often the precursor to such ‘monster’ sawtooth crashes, therefore, since they remove some of the stabilizing EPs by expelling them across the $q = 1$ surface. Sawteeth could degrade the performance of burning plasmas by redistributing the

α-particles from the plasma core to the periphery, leading to a loss of core heating. Several methods for pacing sawteeth have been developed, leading to more frequent, but weaker, sawteeth, with an overall positive effect on the plasma performance (see [28] section 5.2). The capability to control sawteeth is important for ITER and other future burning plasma devices, where this is the first line of defence against the triggering of NTMs. Sawteeth are also a candidate mechanism to remove helium ash and other impurities from the core region of burning plasmas.

Ballooning instabilities were subject to detailed studies around the 1990s but have now gained renewed interest. The stability limits against ballooning modes are most likely reached near the plasma periphery, where, under certain circumstances, some small fraction of the EP population may be found, either as a result of the core redistribution processes noted above or due to NBI deposition. The trapped EP precessional drift frequencies may coincide with those of ballooning modes and lead to destabilization, which could degrade the β-limit in the pedestal region. The interaction of EPs and ballooning instabilities should therefore be re-examined using updated ITER scenarios and profiles.

RWMs can become unstable in burning plasmas at plasma pressures above the no-wall β-limit, such as predicted for the ITER steady-state scenario [59]. To make predictions for ITER, it is necessary to understand the interaction between EPs and RWMs in the wall-stabilized regime, a process which has been investigated in numerous experiments and simulations. RWMs could be destabilized by resonating with the precession frequencies of EPs or of thermal ions. The EPs and the thermal particles can stabilize RWMs above the ideal no-wall β-limit. However, if the EP pressure is large, trapped EPs can drive off-axis fishbones unstable, which is an EPM emerging from the RWM.

Recently, it has been observed on several tokamaks that ELMs lead to EP losses, as observed by FILD measurements. Despite losses at typical NBI energies, lost EPs with energies larger than the NBI injection energy were observed, an effect attributed to EP acceleration by the ELM. However, such accelerated EPs have also been observed in the absence of ELMs. Furthermore, it has been found that energy and momentum transfer between the EPs and the ELMs influence various characteristics of the ELMs, for example their growth rate and amplitude.

Finally, an unresolved issue deserving attention in the future is the behaviour of α-particles during disruptions. It is at present unclear by how much the α-particles will increase the wall heating during disruptions. Similarly, the wall heating due to REs that can be generated during the disruption CQ, a subject of ongoing analysis, needs to be characterized quantitatively.

8.4 EP-driven instabilities and energetic particle modes

EP-driven instabilities are observed on any tokamak or stellarator with a significant EP population. The EP velocities exceed several key characteristic velocities related to EP instabilities in burning fusion plasmas, such as the Alfvén velocity, the sound velocity or the diamagnetic drift velocity, allowing EPs to interact with a range of different instabilities. Characteristic frequencies of the EPs are the gyrofrequency, the bounce and transit frequencies and the toroidal precession frequency, which can accordingly resonate with modes in a wide frequency range, as detailed in [28] section 6. Instabilities found at the highest frequencies, near the ion cyclotron frequency, are the GAEs and CAEs. Instabilities at intermediate frequencies, near the Alfvén frequency, are the TAEs, EAEs, and NAEs, as well as RSAEs. Instabilities at the lowest frequencies, near the diamagnetic frequency, are the BAEs, BAAEs, LFAMs, and EGAMs. These EP-driven modes have, in addition, kinetically modified branches, adding to the zoology of EP-driven modes. The high-frequency modes are typically driven by gradients in velocity space and lead to velocity-space transport, whereas the intermediate to low-frequency modes are typically driven by spatial gradients and lead to spatial transport. However, modes with toroidal mode number $n = 0$, such as the EGAMs, are not driven by spatial gradients.

Modes experience strong drive when the frequency of the mode is in resonance with the gyro-, transit, bounce or precession frequencies of the EPs and when the width of the mode is comparable to the width of the EP orbit [60]. In future burning plasma devices, such as ITER, EP-driven instabilities are therefore expected to exhibit higher mode numbers and to be more narrowly localized in the radial direction than in current experiments, which could enhance local transport and radial avalanches [59]. This effect cannot be validated in contemporary devices and requires validation in future, larger-scale experiments. EP-driven modes are typically found in regions where continuum damping is absent or weak (see [28] section 6.1). However, at high EP pressure, the mode drive can become larger than the continuum damping, forming distinct EP mode branches, referred to as EPMs. As noted previously, EPMs often emerge from EP-driven modes in the least damped parts of the continuum.

The strong dependence of EP-driven instabilities on plasma profiles, in combination with the importance of resonance effects, precludes the use of scaling relationships, such as those in the transport and confinement area (see section 3), for predicting the behaviour in reactor-grade plasmas. For example, one challenge in developing a predictive capability for ITER is that the most unstable mode numbers will be larger than those in current tokamaks. The best approach to predicting the evolution of EP-driven instabilities in burning fusion plasmas, therefore, is to validate codes using existing experimental data and then to use these validated codes in

making predictions for burning plasmas. Further validation of such predictions using data from experiments with and without burning plasmas in ITER, as well as in other burning plasma devices under design or construction, will be essential.

Both linear and nonlinear modelling are used in the study of EP-driven instabilities. Linear stability analysis (see [28] section 6) is essential, since weakly damped modes close to marginal stability are excited by EPs in many experiments. The onset of AE modes can often be predicted accurately, and their measured mode structures near to marginal stability are well understood, and are often in remarkable agreement with linear modelling. Damping mechanisms are also well understood, and the relevant predictions nowadays typically match experimental data, including direct measurements obtained with active antennas. Similarly, for the drive terms, the predictions are consistent with the observed stability thresholds. The often-excellent agreement between linear theory and experimental data has been corroborated in several benchmark studies performed by the ITPA TG for EP physics based on a vast database provided by the major tokamak experiments with significant EP populations (see [28] section 6). Additionally, predictions made using different codes agree among each other within an expected margin. The thorough understanding of the mode structure and frequencies which has been developed allows the inference of plasma properties by MHD spectroscopy. For example, the measured evolution of RSAE frequencies directly shows the evolution of the minimum of the q-profile, q_{min}.

Since publication of the PIPB [58], the low-frequency modes, BAAEs, LFAMs, and EGAMs, have been thoroughly investigated to map out their basic properties. It has been found that BAAEs are more stable than LFAMs for most typical experimental conditions, and that measured mode structures match the LFAMs well. Similarly, while GAMs have been known since the late 1960s, EGAMs have been studied only in recent years. EGAMs are standard GAMs excited by EPs. They are thought to be important, since the measured neutron emission rate can drop strongly in the presence of EGAMs, suggesting that the modes radially redistribute EPs. It has been proposed to channel the α-particle energy via GAMs to the fuel ions, but no efficient power transfer mechanism has yet been identified (see [28] section 9).

Nonlinear modelling, detailed in [28] section 7, attempts to capture the nonlinear behaviour of the EP-driven modes, including the dynamic balance between the mode drive and mitigating effects, for example flattened regions in phase space, turbulent cascades, zonal flows and zonal currents. EP-driven instabilities set limits on the achievable profiles and parameter ranges, since the EP transport is enhanced once the profile gradients become larger than a critical gradient, as has been observed on several tokamaks. Nonlinear effects arising from EP-driven instabilities are highly complex and strongly depend on several parameters: the q-profile, mode classification, mode frequency range, and the number of concurrently active modes. While the behaviour of individual EP-driven modes is well understood, nonlinear effects continue to challenge our understanding, particularly when several modes are active at the same time.

Simulating nonlinear EP-driven modes in future burning plasma experiments presents significant challenges: α-particle transport in burning plasmas is expected to be primarily influenced by interactions with various instabilities which lead to larger transport than neoclassically expected levels. Several factors complicate modelling of the fusion reactor regime, among them the large span in relevant time and length scales. Instabilities have time scales on the order of microseconds whereas the α-particle slowing-down time is on the order of seconds. The length scale of the ion Larmor radius is on the order of centimetres, whereas the plasma size is on the order of several meters. Furthermore, no contemporary experiments can access the reactor parameter regime, making model validation impossible prior to the start of DT operation in ITER or in other burning plasma devices.

As discussed in [28] section 7, these challenges have led to much progress in the development of a range of approaches, including global full-f gyrokinetic, fluid-kinetic, gyrofluid, quasilinear, and various semi-analytic methods. Key physics issues are mode-coupling (wave-wave) nonlinearities versus wave-particle nonlinearities, the interaction of α-particles with AEs and microturbulence, global versus local (flux tube) interactions, the simulation of several EP species (such as beam-injected, ICRF-accelerated and α-particles), the roles of zonal flows and zonal currents, perturbative quasilinear versus non-perturbative nonlinear dynamics, collisionless versus collisional regimes, the presence or absence of external sources and sinks, and linear critical gradient profiles versus nonlinearly flattened profiles. Overall, however, there is no single approach encompassing all relevant effects while, at the same time, computationally efficient and robust enough to allow simulation over an α-particle slowing down time, or even a small fraction of it. The reduced models involve varying degrees of simplification and are designed to retain key physics allowing at least some of these challenges to be addressed in new ways, complementing the other models.

Future research needs to focus on the nonlinear evolution of the interaction between EPs and the instabilities, and on their interaction with turbulence, including any synergistic effects between turbulence and EP-driven instabilities. Moreover, 3D effects caused by deviations from toroidal symmetry should be considered in linear and nonlinear modelling and simulation. Nonlinear effects can be addressed by gyrokinetic theory and by simulation, which is developing rapidly with the increasing computer power available.

The behaviour of EPs and of EP-driven instabilities should be considered when optimizing plasma scenarios for burning

plasmas. Several actuators have been identified that can influence the drive and damping of EP-driven instabilities or the EP distribution ([28] section 10). Candidate actuators relying on heating (NBI, ECRH and ICRF heating), on current drive (ECCD, ICCD, and NBCD), or on RMPs have been tested in several experiments, though further research is necessary, in particular, to understand the exact nature of the mechanisms influencing the wave dynamics. Synergistic effects, such as the interaction between AEs and turbulence or the suppression of turbulence by EPs, complicate the analysis. Experiments have shown that EP current drive has significant potential for scenario optimization. However, systematic experimental studies and further development of integrated modelling are required to optimize future burning plasmas. Proving the effectiveness and efficiency of control schemes for use in ITER and other burning plasma devices remains a key ongoing research topic.

EP modelling and simulation currently address carefully designed experiments that can provide the clearest possible picture of EP physics. However, the self-organization of burning plasmas will likely limit the experimental accuracy of modelling predictions, and additional physics will probably need to be considered. Integrated modelling that includes a broader range of physics processes other than EP physics should provide a more holistic understanding of the role of EPs in the context of burning plasmas. Such integrated modelling of an entire discharge from plasma formation to termination, in which EP physics will naturally play a central role, is likely to provide new understanding and new opportunities for optimization strategies.

8.5 Diagnostics of energetic particles and instabilities

The development, in recent years, of several high-resolution diagnostics has provided access to extensive experimental data on EP phase-space distribution functions and on fluctuations associated with modes interacting with EPs (see [28] section 4). For example, the TIP is a high-resolution fluctuation diagnostic which has demonstrated the capability to detect various AEs and low-frequency modes, e.g., NTMs, at DIII-D. The TIP on ITER will be sensitive to AEs, even in the plasma core, while ECE diagnostics, microwave reflectometry, and magnetic pick-up coils will also provide high-resolution measurements of instabilities. A compelling application of such fluctuation diagnostics is the determination of q_{min} through MHD spectroscopy (discussed in [28] section 4.4.6).

Several high-resolution diagnostics for both confined and lost EPs provide detailed measurements of the EP phase-space distribution. ITER will detect EPs by GRS and NES and cameras, CTS, NPAs, FILD and possibly ICE spectroscopy. High-resolution GRS measurements now not only identify peaks in gamma-ray spectra but can even resolve the spectral shapes caused by Doppler broadening, giving experimental access to the EP velocity distribution function. Additionally, GRS detectors can diagnose the RE distribution function since the bremsstrahlung emission generated by the REs is in the same energy range as the gamma-rays from nuclear reactions. Diamond-based NES detectors can deliver high-resolution energy spectra of DT neutrons, yielding a direct characterization of the fusion reactions. New fast digitization techniques have enhanced the temporal and spectral resolution of CTS measurements. These ITER-relevant diagnostics and new diagnostics, such as FIDA and INPAs, have been implemented on several contemporary tokamaks and stellarators, providing high-resolution data on both confined and lost EP populations. In particular, newly installed ICE detectors are expected to provide the experimental basis required to allow development of an improved understanding of the relationship between EPs and ICE.

Multi-view sets of these high-resolution EP diagnostics have been, or are being, installed at leading tokamaks, and this has driven the development of new methods allowing integrated data analysis. Velocity-space tomography and phase-space tomography have allowed the determination of 2D EP velocity distribution functions, and even of 3D EP distribution functions, in constants-of-motion phase space. These methods are likely to improve in the coming years as new EP diagnostics are installed and physics-based *a priori* input, e.g., collisional physics and wave-particle interaction physics, can be exploited in inversion formalisms. Additionally, ML techniques are likely to open new avenues of research and understanding.

CTS and GRS are primary diagnostics for measuring confined α-particles at ITER, while FILD and potentially ICE could be used to diagnose lost α-particles. Understanding ICE is especially important given that the set of EP diagnostics planned for ITER is rather limited due to feasibility issues for such diagnostics in fusion reactors. The GRS and CTS systems at ITER have almost perpendicular views. It is not possible, therefore, to distinguish co- and counter-going EPs at ITER, unless an additional oblique GRS sightline were to be installed. Oblique sightlines are highly recommended for thorough EP diagnostics in burning plasma experiments but are often associated with complex integration issues. GRS can diagnose high-energy α-particles in the MeV-range, and CTS is sensitive down to about 300 keV for EPs with small perpendicular velocities [61]. An analysis based on Be impurity-based GRS and CTS allowed inference of the α-particle velocity distribution function in the MeV-range. Alternative GRS reactions exist that can be used instead of the Be-α reaction for burning plasma experiments that do not use Be in the PFCs.

Diagnostics of EPs and instabilities are crucial to confront modelling and simulation tools with measured data. While neoclassical confinement of EPs in 3D geometry is well understood in MHD quiescent plasmas, much work is still

needed to advance our understanding of the interaction of EPs with thermal and EP-driven instabilities, EPMs, turbulence and synergistic interactions among them.

9. Diagnostics

The transition from current experimental devices to ITER and future reactors will mark a profound shift in operational context, with significant implications for plasma diagnostics. Next-generation fusion devices are distinguished by their long-pulse or steady-state operation, exposure to intense neutron fluxes, and the use of metallic, highly reflective PFCs, all of which present unique diagnostic challenges. In addition, available space for diagnostic access is more limited, and the number of permissible diagnostic systems is reduced due to engineering constraints and activation risks. As a result, diagnostics must become more compact, robust, and radiation-hard, with increasing reliance on indirect measurements and AI-driven inference techniques. Furthermore, while diagnostics in present devices have often focused on scientific exploration, future systems must prioritize reliability and real-time capability to support plasma control and machine protection. These evolving requirements are driving the development of integrated, synthetic, and model-based diagnostic strategies that can function effectively under reactor-relevant conditions.

Since publication of the PIPB Diagnostics review [62], significant advances have emerged in the realm of diagnostics for burning plasmas and these are discussed in detail in [29]. Much of this progress has been fuelled by the diagnostic strategies envisioned for ITER, with several ITER systems successfully passing their Final Design Reviews. Extensive efforts have been dedicated to addressing issues previously identified as problematic in 2007, including radiation effects on diagnostic components, degradation of first mirrors due to sputtering and deposition, *in situ* cleaning of first mirrors, effects of background reflections from shiny metal walls, and calibration of diagnostics.

Moreover, substantial strides have been made in enhancing the potency of sources such as lasers, beams, and microwave emitters, alongside the development of more sensitive detection systems spanning the entire spectral range. Despite their heightened sensitivity, some of the latest detectors can operate in closer proximity to the demanding conditions of the burning plasma environment.

Recognising that forthcoming burning plasmas post-ITER will present even more severe challenges, considerable focus has been directed towards integrated data analysis and the creation of synthetic diagnostics. Additionally, continuous efforts are underway to refine and validate more advanced control algorithms using current experiments as a testing ground. The following discussion provides a succinct overview of the key highlights in diagnostic developments since 2007, while comprehensive details of the advances made, together with relevant references to the original work, can be found in [29] within the special issue. Highlights in the fields concerned with specific diagnostics disciplines are presented first, followed by the more generic areas of progress.

9.1 Microwave diagnostics

Microwave techniques (discussed in [29] section 8.6) have significant advantages in the diagnosis of burning plasmas, owing to their compatibility with the challenging environment. Most notably, their ability to make use of sources and detectors located at a distance from the plasma, beyond nuclear radiation shielding, coupled with the resilience of in-vessel components, such as waveguides, to radiation makes microwave diagnostics essentially indispensable in future fusion devices.

Although electron cyclotron emission provides a dependable diagnostic for measuring electron temperature profiles in numerous contemporary plasmas ([29] section 8.6.3), its application in burning plasmas faces challenges stemming from harmonic overlap and relativistic down-shifted radiation. These constraints limit the diagnosable region within the plasma. Technological advances, including low-loss transmission lines and more sensitive detectors, have significantly enhanced ECE diagnostics. In particular, the integration of SoC technology has facilitated the miniaturization of the diagnostic backend, improving signal-to-noise ratios ([29] section 8.6.9). Additionally, the advent of new sources in the THz region has expanded diagnostic capabilities, enhancing the accuracy of correlation ECE for temperature fluctuations.

Ion collective Thomson scattering ([29] section 8.6.6) has evolved from a demonstration system in its early applications to a routine diagnostic, offering accurate measurement of the fast ion population with high temporal resolution. Typically employing a powerful gyrotron as a source, ion CTS systems yield details on the fast ion population by measurement of scattered radiation from various directions.

Reflectometry ([29] section 8.6.7 and 8.6.8) has also progressed significantly, with reliable systems operational at numerous fusion devices. These systems not only measure electron density profiles and fluctuations at the plasma edge, but also determine plasma rotation speed through Doppler reflectometry. Reflectometry holds the potential to measure plasma shape and position, potentially supplanting magnetic diagnostics in environments even more demanding than that of ITER.

9.2 Laser aided diagnostics

Incoherent Thomson scattering ([29] section 8.5.2-6) is recognized as a standard diagnostic method for measuring electron temperature and density profiles, continually advancing towards heightened spatial and temporal resolution.

In the quest to diagnose the core of hot fusion plasmas like ITER, two-colour Thomson scattering ([29] section 8.5.4) is anticipated to amplify spectral sensitivity, while ongoing exploration into polarimetric Thomson scattering ([29] section 8.5.5) aims to bolster measurement precision within the hot plasma core.

Toroidal interferometry/polarimetry ([29] section 8.5.8) has demonstrated successful testing, leveraging the rapid temporal resolution of interferometry in tandem with the resilience of polarimetry against fringe jumps. Consequently, this technique is poised to become the primary diagnostic tool in ITER for extracting electron density profiles. Another notable advance is the dispersion interferometer, offering a robust density measurement unaffected by mechanical vibrations.

Polarimetry has emerged as a dependable method for gauging the internal plasma magnetic field and magnetic fluctuations, with considerable efforts directed towards mitigating signal contamination from the Cotton-Mouton effect ([29] section 8.5.9-10).

Laser-Induced Fluorescence finds application in measuring parameters of ions and neutrals near the plasma edge, with proposed integration into the divertor system in ITER, alongside divertor Thomson scattering ([29] section 8.5.6-7). Moreover, LIBS and LIDS, although nascent in 2007, have since become prominent techniques for assessing the quality and surface composition of PFCs and are now widely utilized in diagnostics within the field ([29] section 8.1.4).

9.3 Spectroscopy

Advances in passive spectroscopy encompass two primary domains: the refinement of detection systems for heightened sensitivity and accuracy, and a deeper comprehension and modelling of atomic processes within the plasma. These dual avenues of progress continue to contribute to progress across spectroscopic measurement systems. Passive spectroscopy techniques serve as valuable tools for diagnosing various plasma parameters. However, transitioning from C-based to metal walls introduces additional challenges in identifying suitable emission lines amidst a backdrop of numerous background reflections ([29] section 8.4).

Active spectroscopic techniques ([29] section 8.3), including CXRS, BES and MSE, rely on analysis of emission from neutral beams injected into the plasma. Like passive spectroscopy, continuous enhancements in detectors and spectrometers characterize this field. Moreover, significant efforts are directed towards improving the quality of the neutral beams, focusing on factors such as power, divergence, and energy spread. As a result, diagnostic capabilities are expanding, with CXRS now employed to analyse fast ions, including confined and escaping α-particles, as well as slowing-down beam ions.

9.4 Fusion product diagnostics

Fusion product diagnostics ([29] section 8.2) have emerged as pivotal tools for future fusion endeavours such as ITER and its successors, offering direct insights into the intricate dynamics of fusion processes. Recent advances in this field have yielded innovative approaches, many of which underwent testing during the JET DTE2 [22] and DTE3 [63] campaigns. Significant focus has been directed, in particular, towards enhancing neutron and γ-ray diagnostic systems for ITER.

Fast-ion loss detectors ([29] section 8.2.3), deployed across various fusion facilities, have seen advances in both their hardware and modelling capabilities. A critical consideration relates to the accurate prediction of the trajectories of fast ions exiting the plasma and determining optimal distribution of detector locations accordingly.

A paramount objective involves devising an *in situ* calibration strategy for ITER's neutron diagnostics, employing a radioactive source within the vessel ([29] section 8.2.9). Minimizing intervention time is crucial to ensure minimal disruption to future experimental campaigns.

9.5 Measurement of power fluxes and retained fuel

Monitoring the first wall through Infrared techniques is critically important in the burning plasma environment for the detection of hot spots on the PFCs and ensuring their protection against over-heating. However, in fusion devices with metallic PFCs, this task is challenging, necessitating the differentiation of genuine hot spots from mere wall reflections. To address this complexity, advanced analysis techniques, including those rooted in AI, have been developed in some detail and trialled on fusion devices featuring metallic PFCs ([29] section 8.1.5).

In the domain of bolometry, considerable efforts have been focused on engineering radiation-hard bolometers capable of withstanding the rigorous environmental conditions within ITER, given their close proximity to the plasma. While the primary system for ITER relies on resistive bolometers, extensive research has also been dedicated to enhancing imaging bolometers to better adapt to the demands of the burning plasma environment ([29] section 8.1.2).

9.6 Environmental effects

Burning plasmas create an exceptionally challenging environment for diagnostic components situated near the plasma. Many diagnostic techniques, including laser diagnostics, and active and passive spectroscopy, rely heavily on optical-quality first mirrors, which are susceptible to various forms of plasma-induced degradation, such as erosion, deposition by escaping ions, nuclear swelling and related deformations, as well as transmutation. Consequently, substantial efforts are being dedicated to developing robust

mirror solutions tailored to the range of diagnostics foreseen in ITER and to the associated wavelength ranges ([29] section 8.8).

To ensure sustained mirror quality over extended operational periods, a range of *in situ* mirror cleaning techniques have been devised for ITER. Moreover, *in situ* calibration techniques are employed to assess mirror characteristics between plasma discharges, facilitating corrective measures to compensate for mirror deterioration.

Significant emphasis is also being placed on developing radiation-hard sensors capable of withstanding ITER's harsh nuclear environment without succumbing to radiation-induced defects ([29] section 8.7). Strategies include the utilization of mineral-insulated cables and the integration of radiation-hard Hall probes to complement magnetic diagnostics. In general, diagnostic system layouts are meticulously designed to position radiation-sensitive components as far from the plasma as is feasible, mitigating their exposure to detrimental radiation effects.

9.7 Integrated data analysis, synthetic diagnostics and plasma control

Traditionally, data from individual diagnostic systems have been analyzed in isolation from those of other diagnostics, occasionally leading to discrepancies in measured parameters across the different measurement techniques. Increasingly, significant emphasis is being given to developing an integrated approach to data analysis and validation, whereby data from diverse diagnostics are amalgamated to produce a more accurate set of measured parameters ([29] section 8.10). This integrated approach is particularly crucial for future fusion devices, where the number of available diagnostics may be limited. For instance, in the realm of energetic particles, a range of diagnostic techniques are combined using tomographic methods to ascertain the fast particle population.

In a fusion reactor, the fundamental control groups can be categorized based on their interconnections via physics, where a control action in one domain influences others, and via shared actuators ([29] section 8.9). Consequently, a supervisory control system must ascertain the priority of various control tasks, their interconnections, and interfaces with safety and interlock systems. The systematic development of different controllers, considering these interconnections, necessitates a model-based approach involving a ‘plant simulator’ – a digital twin of the actual fusion plant. This simulator should encompass synthetic diagnostics emulating real plant diagnostics. Control signals are then fed to both the real plant and the simulator, and any disparities between the limited data from the plant and the simulator are utilized to refine the simulator and adapt the control strategies.

10. Implications of ITPA research for fusion energy development

While the various communities across the international fusion energy program are following roadmaps towards the generation of ‘fusion electricity’ with varying timescales, a common element of all programs is the operation of a burning plasma experiment which allows the key physics processes and operational issues of plasmas with significant α-heating to be studied and optimized in advance of the step to a device capable of electricity generation. For the major actors, the ITER project is expected to fulfil this role, though other burning plasma experiments may be operating on a similar timescale if current developments prove successful. The approach to electricity production through an intermediate burning plasma device reflects an appreciation of the additional complexity (with the potential for nonlinear behaviour) in operation and control of a plasma with predominant α-heating: confirmation in the burning plasma regime of current expectations for fusion gain, plasma transport and stability, plasma edge and divertor behaviour, the interaction between the α-particle population and the thermal plasma, etc. will provide a deep understanding of plasma behaviour in the burning plasma regime and will be fundamental to establishing the scientific basis (and certain elements of the technology basis) for the subsequent development of electricity-generating fusion power plants. Given the experimental capabilities of ITER, it is likely to provide the most comprehensive insights into burning plasma physics in long-pulse operation (while also addressing many aspects of the technical challenges) at a scale and level of integration approaching those appropriate to a reactor-class device, but other burning plasma experiments under development can also make significant contributions.

The wide-ranging fusion physics R&D studies carried out during the period reviewed in this special issue have been motivated to a significant extent by the need to resolve ITER operational issues. However, the results of these studies have wider implications: they provide support to the design of tokamak burning plasma experiments generally and they inform the preparation for the operation of burning plasma devices through improvements in the understanding of key physics processes, strengthening of the predictive capability towards plasmas at the scale required for burning plasma operation, and development of the ‘tools’ required to implement robust and reliable long-pulse operation of ‘generic’ burning plasma experiments. The ITPA’s research program, carried out in close collaboration with the major facilities and research groups in the international fusion program, is therefore designed to advance the realization of fusion energy by supporting the rapid and effective operation of the forthcoming generation of burning plasma devices.

Plasma scenarios and control. A fundamental issue in the operation of burning plasma devices is the choice of candidate plasma scenario(s) for the operational program. Some subset of available scenarios generally provides an initial design basis for a specific device, However, further modes of operation established through the ongoing fusion plasma research can offer the potential for higher DT fusion performance and/or more robust operation. An important aspect of the research program, therefore, is to characterize the access conditions for such modes, to understand, as far as possible, the physics determining the observed behaviour and the extrapolability to burning plasma conditions, and to evaluate the flexibility of devices such as ITER to accommodate new modes of operation within an established design.

While the ITER design basis scenarios have provided the focus for the ITPA's research, results reported in this special issue range more widely, addressing various aspects of several alternative plasma scenarios with the potential to provide the basis for high fusion gain operation in ITER and other burning plasma devices. Much of the discussion presented in subsequent papers revolves around the ELMing H-mode, hybrid mode (within the ITER design basis, essentially a long-pulse variant of the conventional ELMing H-mode with a significant non-inductive component of plasma current) and fully non-inductive steady-state operation. Nevertheless, the 'improved H-mode', which can be considered a more refined version of the hybrid mode, the QH-mode, having the advantage of a 'quiescent' plasma edge without ELMs, and the I-mode, a plasma regime with neither a significant pedestal in the edge electron density nor ELMs, exhibit various characteristics which, if extrapolable to the burning plasma regime, could provide attractive options as baseline operating scenarios. Relevant aspects of these operational modes are therefore discussed in subsequent papers. It must be recognized, however, that the experimental database and physics understanding of these alternative scenarios are necessarily more limited than are available for, e.g., the ITER design basis scenarios, and substantial R&D will be required in the years ahead to establish a more robust operational basis for their application to the first generation of burning plasma experiments.

Realization of candidate integrated scenarios for burning plasma experiments has been supported by several lines of research over the timeframe reviewed by the special issue (see, in particular, [27]). Substantial experimental studies of numerous practical aspects of scenarios, including plasma initiation, current ramp-up and access to high confinement regimes, termination of high confinement and current ramp-down, have allowed potential solutions suitable for burning plasma experiments to be developed and have strengthened the experimental database for comparison with scenario simulation tools, which have been under continuous development. While these numerical codes are not yet capable of a fully integrated, first-principles based simulation of plasma scenarios from the plasma core to the divertor target, considerable progress has been made to expand the range of physics that can be handled in an 'integrated' simulation. The code validation activities enabled by this research are of considerable significance, both in refining operational planning for the first generation of burning plasma experiments and in advancing the physics basis for the conceptual design of the DEMO- and FPP-class devices intended to follow this first generation.

Sustainment, in long-pulse or steady-state plasmas, of any of the candidate plasma scenarios for burning plasma operation will rely on a robust capability for integrated plasma control. While the measurement capability available to the initial generation of burning plasma devices might not differ substantially from that available in existing devices, it appears inevitable that DEMO- and FPP-class devices, and future commercial fusion power plants, will have a more restricted complement of measurement systems. Significant constraints will likely flow from overall design requirements in reactor-class devices. In addition, the 'hardness' of diagnostic systems, in particular, of those elements located in the plasma proximity, against the demanding burning plasma environment will also influence the selection and performance of measurement systems.

The range of available control actuators is also likely to be more limited, while the demands for real-time control of plasma parameters (e.g., plasma equilibrium, density, fuel mixture, current profile, heat and particle exhaust, fusion power/gain, as well as a range of MHD instabilities) is likely to grow to ensure maintenance of a stable burn point. Implementation of robust control algorithms will also be complicated by the decline in effectiveness of auxiliary heating systems in the presence of significant α-heating. The progress on model-based control strategies [27] is therefore an important development, but much further research on the application of this approach to existing tokamaks will be necessary to prepare for its successful exploitation in burning plasma operation. A key aspect of this research will be to determine effective algorithms for optimum sharing of the limited control actuators available to burning plasma devices. For the longer term, the likely limitations on the range of both measurement and actuator capability in fusion reactors imply that an important contribution will be required from the first generation of burning plasma devices on the implementation of robust control under such constraints. Future model-based strategies might then incorporate ML/AI-derived control algorithms derived from databases assembled from the first generation of burning plasma experiments. It can be expected therefore that the current R&D activities on the use of ML techniques, for example in disruption prediction, will expand

substantially in the future and encompass a wider range of control applications.

Diagnostics and data integration. In the context of optimizing burning plasma operational scenarios and establishing the required control functionality, an important contribution has come from the progress reported in the coordinated R&D activities addressing key issues for diagnostic operation in the burning plasma environment [29], where control, protection and physics applications will make substantial demands on the available measurement capability. Continued progress in the recently initiated activity on integrated data analysis can be expected to support the development of operational scenarios through more robust methods for deriving plasma parameters from diagnostic measurements. This methodology could ultimately be of value in real-time control applications in burning plasma devices with more limited access for diagnostics. Indeed, the development of 'synthetic diagnostics' is already an integral aspect in the development of model-based control algorithms.

While the specific diagnostic systems on which current devices rely to support reliable operation and physics studies are confronted with a variety of challenges in their implementation within the burning plasma environment, several generic issues must be addressed to establish the technical basis for a wide range of diagnostics. An obvious case is the selection of (mostly ceramic) materials which can survive in the burning plasma neutron and γ-radiation environment. The principal effects of concern relate to accumulated radiation dose, but in-vessel magnetic coils are also subject to radiation-induced effects which can be of significance during burning plasma operation, causing spurious signals to be generated in a critical measurement system for equilibrium control. Instruments such as bolometers and Langmuir probes, with a significant ceramic content and exposed directly to the neutron flux (as well as γ-rays from nuclear reactions in the tokamak structure) are perhaps most at risk, but signal feedthroughs and diagnostic windows also suffer radiation damage despite the possibility of locating them behind substantial shielding. A long-term program involving the IO, the ITER Members Domestic Agencies and the ITPA Diagnostics TG has been implementing R&D to understand the possible radiation-induced effects and to develop viable solutions for ITER (and similar burning plasma devices) [29]. Satisfactory solutions to the principal challenges have been developed for ITER, providing a basis for longer term R&D towards the identification of solutions for later phases of fusion energy development, where such materials could be subject to radiation fluences which are 1 – 2 orders of magnitude higher than in ITER-class devices.

Similarly, active and passive diagnostic systems making measurements across much of the em spectrum (certainly from VUV to mm-wave wavelengths) require mirrors for transmission and/or collection of em radiation. The so-called 'first mirrors', which must view the plasma directly to fulfil this function, are subject to a hostile environment with significant particle fluxes, including energetic charge-exchange neutrals, thermal loads, particularly those from neutron fluxes and intense radiation loads produced by thermal transients, em loads during disruptions/VDEs and particle bombardment during glow discharge cleaning, which will likely be required for wall conditioning. Extensive R&D studies have been undertaken, coordinated by an ITPA working group in close collaboration with major ITER stakeholders, to study the possible impact of plasmas, to develop manufacturing solutions to reduce the plasma effects, and to investigate a range of protection, mitigation and cleaning techniques. The progress in this R&D has made invaluable contributions to the design of ITER diagnostic systems and will also inform the thinking on viable solutions for subsequent burning plasma devices, which can be expected to draw extensively on ITER's operational experience.

In addition to pursuing practical solutions to these generic challenges for the design and construction of ITER measurement systems, the ITPA Diagnostics R&D program has been active through a series of working groups in the study of the specific challenges which must be resolved to ensure that the required range of diagnostics is available in ITER and can meet the demanding performance requirements to support ITER operation and the goals of the IRP. The R&D activities have encompassed both active and passive spectroscopy measurement techniques, laser-aided and microwave diagnostics and fusion product diagnostics, of particular significance, of course, for the analysis of plasma performance in burning plasma experiments. While, in the main, addressing specifics of the implementation of these diagnostics in ITER, the results of this R&D are of wider importance for the development of diagnostic systems for future burning plasma devices.

Transport and confinement. As highlighted in the discussion in section 3, the strong dependence of fusion power production on the confinement enhancement factor, H, motivates the design of burning plasma devices around plasma scenarios with 'enhanced' (relative to L-mode) energy confinement to minimize the device scale required to achieve a specified fusion gain. To strengthen confidence in the fusion performance predictions for burning plasma experiments, a substantial effort has been implemented over several decades to improve the predictive capability for access conditions to enhanced confinement regimes at the burning plasma scale (e.g., the H-mode power threshold), to refine the predictions of global confinement time in relevant regimes (the type-I ELMing H-mode has been subject to the most detailed studies and therefore has the most extensive database), and (ultimately most importantly) to provide a more quantitative understanding of the physics processes determining heat,

particle and momentum transport in these plasma regimes [23]. Moreover, the progress made in understanding of the physics processes within the narrow plasma edge pedestal [24], together with the recognition of the critical role played by the pedestal in regulating core heat transport in several enhanced confinement regimes, has provided a continuing stimulus to develop an integrated first-principles based description of transport between the plasma centre and the separatrix.

Perhaps the most significant progress in this area since the publication of the PIPB is evident in the deeper understanding of the principal physics processes determining heat transport and the improvements in first-principles based modelling, both at the level of detailed gyrokinetic modelling and in the development of reduced models, allowing more ambitious integrated models of heat transport across the plasma. These advances rely not only on continuing evolution of theoretical analysis, but also on improvements in numerical techniques and on the exploitation of expanded HPC capabilities. The current simulation and predictive capabilities are still far from providing the ability to design a burning plasma experiment, *ab initio*, around a plasma scenario derived from such models, but the progress made guides the ongoing research towards more focused studies and more quantitative analysis of current experiments. Nevertheless, the (long-standing) recognition of the interdependence of heat, particle and momentum transport processes, the growing appreciation of the role of energetic particle populations in heat transport processes (e.g., the potential for turbulence stabilization), and the interaction with MHD stability boundaries, particularly in the plasma pedestal, emphasize the significant challenges that must be resolved in working towards a comprehensive first-principles based description of heat transport (and equally, as discussed below, towards first-principles based descriptions of particle and momentum transport).

In the light of these outstanding issues, which inevitably influence the conditions for accessing regimes of enhanced confinement, research has continued to refine, where possible, the empirical scaling for such access to provide a practical approach to modelling of plasma scenarios in the burning plasma environment and to support ongoing design studies of reactor-class devices. Although the experimental results discussed in [23] emphasize that numerous factors which do not appear in the reference scaling for the H-mode power threshold (see Eq. (2)) influence the heating power required to access the H-mode regime and that uncertainties in the extrapolation to burning plasma parameters are known to be large, it has not proved possible to date either to improve on the reference scaling or to provide a quantitative physics-based description of the requirements for H-mode access. In addition, although there has been continued progress in the experimental analysis of the plasma evolution through the L-H transition [24], there is still no satisfactory theoretical description of this nonlinear bifurcation which yields reliable quantitative predictions for the transition in existing and future devices. These sources of uncertainty, therefore, remain a significant limitation in simulating end-to-end burning plasma scenarios based on the standard H-mode and, by extension, in the simulation of burning plasma scenarios based on other enhanced confinement regimes. Future research to reduce these uncertainties will be important in reducing the operational risk for burning plasma experiments, and the validation of modelling predictions in the first generation of burning plasma devices will be essential for strengthening the design basis for subsequent generations of reactor-class devices.

The substantial work to expand the experimental database for H-mode energy confinement and to improve on the long-standing reference scaling, IPB98(y,2), has incorporated data from areas of operating space more relevant to burning plasma operation and, significantly, has added data from experiments with all-metal PFCs in AUG and JET, an aspect of particular relevance to burning plasmas. Two initial conclusions of importance for extrapolation of confinement times to ITER and later devices are that the most relevant form of the empirical scaling derived to date has a reduced dependence on major radius (though a somewhat stronger dependence on plasma current), while the 'dimensionless' form of the scaling based on physics variables [33] has a 'Bohm-like' dependence on the normalized ion Larmor radius, ρ_i/a, rather than the 'gyro-Bohm' dependence favoured by theory and by previous analyses. These results tend to reduce the confinement time predictions for tokamaks at the larger scales characteristic of ITER and DEMO-class devices, and therefore motivate continuing studies to understand such discrepancies and to establish a more robust design basis for predicting performance in future burning plasma experiments. It will be necessary not simply to expand the experimental database derived from experiments with all-metal PFCs, but also to address additional influences on core confinement, such as (partially-) detached divertor operation, ELM control with RMPs or pellet-injection (and plasma regimes exhibiting 'natural' ELM-free behaviour [24]), plasmas with $T_e \geq T_i$, etc. Since it is common practice to normalize energy confinement in other enhanced confinement scenarios against the standard H-mode confinement scaling, this issue has implications across the range of candidate operation scenarios for burning plasma experiments.

Notable progress has been made in characterizing particle transport, particularly in relation to strengthening the observed trend that peaking of the density profile increases with decreasing effective collisionality, ν_{eff}, which is favourable for burning plasma operation. Nevertheless, there is a wider range of issues within the area of particle transport, fuelling and exhaust which can have a significant influence on the sustainment of high fusion gain operation and where the

current uncertainties in fuelling efficiency, impurity (including helium) transport and particle exhaust efficiency require more extensive studies in the future. This remains true also for momentum transport, where there has been substantial progress in the theoretical description in recent decades, but where significant quantitative differences with experimental observations persist and there is no reliable basis for the prediction of plasma rotation under the conditions relevant to burning plasma experiments, with low external torque drive from auxiliary heating systems.

Pedestal and ELM physics. A critical area of 'core' physics which has a profound effect on plasma performance relates to understanding and quantifying the various processes which determine the behaviour of the edge pedestal, the narrow region inside the separatrix which plays an influential role in the development of several plasma regimes of interest to burning plasma experiments [24]. The importance of the physics processes in this region is associated not simply with its 'interface' role between the physics processes determining the evolution of the confined plasma and those occurring in the plasma external to the separatrix which moderate the interaction between the plasma and the PFCs, but also stems from the long-standing recognition that the pedestal behaviour has a strong influence on the overall confinement quality of the plasma. The impressive progress made in diagnostic systems, allowing measurements of numerous parameters in this layer with high temporal and spatial resolution, has complemented the progress made in the theoretical and modelling analysis of transport and stability processes in the pedestal. The improvements in measurement capability have allowed more detailed validation of analyses emerging from advanced computational models developed over the past ~20 years, leading to improved predictions for pedestal parameters and for conditions at the onset of ELMs. Nevertheless, given the sensitivity of such predictions to magnetic shear in the pedestal, the limitations in the diagnostic resolution of current profiles in this region remains a key issue for the further refinement of the predictive capability.

While the physics understanding of the local plasma conditions determining the onset of the ELM crash has advanced significantly during the period reviewed here, perhaps the more significant development for the reliable operation of burning plasma experiments is the progress made in the development and widespread application of techniques for the control of ELMs, motivated by the recognition of the implications of transient heat pulses produced by ELM crashes at the burning plasma scale for the lifetime and integrity of divertor PFCs. ELM suppression by the application of RMPs has expanded rapidly since the initial promising results in DIII-D and has now been demonstrated as a reliable technique using a range of applied perturbation spectra in several tokamaks. This experimental progress has been accompanied by substantial theory and modelling advances in describing the nonlinear physics related to the penetration of the magnetic perturbations and in providing an understanding of the access conditions for ELM control in terms of the applied field strength and (toroidal and poloidal) magnetic spectrum. The overall modelling capability yields predictions of access conditions for ELM control (e.g., accessible window in q_{95}) which are in quantitative agreement with experimental observations in present devices. Significant progress has also been achieved in the application of alternative ELM mitigation techniques via pellet injection (pellet ELM pacing) or repeated vertical plasma displacements (vertical kicks), both of which reduce the impact of ELMs on PFCs by increasing the ELM frequency above the 'natural' frequency which would otherwise prevail. The wider application of these methods has also been accompanied by the development of a deeper understanding of the physics processes determining the observed behaviour. This research has therefore expanded the range of ELM suppression/mitigation tools available and provided guidance on the principal physics processes at work, giving increased confidence in the predictive capability towards future burning plasma applications. Nevertheless, the ongoing R&D activities will have to demonstrate that use of one or other of these techniques is consistent with other requirements of burning plasma scenarios, such as achieving the very large increase in triggered ELM frequency normalized to the natural ELM frequency (in the case of techniques exploiting pellet injection or plasma position excursions), operation with low edge collisionality, core fuelling by pellet injection, stable radiative divertor operation, etc.

As noted previously, several regimes with H-mode like enhanced confinement but without ELMs have been observed in tokamak experiments (e.g., QH-mode, EDA H-mode, I-mode). The potential advantages of burning plasma operation under such conditions has motivated significant experimental and theoretical studies over the past ~20 years to expand the parameter range over which these regimes can be sustained, as well as to deepen the understanding of the physics determining access to the regimes, particularly the processes suppressing the appearance of ELMs. These studies have also benefited from the improvements in diagnostic capability, theory and modelling which has supported the significant progress in pedestal studies reviewed in [24]. Access conditions for these regimes, though not quantified as extensively as the ELMing H-mode power threshold scaling, are sufficiently well understood to allow reliable production of the regimes in several existing tokamaks, to a significant extent independently of the nature of the PFCs, an important result for their exploitation in future burning plasma devices. In addition, progress has been made in understanding the role of pedestal-localized MHD and turbulent instabilities in modifying the pedestal pressure to avoid the onset of ELMs. The advances made, both in expanding the operational access

space for these regimes and in understanding the underlying physics, offer considerable promise for the future. However, the experimental database remains limited: e.g., in relation to the characterization of the dependences of access conditions and energy confinement on plasma parameters and concerning the compatibility of these regimes with the other aspects of burning plasma operation noted above in considering the viability of ELM control techniques in the burning plasma environment. These limitations emphasize areas in which continuing R&D will be required to confirm the suitability of these regimes as candidate scenarios for burning plasma experiments and reactor-class devices.

Disruptions and MHD stability. ELMs are, of course, but one of a range of MHD instabilities which can impact both the fusion performance of the burning plasma and, potentially more seriously, the interactions between the plasma and the surrounding PFCs and confining structures. The most significant of these are the major disruption and the VDE, which are distinct manifestations of MHD instability, but which often occur sequentially in rapid plasma terminations. Since several other forms of MHD instability (e.g., NTMs, RWMs, EFs) are often initiating instabilities for major disruptions, the overall thrust of research on MHD phenomena in recent decades has aimed (as in the case of ELMs) both at improving the quantitative understanding and characterization of MHD instabilities and at developing reliable control methodologies and techniques to avoid instability boundaries, to suppress instabilities should they appear and to mitigate their worst effects should they grow uncontrollably [25].

While ITER is structurally designed to withstand a significant number of disruptions/VDEs during its operational lifetime, it is recognized that the severity of potential thermal, em and RE loads at the power reactor scale implies that essentially no full-power disruptions/VDEs can be tolerated. Nevertheless, all burning plasma devices must be able to demonstrate structural integrity against the worst expected disruption/VDE, as well as some degree of PFC resilience to the associated thermal and RE loads on the divertor and first wall. Developing an improved quantitative characterization of the loads on plasma-facing and confining structures generated by disruptions/VDEs has therefore been a significant element of the continuing R&D in this area. Combined with the increasing sophistication of MHD and plasma edge simulation codes, the expanded and improved experimental database has allowed the specification of an improved design basis for burning plasma devices. For example, advances in the simulation capability for so-called asymmetric VDEs, in which the instability is modelled as a vertically displaced plasma undergoing a kink instability, suggest that the associated radial forces on the tokamak structures will be less than previously predicted.

Such developments have not, however, eliminated the need for dealing with the worst potential consequences of disruptions/VDEs at the burning plasma scale (and subsequently in reactor-class devices). As discussed in sections 5.1.3 to 5.1.5, and motivated to a significant extent by the specification of ITER's requirements, this need is being addressed by the establishment of a three-pronged strategy involving: (i) the real-time recognition and avoidance of operational boundaries which trigger disruption-generating instabilities, (ii) active control techniques for the suppression of such instabilities should they arise, and (iii) validation of mitigation methods to limit the severity of disruption/VDE thermal, em and RE loads. Extensive international collaborations have focused on these challenges over recent decades and, although the community does not yet have a comprehensive demonstration that all the required tools are in hand for failure-proof (or nearly so) implementation in burning plasma experiments, very substantial progress has been made in all areas.

Identifying the trends in plasma evolution which inevitably result in instability triggering and disruption has been a subject of experimental research since tokamaks emerged as a promising magnetic configuration for the realization of fusion energy. While physics understanding has grown during this period, the resultant predictive capability has fallen short of that expected to be required in burning plasma devices. The increasing application of AI/ML-based methodologies to the analysis of experimental data, which has expanded significantly since publication of the PIPB, has contributed to a significant improvement in the identification of imminent disruptions and has therefore provided a more reliable basis for the implementation of disruption avoidance, either by steering of the plasma scenario away from instability boundaries, or by the triggering of active control of instabilities while they are still at low amplitude. An important aspect of the ongoing AI/ML-based studies will be to demonstrate that experimental databases used for training of the necessary algorithms in current devices, perhaps enhanced with experimental data obtained in low performance plasmas in a specific burning plasma device, will provide a reliable disruption prediction/detection capability across the full operating space of the burning plasma device.

The disruption avoidance component of the overall strategy can broadly be subdivided into two strands: (i) real-time state recognition and active steering of the plasma state into a more stable area of the operating space; (ii) triggering of active control techniques for suppression of instabilities which have been detected (in some cases, simultaneous implementation of both strands may be necessary). As discussed previously, research on integrated control methodologies, including model-based control, has expanded significantly in recent decades, though further R&D in this area will be a necessary prelude to their application to burning plasma experiments. The second strand, related to the development and application of reliable techniques for active control of MHD instabilities,

has become a major area of research, closely linked to studies on the development of a more quantitative understanding of the physics determining the growth of MHD instabilities (including kinetic effects, which will be of greater significance in burning plasmas). While techniques for the control of sawteeth, NTMs and RWMs have all benefited from this research, the very significant progress in the theoretical treatment of how the plasma responds to magnetic error fields has been particularly striking and has had a significant impact on the approach to EFC using the dedicated coils with which many devices, including ITER, are now equipped. The advances in theoretical understanding of the 3D plasma response to error fields has confirmed the need for very tight tolerances on magnetic field symmetry in device construction, an approach which is being implemented, for example, in ITER assembly. They have also informed the emergence of the concept of 'dynamic' EFC, which foresees active variation of the correction field to accommodate the plasma response and optimize the correction field in real-time.

It was already appreciated during the ITER design studies that disruption/VDE mitigation was an essential technology to support ITER operation, aimed at avoiding or minimizing downtime due to the impact of disruptions at high plasma performance, in particular at high plasma current and high fusion power. Nevertheless, the launching of ITER construction and a more detailed analysis of the potential impact of disruptions on the tokamak structures, as well as further consideration of the suitability of the 'killer pellet' and MGI techniques then available, stimulated a focused effort to improve ITER's disruption mitigation capability and launched extensive international collaborations, both to improve the available technology and to optimize the methodology for its application in ITER. Though primarily centred around ITER's needs, the results of this R&D have much wider implications for burning plasma devices and for fusion energy development. The realization of SPI technology and its application in several devices, for pre-disruptive plasma termination, radiative dissipation of the stored plasma energy, control of the plasma current decay time and suppression/mitigation of REs, has had a profound impact on the preparation of the ITER disruption mitigation strategy and has provided a significantly expanded database for the validation of numerical codes for the modelling of disruption and disruption mitigation phenomena. Further integrated testing of the strategy in existing devices will be important for the preparation of reliable ITER operation, while optimization of the ITER implementation through the ITER research program on burning plasmas will be essential to inform the specifications of a fully (essentially 100%) reliable disruption avoidance/mitigation strategy for DEMO- and FPP-class devices.

Divertor, SOL and PWI. It has long been recognized that resolving the challenges associated with managing PWI in the burning plasma regime is critical to the feasibility of realizing fusion energy as a commercial power source for the future. The scope of the challenges to be addressed is substantial: local power loads must be limited to $\leq$10 MWm^{-2} in stationary operation due to thermo-mechanical constraints on solid PFCs, though a limited number of 'slow' transients of up to ~20 MWm^{-2} may be acceptable, while exhausting up to several hundred MW of α-power from the core plasma; simultaneously, adequate particle exhaust must be maintained to extract unburned fuel, helium ash and impurities from the divertor plasma; as discussed above, transient power loads produced by, e.g., ELMs or disruptions, must be handled in real-time or (preferably) eliminated; the SOL and divertor plasma temperature must be limited to well below 10 eV to avoid excessive sputtering of impurities and satisfactory 'screening' of the core plasma must be achieved to limit the impact of residual impurity production on fusion power; while satisfying these constraints to sustain high fusion power in stable long-pulse or steady-state operation, the overall operational regime must ensure that the (first wall and divertor) PFC erosion rate is sufficiently small to achieve an adequate lifetime (e.g., beyond that determined by radiation damage to blanket and divertor structures in the burning plasma environment), that PFC erosion does not generate excessive quantities of dust, which has safety implications, and that tritium retention in PFC/in-vessel structures remains within regulatory levels, which is also a safety-related issue. As a final challenge, the 'boundary conditions' associated with a comprehensive solution to these issues must integrate seamlessly with the 'boundary conditions' of a core plasma scenario capable of achieving the specified fusion power output.

This suite of issues has been at the forefront of the R&D activities pursued in the divertor and SOL physics area for several decades, and has assumed greater urgency since publication of the PIPB, as the essentially simultaneous launching of ITER construction in 2007 emphasized the need to establish feasible solutions for implementation in burning plasma operation in ITER and other next-step devices. As discussed in [26], significant progress can be shown both in experimental results and in the capability to model the complex SOL/divertor/PFC interaction, which is essential in extrapolating candidate operating regimes to the burning plasma scale. A major contribution to the experimental basis for the required R&D studies has been made by those facilities which have transitioned from C-based PFCs to all-metal PFCs (AUG, JET-ILW, EAST, WEST, KSTAR), as discussed previously in section 6.1. An immediate benefit evident following these transitions is that analysis of in-vessel hydrogenic retention in AUG and JET-ILW revealed a reduction by a factor of ~10 relative to earlier measurements using C-based PFCs. While relieving pressure on operational planning (to limit in-vessel tritium retention) for ITER, this

does not eliminate the need to develop efficient tritium removal techniques for application in ITER and subsequent burning plasma devices.

Although, in the longer term, some form of advanced divertor configuration might mitigate the challenge of stationary power handling to some extent, as mentioned in section 6.3, there seems no alternative to the exploitation of a radiative, or dissipative, divertor regime, in which a substantial fraction of the exhaust power in the SOL is dissipated by impurity radiation before reaching the divertor targets and a (partially-) detached divertor limits the divertor plasma temperature in front of the targets to an acceptable level to avoid excessive sputtering. Experiments in several devices have confirmed that such a regime can be established in devices with all-metal PFCs using 'seeding' by light impurities (N_2, Ne, Ar or mixtures) to enhance impurity radiation and reduce the divertor heat flux to the required value while maintaining a high confinement core plasma. Indeed, novel discoveries such as the XPR, discussed in section 6.2.3, may offer promising opportunities for future development in future R&D activities.

A key element in the optimization of such regimes, and a necessary input to 2D and 3D modelling of these regimes in existing and future devices, is the development of an adequate understanding of SOL perpendicular transport (parallel transport is, in essence, well understood), since, when corrected for the magnetic geometry in the divertor, the resultant width of the power-flow channel determines the degree of mitigation required to achieve an acceptable heat flux to the divertor targets. In this respect, the multi-machine scaling of SOL heat flux widths in low-density H-modes developed within an ITPA collaboration (the 'Eich scaling' discussed in [26] section 2.2), which exhibits an inverse scaling of the heat flux width with poloidal magnetic field, is a significant advance. However, the scaling's prediction of a narrow, ≤1 mm, SOL heat flux width in ITER burning plasmas with $Q \geq 10$ has been contradicted by simulations with advanced turbulence numerical codes. The latter predict a transition to a different turbulence regime at the ITER scale for burning plasmas beyond 10 MA, resulting in a SOL heat flux width of several mm. This implies a need for further R&D to explore possibilities for identifying the more reliable approach to predicting this parameter for ITER (and other burning plasma devices) and validating the related assumptions made in simulations of ITER divertor operation. Nevertheless, it should be highlighted that detailed studies of SOL and divertor heat flux behaviour performed through the ITPA collaborations have made a significant contribution to the specification of the detailed geometry of divertor and inner wall PFCs in ITER, in the latter case supporting limiter plasma operation at $I_p \leq 5$ MA.

The period under review has also seen steady progress in numerical simulations of various quasi-stationary processes in the SOL and divertor plasma, benefiting from improved physics understanding and from expansion of the availability of HPC, with extensive validation activities using the continuous improvements in measurement capability which have been implemented during this time. Physics processes in the SOL, including 2D and 3D turbulence, radiative divertor regimes, PWI impact on PFCs, including more detailed analysis of plasma transients, 'material-oriented' simulations of processes such as material migration, PFC material damage and fuel retention are among the areas of PWI studies which have benefited from this progress. As HPC power continues to expand and there is increasing exchange in closely related areas of tokamak and stellarator R&D, 3D simulations of SOL and divertor plasma behaviour can be expected be applied more extensively (as required, e.g., in ELM-suppressed plasmas using RMPs).

To better understand the risks of excessive impurity production and PFC damage associated with transient events such as type-I ELMs, disruptions and VDEs at the burning plasma scale, a significant range of R&D studies has been undertaken in tokamaks, in linear plasma simulators and in high heat flux test stands. Not unexpectedly, these experiments have confirmed that PFCs in burning plasma devices will suffer significant melting due to unmitigated disruptions and VDEs beyond a very modest parameter range (where fusion power production would be negligible), a process exacerbated by substantial em forces. This re-emphasizes the focus on developing an effective disruption mitigation and avoidance strategy, as discussed above. Studies of W PFCs under heat loads corresponding to unmitigated ELMs in ITER have provided data on surface melting and melt motion, while experiments at reduced energy loads have revealed that the threshold for the initiation of surface cracking is below the energy load corresponding to unmitigated ELMs, though further R&D will be required to define the acceptable operational space in ITER and other burning plasma devices quantitatively. These studies have, moreover, highlighted the several mechanisms contributing to the production of dust in the burning plasma environment, an issue which assumes greater significance in the high duty cycle (or steady-state) operational framework relevant to DEMO- and FPP-class devices. The results of this R&D, together with similar studies performed for Be PFCs, have provided invaluable validation data for numerical codes used to simulate material melting and melt motion at the ITER scale, supporting the analysis which has informed the decision to change the first wall PFCs from Be to W in the new ITER Baseline. An additional factor which is yet to be addressed in developing a comprehensive understanding of possible PFC damage under prolonged exposure to burning plasmas in DEMO- and FPP-class devices is how neutron-induced material damage will combine with the effects of plasma exposure and thermal loading, possibly reducing the

acceptable operational space further. This issue will likely only be accessible to experimental R&D once suitable exposed samples are available from 14 MeV neutron sources (e.g., IFMIF-DONES or a VNS).

Energetic particle physics. A central element of burning plasma physics is, of course, the physics processes associated with the α-particle population, which will, as the fusion gain rises beyond 5, exert a determining influence on the plasma temperature profiles via the self-consistent coupling with the DT fusion reaction rate. The α-population can also, *inter alia*, drive a range of 'fast particle instabilities', interact with MHD instabilities of the background plasma and possibly influence the development of the turbulent instabilities which determine heat and particle transport in the thermal plasma. At the time of publication of the IPB [57], numerous aspects of the 'neoclassical' behaviour of EP populations had already been confirmed, predominantly in experiments in which anisotropic EP populations were generated by auxiliary heating systems, and initial evidence of electron heating by α-particles had been obtained in TFTR and JET DT experiments. Detailed observations (and theoretical analyses) of several classes of AEs were also available. In subsequent decades, the processes which are expected to be significant in burning plasmas, i.e., the (simultaneous) nonlinear interactions between EP populations and various classes of instabilities, which can both determine the growth and evolution of the instabilities and influence EP confinement, have moved to the focus of EP studies [28]. This research has benefited from the increasing sophistication of numerical codes, in parallel with increasing HPC availability, and from the expansion in the range and availability of diagnostic systems dedicated to the measurement of confined and lost EPs.

This latter development has been of value across the entire range of EP studies in recent years, supporting, for example, more detailed studies of the interactions between EP populations and 'classic' MHD instabilities of the thermal plasma, such as sawteeth, fishbones, NTMs, RWMs, ballooning modes, etc. The principal concerns relating to such instabilities are that the free energy associated with the EP populations could increase the virulence of instabilities and that interactions between the EPs and the instabilities could degrade the EP confinement, reducing their heating efficiency and/or causing anomalous heating of first wall PFCs. While experiments have confirmed the potential for deleterious effects on both the thermal plasma instabilities and on the EP populations, perhaps the major conclusion is that extrapolation to the burning plasma scale is sufficiently uncertain that the approach to high fusion gain scenarios will need to be carefully prepared to assess, and reduce insofar as possible, such interactions. Nevertheless, it is recognized that sawtooth behaviour in the plasma centre will need to be actively controlled to eliminate undesirable effects, e.g., NTM triggering following periods of sawtooth stabilization by α-particles, while maintaining beneficial effects, such as the elimination of impurities and helium ash from the plasma centre.

The capabilities provided by the continued development in EP diagnostics and numerical codes has been of fundamental importance in the study of EP-driven instabilities. While code simulations of the linear stability and initial growth rates, as well as the structure, of AEs have been confirmed by numerous experimental observations and detailed code comparisons have provided additional confidence in the theoretical understanding of these aspects of AE physics, the nonlinear behaviour, which is critical to the development of a quantitative description of the consequences of the interaction between the AEs and the EP populations, remains challenging. A further complication is the growing range of EP-driven instabilities (most recently, EGAMs) which have been observed during the period since publication of the IPB and PIPB and which might play a determining role in burning plasma performance. Finally, it is appreciated that, at the burning plasma scale, several plasma parameters key to determining the strength and nature of the EP-instability interaction will be significantly different from the values in existing plasma experiments, the unstable mode numbers are likely to be higher than in existing experiments and multiple modes might be excited simultaneously, potentially allowing a cascade of modes and associated EP transport across the plasma cross-section. While recognizing the demanding nature of making reliable quantitative predictions on the role of EP-driven modes in future burning plasma experiments, the continuing progress in the development of a range of approaches to the numerical simulation of the nonlinear behaviour and the possibility of validating code calculations against experimental measurements provided by the expanding diagnostic capability constitute an important contribution to improving the predictive capability, even if the final validation will rely on operation of the first burning plasma devices. It is also possible that the insights derived from such modelling and validation studies could lead to the identification of operational regimes which are more robust against the impact of these instabilities.

It should also be underlined that an area of EP physics that has seen significant progress in recent decades concerns the theory of RE production and interaction with the background plasma [28, 56]. The appreciation of the risks posed to the integrity of in-vessel components by the very high energy (tens of MeV) REs likely to be produced in post-disruptive plasmas at the burning plasma scale [25] has stimulated a redevelopment of the theory of RE generation and its application to the somewhat pathological plasma environment following major disruptions. This has probed, in particular, the potential interactions with plasma instabilities (including those possibly generated by the REs) and em radiation processes (bremsstrahlung, synchrotron) which could impact

RE acceleration and confinement. These theoretical developments have been complemented by significantly improved experimental data on the growth, confinement and decay of REs in the post-disruptive plasma. An important aspect of this research has been the exploration of possible control techniques to limit the growth of post-disruptive REs to reduce the risk of damage to PFCs. Promising concepts for future development have been identified, although further experimental, theoretical and computational studies will be required to confirm the feasibility of such control techniques for burning plasmas such as ITER should mitigation not prove effective.

11. Conclusions

The goal of providing a robust physics basis for the operation of burning plasmas experiments has provided the guiding framework for tokamak fusion research in recent decades. Exploiting international collaborative research, in which the ITPA has played a major role and which has built on expanding experimental, theoretical and computational capabilities within the magnetic fusion community, the understanding of key physics issues for the realization of high gain burning plasmas has been substantially deepened. In addition, significant progress has been achieved in addressing potential limitations to burning plasma performance (e.g., real-time control of MHD instabilities). Viable options have also been identified for major elements of candidate plasma scenarios, including high-performance core plasmas, power and particle exhaust regimes and advanced plasma control techniques. In areas such as real-time plasma control, Diagnostics, H&CD systems and PFCs, physics R&D has also contributed to advancing several aspects of the required technological capabilities for burning plasma facilities. This special issue reviews the progress made in these various areas and discusses the remaining challenges which will continue to be addressed in the future as the magnetic confinement program prepares to exploit ITER and other burning plasma devices currently on the horizon.

Section 2 introduced the distinction generally made in discussing the issues associated with successful operation of a burning plasma experiment between, firstly, those aspects intimately influenced by the presence of the energetic α-particle population (together with any significant energetic particle population generated by additional heating) and, secondly, scale-dependent plasma physics behaviour, which can encompass a significant range of physics issues. Nevertheless, the experimental results discussed in this special issue illustrate that even in current experiments the energetic particle populations influence physics processes such as heat transport (e.g., [23, 28]) and macroscopic MHD stability (e.g., [25, 28]), implying that this traditional distinction may dissolve as tokamaks transition to the burning plasma regime and that the role of the energetic particle populations will need to be explicitly considered across all aspects of plasma behaviour. Moreover, it is now well-recognized that a comprehensive simulation of burning plasmas will require the integration of complex physics processes from the plasma core to the first wall and divertor targets. Such a quantitative integrated description of plasma processes across the full plasma bore, including the regions beyond the separatrix, is unlikely to be available before the first experiments in the burning plasma regime, but the development of as many of the key elements of this capability as is practicable and their validation in the burning plasma environment will constitute a substantial contribution to the future development of fusion energy. Among the many significant results presented in the following papers, this conclusion should provide a key signal for the future development of the R&D program preparing for burning plasma operation.

Indeed, a key overarching theme that emerges from the collective progress reviewed in this special issue is the evolution from empirically-driven solutions and scenario optimization towards the development and application of validated, physics-based models that can robustly project performance and guide scenario design for burning plasma devices. While empirical scalings have been essential tools in the progress achieved in tokamak experiments to date, the inherent complexity of the burning plasma regime demands an integrated, scientific approach rooted in first-principles understanding. The ITPA's contributions across a wide range of physics topics have played a central role in this evolution, providing not only physical insight but also the foundations for predictive capabilities. As the international program prepares for ITER and other burning plasma devices, maintaining and advancing this model-based, science-driven approach will be essential to ensure that proposed solutions are both technically viable and reliably extrapolable to reactor-scale performance: put simply, the necessary technological demonstrations associated with burning plasma operation must be underpinned by the development of a deep scientific understanding of burning plasmas if the fusion program is to achieve its overall goals.

The substantial progress on many areas of tokamak fusion physics reported in the following papers will provide important elements of risk mitigation for the operation of burning plasma experiments, with MHD control, disruption mitigation and avoidance, heat and particle exhaust as obvious examples. These advances should also support a more focused planning for experimental operation at the burning plasma scale. Nevertheless, the novel role played by the α-particle population and the significant expansion in (physics) parameter range associated with burning plasmas imply that the production and detailed study of burning plasmas will be an essential step in providing assurance that a reliable and robust physics basis is available to support the construction and operation of fusion devices with a significant

net electrical power output. It should, perhaps, also be noted that, although the following reviews address challenges in the preparation of the physics basis for the operation of tokamak burning plasma experiments, there are numerous commonalities in tokamak and stellarator physics and therefore numerous areas where the progress achieved in the studies of tokamak plasmas will be beneficial to the planning of future stellarator burning plasma experiments.

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