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Integrated operation scenarios

Chapter 6 of the special issue: on the path to tokamak burning plasma operation

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Received 1 November 2024, revised 10 June 2025

Accepted for publication 24 June 2025

Published 12 August 2025



Abstract

Here we report the progress of the development and optimization of operational scenarios for ITER and beyond, focusing upon baseline, hybrid, and steady-state scenarios since 2007. This includes advancements made by the integrated operation scenarios (IOS) topical group of the international tokamak physical activity as well as contributions from the broader tokamak community. The key area of research involves developing IOSs that encompass tokamak physics, operation, and technology by utilizing integrated modeling and control strategies. This requires leveraging available actuators to simultaneously control plasma position and shape, MHD activities that could lead to disruptions, transport, plasma-wall interaction and power exhaust, fuel cycle, fusion burn, and tritium breeding. The control extends from the plasma initiation phase, through the current ramp-up, flat-top, start and end of the fusion burn, and current ramp-down, to the plasma termination phase. A review of the currently developed scenarios and modeling is provided in terms of (i) optimizing plasma initiation in ITER, (ii) preparing for the low activation phase to fully commission all tokamak systems and establish and validate physics and scenario conditions in preparation for deuterium-tritium (DT) operation, (iii) developing and preparing baseline and hybrid scenarios to demonstrate the feasibility of achieving these regimes within device constraints, (iv) exploring steady-state scenarios to meet ITER's steady-state goals, (v) evaluating and preparing actuators for ITER, (vi) developing integrated control solutions using shared actuators. The most notable achievements include; (i) the development of ITER demonstration discharges by matching various dimensionless parameters, (ii) the development of scenarios in an ITER-like tungsten environment and DT operation, and (iii) the development of scenarios in superconducting tokamaks, enabling long-pulse operations with similar coil constraints to ITER. Along with these significant achievements, outstanding issues and recommendations for further research and development are provided. Importantly, this study goes beyond simply updating the ITER Physics Basis; it carries profound implications for the broader field of burning plasma research, offering valuable insights and guidance for the next generation of fusion experiments and devices.

This paper is part of the Special Issue: *On the Path to Tokamak Burning Plasma Operation: A collection of papers prepared by the ITPA Topical Physics Groups reviewing progress in the development of the physics basis for burning plasma operation.*

Keywords: ITER, operation scenario, integrated modeling, tokamak control, burning plasma, ITPA

(Some figures may appear in colour only in the online journal)

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Acronym list

AI	Artificial intelligence
AE	Alfvén eigenmode
ATD	Approximate-then-design
AUG	ASDEX upgrade
BgB	Bohm-gyroBohm
CEM	Coherent edge-localized mode
CF	Core fueling
CD	Current drive
CIN	Central interlock network
CMIS	Control of minority/impurity species
CRM	Category response matrix
CS	Central solenoid
DCS	Discharge control system
DMS	Disruption mitigation system
DT	Deuterium–tritium
DTA	Design-then-approximate
ECE	Electron cyclotron emission
ECH/ECRH	Electron cyclotron heating/Electron cyclotron resonance heating
EFCC	Error field correction coil
EHO	Edge harmonic oscillation
ELM	Edge localized mode
ELMCC	ELM control coil
ETG	Electron temperature gradient
FIRE mode	Fast ion regulated enhancement mode
FNC	Field null configuration
FPO	Fusion power operation
FSM	Finite state machine
FW	First wall
GIS	Gas injection system
GTM	Generative topographic mapping
H&CD	Heating and current drive
HFS	High field side
HHFW	High harmonic fast wave
HS	Heating system
IBL	ITER baseline
ICRF	Ion cyclotron resonance frequency
ILW	ITER-like wall
IOS	Integrated operation scenario
IPB	ITER physics basis
IRP	ITER research plan
ITER	International thermonuclear experimental reactor
ITB	Internal transport barrier
ITG	Ion temperature gradient
ITM	Integrated tokamak modeling
ITPA	International tokamak physics activity
JA	Joint activity
JEX	Joint EXperiment
LFS	Low field side
LH	Lower Hybrid
LOC	Linear ohmic confinement
LSTM	Long short-term memory
LQR	Linear quadratic regulator
MHD	Magnetohydrodynamic
ML	Machine learning
MP	Machine protection
MPC	Model predictive control
MSE	Motional Stark effect
NBI	Neutral beam injection
NN	Neural network

NTM	Neoclassical tearing mode
ODE	Ordinary differential equation
ONFR	Off-normal and fault response
PCC	Pulse continuous control
PCS	Plasma control system
PCSSP	PCS simulation platform
PDE	Partial differential equation
PETRA	Plasma event TRiggering and alarms
PF	Poloidal field
PFC	Plasma facing component
PFPO	Pre-fusion plasma operation
PIS	Pellet injection system
PLCS	ParticLe control system
POPCON	Plasma OPERATION CONtour
PWI	Plasma wall interaction
QLKNN	QualiKizNN
R&D	Research and development
RF	Radio frequency
RL	Reinforcement learning
RMP	Resonant magnetic perturbation
RTPS	Real-time protection sequencer
RWM	Resistive wall mode
SISO	Single-input-single-output
SDN	Standard data network
SNU	Switching network units
SOB	Start of burn
SOC	Saturated ohmic confinement
SOL	Scrape-off layer
SPI	Shattered pellet injection
SRO	Start of research operation
TBM	Test blanket module
TEM	Trapped electron mode
TG	Topical group
TPC	Trapped particle configuration
V&V	Validation and verification
VDE	Vertical displacement event
VS	Vertical stability
VTM	Vessel thermal map

1. Introduction²⁴

Coordinated by Yong-Su Na

1.1. Requirement of scenarios

The goal of fusion research is to achieve practical, sustainable, minimally-polluting energy. This requires (i) high fusion performance, (ii) steady-state operation (as close to it as possible), and (iii) long duration in a tokamak. For high fusion performance, the triple fusion product, $n\tau_E T$ needs to satisfy the Lawson criterion [2] to achieve a positive net energy. The

²⁴ The progress on R&D described in this chapter includes topics of relevance to ITER, as well as to burning plasmas more generally. In some cases, the underlying physics R&D activities were motivated by specific features of the ITER 2016 staged approach baseline (see IRP for 2016 staged approach baseline ITR-2018-3 (provisional version) and ITR-2024-5 (final version) for details)). The new IBL 2024 currently under elaboration proposes modifications to several of the features and capabilities (e.g. first wall material, heating and current drive mix, etc) [1]. This will require additional R&D beyond that described in this chapter.

fusion multiplication factor or fusion gain, Q or Q_{DT} is a figure of merit for fusion performance, where Q refers to the ratio of power produced by fusion reactions to the external power supplied. For steady-state operation, the plasma current needs to be fully sustained by non-inductive (non-ohmic) means such as self-generated bootstrap current and external CD to minimize flux consumption. The non-inductive CD fraction, f_{NI} , serves as a key figure of merit for steady-state operation. Lastly, for long-duration operation, the pulse length must be sufficient to maintain a stationary condition, where the plasma profiles remain unchanged over various physical time scales, such as current diffusion time and wall saturation time. This is essential for plant reliability.

Numerous DEMO designs aim for Q values in the range of 10–50 to demonstrate that fusion power can become a viable and economically feasible energy source [3–5]. These designs generally feature a high non-inductive CD fraction, with the exception of EU-DEMO. The pulse length is on the order of thousands of seconds, for example, 7200 s in the case of EU-DEMO [6, 7].

Experiments with DT plasmas in TFTR [8] created 1.6 GJ fusion energy during three years. The peak P_{fus} was 10.3 MW from 3.7×10^{18} reactions per second, and peak fusion energy created in one discharge was 7.6 MJ. Subsequent experiments in JET achieved a peak P_{fus} of 16 MW ($5.8 \times 10^{18} \text{ s}^{-1}$ during DTE1 [9]), and a discharge producing 69 MJ during the recent DTE3 (consuming 0.2 mg of D and T). Although high non-inductive CD fraction could be achieved in advanced operation scenarios, the pulse length was severely limited compared with the current diffusion time. This leaves a large gap between past fusion experiments and DEMO.

To track progress from current research to practical fusion, as the next step, ITER is being constructed with the objectives of achieving $Q \geq 5$, non-inductive CD fraction f_{NI} up to 1.0, and a pulse t_{dur} duration of ≥ 300 s.

To achieve these goals, careful design of the sequences of actions that define the plasma discharge, so-called operation scenario is required. The operation scenarios need to be designed to account for the strong non-linear interactions between kinetic and magnetic profiles in the main plasma and their interactions with the SOL and divertor region, as shown in figure 1.

A limited number of actuators will be available to take actions for scenario development in ITER or any other high fusion gain device. They include magnetic field coils, NBI and wave HSs, gas and pellet injection, and divertor pumping. These actuators regulate magnetic fields, H&CD, torque, fuelling, and particle & power exhaust in order to obtain and control the plasma equilibrium, temperature profile, current density or q -profile, rotation profile, density profile, and neutral pressure and impurity content, respectively as depicted in figure 1. This also encompasses the control of MHD instabilities such as sawteeth and NTMs using waves. Additionally, potentially damaging transient events such as ELMs and disruptions should be suppressed or mitigated with, for example, using RMPs and SPI, respectively. Using these actuators, fusion power and Q , non-inductive CD fraction, and neutron flux and fluence may be controlled. The tokamak operation

scenarios need to be carefully designed from plasma initiation to termination, utilizing the available actuators while considering the non-linear plasma dynamics to achieve its goals.

In this chapter, we address the R&D progress and challenges associated with operational scenarios relevant to ITER, which can also be extended to DEMO and beyond. ITER has proposed three main operational scenarios, each serving a distinct purpose, as outlined in table 1; (i) baseline scenario, (ii) hybrid scenario, and (iii) steady-state scenario.

In the IRP [10], additional scenarios are being considered for low activation phases or non-active operation phases fueled by hydrogen and helium, which will be followed by a nuclear operation phase aimed at developing fusion power production with deuterium and tritium. This low activation phase is necessary to bring the tokamak plasma and plant systems up to the desired performance as the plasma current is progressively increased to 15 MA before full machine activation. During this phase, experiments can also investigate ELM and disruption loads, as well as commission the ELM control system using RMP and the DMS using SPI, as shown in figure 1.

Traditionally, the three main scenarios are classified by the shape of the safety factor or plasma current density profile, with schematics shown in figure 2. In the IBL or reference H-mode scenario, the plasma current is fully diffused in the stationary phase where the q -profile is monotonic with a large positive magnetic shear. The plasma performance of this scenario is expected to be that of conventional H-modes.

In the steady-state scenario, the q -profile can have reversed magnetic shear, with the current density having an off-axis maximum in the core region, and the overall plasma current reduced to 9 MA. In steady-state scenarios with weak magnetic shear, the current density remains nearly constant from the axis to approximately half the minor radius. ITBs are one of possible candidates for steady-state operation. With high confinement, high normalized beta ($\beta_N = \beta_t/I_p/aB_t$, $\beta_t \sim nT/B_t^2$), high poloidal beta ($\beta_p \sim nT/B_p^2$), and the resulting high bootstrap fraction, the scenario's requirement for fully non-inductive current appears feasible.

In the hybrid scenario, the q -profile is flat, with the on-axis value, $q(0)$, close to unity and a broad region of low magnetic shear. Since the same non-inductive current sources required for true steady-state operation can be applied throughout the duration of inductive scenarios, the term 'hybrid operation' has been coined to describe this type of operation [11]. The hybrid scenario offers improved confinement and higher β_N compared to conventional H-modes. It is also characterized by a low level of MHD activity. The reduced plasma current and increased β_N open the possibility for much longer pulse operations in ITER, which is suitable for testing TBMs [12]. This is due to the reduced flux required to supply inductive magnetic energy, the higher temperature reducing resistivity, and the increased bootstrap current [13].

The development of these scenarios requires understanding the plasma physics and technology in 'integrated' way. The shape of the q -profile, a key factor in classifying modes of operation, strongly influences transport, confinement, and the stability of both thermal and fast particles. It also directly and indirectly affects the pedestal structure. The value of q_{95}

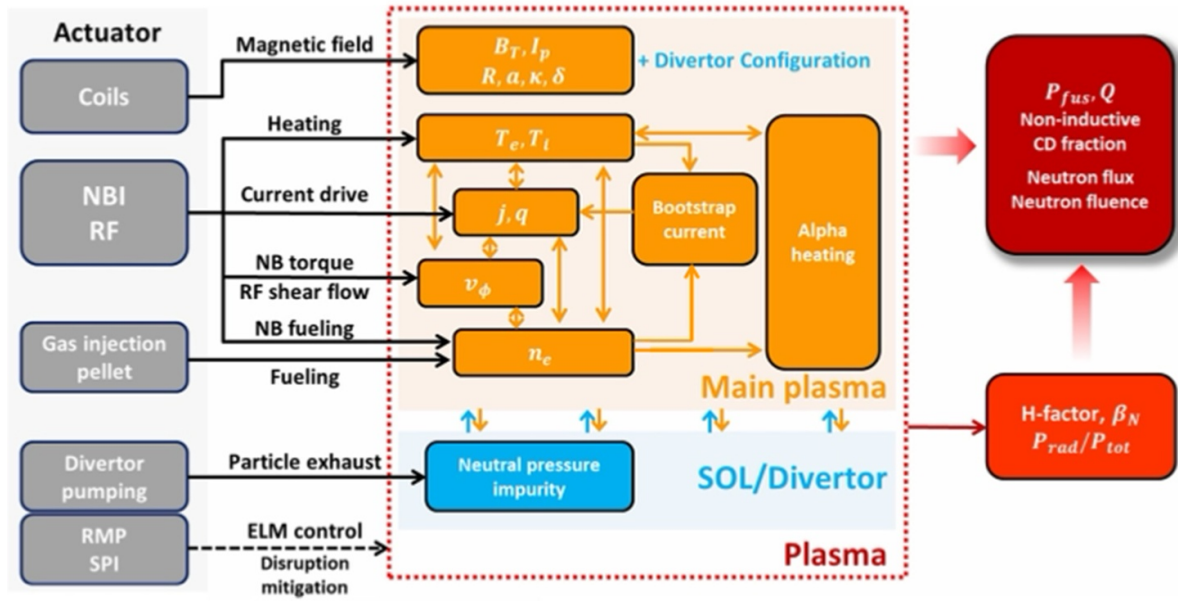


Figure 1. Schematic diagram of relations between actuators, plasma, and output parameters in a high fusion gain device such as ITER.

Table 1. ITER operation scenarios.

Scenario	Plasma current (MA)	Non-inductive fraction	H_{98y2}	β_N	Q	Burn duration (s)
Baseline	15	0.15	1.0	1.8	10	~ 400
Hybrid	~ 12	~ 0.50	1.0–1.2	2.0–2.5	5–10	≥ 1000
Steady-state	~ 9	1.00	≥ 1.3	≥ 2.6	5	3000

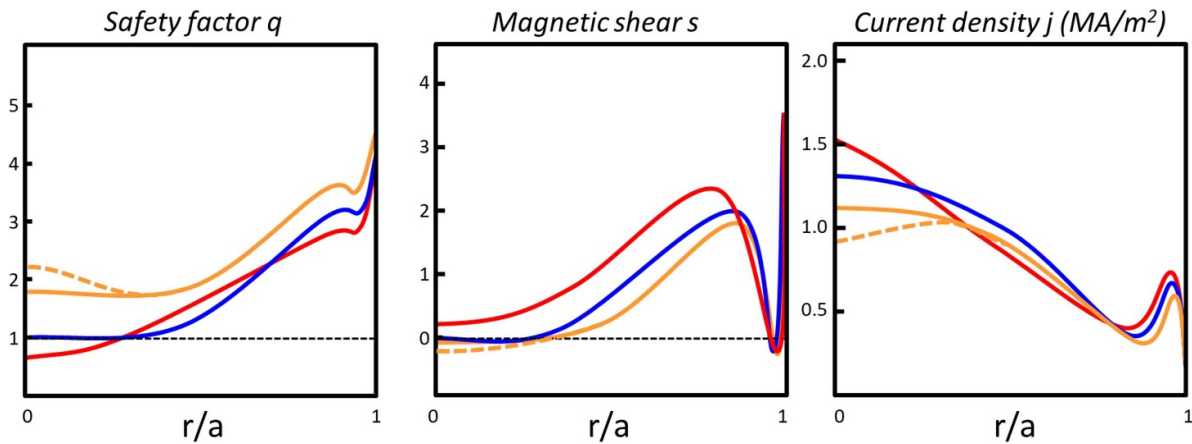


Figure 2. Schematic of the safety factor (q), magnetic shear (s), and current density (j) profile for three main scenarios in ITER; baseline (red), hybrid (blue) and steady-state (orange) scenario.

is correlated with the length of the SOL along the field line. Therefore, as shown in figure 1, the q -profile is interlinked with other plasma profiles through transport and stability mechanisms to determine fusion power and Q , the non-inductive CD fraction, and neutron flux and fluence.

The interaction between the main plasma, the SOL, and the divertor determines the compatibility of a scenario with ITER's tungsten divertor and FW. A detailed discussion of each mechanism is provided in other chapters of this special issue.

According to the IRP [10], ITER operations must demonstrate the integrated operation of technologies for a fusion power plant, test components required for a reactor, and evaluate concepts for tritium breeding modules, where neutron flux is the most critical factor.

Therefore, the development of ‘integrated operation scenarios’ (IOSs) is essential for the three main scenarios encompassing tokamak physics, operation, and technology, because (i) operating a burning plasma device involves many different physical and technological processes, as shown in figure 1 and

described in other chapters, and (ii) understanding individual physical mechanisms is necessary but not sufficient.

This chapter focuses on how we can integrate those physical mechanisms, controlled with sensors and actuators, into a complete operating scenario. This requires leveraging the available actuators to simultaneously control plasma position and shape, MHD activities that could lead to disruptions, transport, plasma-wall interaction and power exhaust, fuel cycle, fusion burn, and tritium breeding. The control must extend from the plasma initiation phase, through the current ramp-up, flat-top, start and end of the fusion burn, and current ramp-down, to the plasma termination phase.

Many of these issues cannot be fully addressed in present machines, as their experimental conditions differ from those of ITER. However, by applying the concept of ‘self-similarity’ and matching dimensionless parameters, it is possible to explore ITER-like conditions in current machines. This is because non-dimensional parameters define the internal physics of a complex system and serve as indicators of its fundamental state [14]. Therefore, R&D of ITER scenarios on existing machines is a crucial part of preparing for ITER operation by matching dimensionless parameters such as normalized Larmor radius, collisionality, plasma beta, and plasma shape. As shown in figure 1, the normalized fusion performance described by the confinement enhancement factor (H-factor or $H_{\text{scaling}} = \tau_E / \tau_{E,\text{scaling}}$) and β_N can be used to project the fusion gain, Q , in ITER. However, it is important to note that these 0D dimensionless parameters cannot capture all relevant physics, such as alpha particle and impurity effects or the physics of the divertor. Furthermore, multi-dimensional parameters, such as plasma profiles, can result in different plasma conditions even when the 0D parameters remain the same.

Gaining a critical understanding of the physics issues that ITER will face is essential for improving operational efficiency. While some issues can only be fully addressed on ITER, many operational scenario challenges can be explored on existing devices and extrapolated to ITER through modeling and simulations. Integrated, holistic modeling and simulation can establish a physics basis for the proposed operating modes in ITER, in conjunction with results from current tokamak experiments.

Integrated, holistic modeling involves incorporating models of plasma equilibrium and stability, energy, thermal/fast/impurity particle, current and momentum transport, as well as SOL/edge/pedestal dynamics. These are combined with H&CD, CS and PF coils, and fueling, in time-dependent discharge simulations in a self-consistent manner. Such modeling and simulations are important to address the highly non-linear interdependencies of plasma phenomena, as shown in figure 1, and to provide predictions for plasmas in future machines. They also play a key role in diagnostic development. Examples of such modeling can be found in [15–21].

Since it is not possible to reproduce all the physics parameters of ITER plasmas simultaneously in current experiments, simulations are used to project to ITER regimes using theory-based physics models that are being tested against present tokamak experiments. Therefore, V&V and benchmark of

these physics models are key elements in scenario design and predictions for all phases of ITER operation. Examples of this can be found in [15, 17, 22–25]. Note that the terms validation, verification, and benchmark are used in this chapter as follows: Validation refers to the process of determining whether a model, code, or simulation accurately represents experiments or real-world phenomena within its intended scope. Verification refers to the process of ensuring that a model, code, or simulation is correctly implemented according to its design. In contrast, a benchmark refers to a standardized test or reference case used to evaluate and compare the performance, accuracy, or efficiency of different codes or computational models in the context of code or model comparisons.

In particular, V&V and benchmark of H&CD and fueling models are necessary to understand how the ITER actuator systems can shape or maintain the current density for a range of expected temperature profiles, density profiles, and species concentrations. This is needed to assess whether upgrades are required for long-pulse operation. Furthermore, V&V and benchmark are essential for modeling suites that integrate various dynamic physics models, as model assumptions critically affect predictions for fusion performance, achievable pulse length, and the requirements for actuator systems, as well as plasma control in a fusion reactor.

Successful V&V and benchmark of models and codes will allow the extrapolation of scenarios from present devices to ITER conditions with high confidence. Comparing these predictions with initial experiments on ITER will help guide the experimental program and improve the models and assumptions used for predictive simulations.

One of the key challenges for ITER is to develop robust, simultaneous, and integrated scenario control solutions for the core, SOL, and divertor to operate within safety, stability, and controllability boundaries. This is important to guarantee high performance, safe, and disruption-free tokamak operation using a limited set of shared actuators. It involves controlling magnetic equilibrium, MHD modes, kinetic and magnetic profiles, burn conditions, divertor and FW heat flux, as well as impurity and radiation levels.

Active control in tokamak operation is important as performance demands push operations closer to safety, stability, and controllability boundaries. The goal is to maximize performance within safety limits (e.g. within tolerable ranges of heat and particle fluxes on PFCs) while preventing the plasma from crossing stability and controllability boundaries, which would inevitably lead to loss of control and plasma disruptions. Achieving this requires the development of control-oriented models to enable extrapolation to ITER, sophisticated model-based plasma control solutions, and the demonstration of these control solutions on existing machines.

1.2. Goal and tasks of IOSs TG

The IOSs TG of the ITPA has been founded to contribute to establishing operational scenarios in burning plasma experiments, particularly candidate scenarios in ITER by addressing issues mentioned in section 1.1. These include standard inductive operation, hybrid operation, and steady-state

operation. Towards this end, the group encourages and coordinates international collaboration on experimental, code development and modeling activities among institutions and machines around the world to prepare ITER operation. The group identifies and formulates priorities for related R&D activities. Based on these activities, the group recommends guidelines and methodologies for the physics and technical design of burning plasma experiments. The group promotes publications and presentations on its activities.

Through experimental, numerical, and theoretical studies related to and organized by the IOS TG, the following tasks are being pursued:

- Develop and establish operation scenarios, especially those foreseen in ITER, with emphasis on;
 - safe and reliable plasma initiation, current ramp-up, current ramp-down, and plasma termination
 - safe and reliable sustainment of the current flat-top
 - identification of operational boundaries and performance
 - standard-inductive, hybrid, and steady-state operation.
- Identify and evaluate the capabilities of actuators — H&CD, fuelling, particle exhaust, (seed) impurity control, and torque—for achieving identified scenarios
- Assess integrated plasma control solutions, especially for achieving steady-state operation in ITER and future burning plasmas (e.g. burn control), with emphasis on real-time control
- Evaluate and propose suggestions for plasma development in ITER with emphasis on the consistency between plasma/scenario development and hardware status for each phase of machine operation
- Promote integrated modeling activities for ITER scenarios, with emphasis on validating the available scenario modeling codes and exploring the operational boundaries for the proposed ITER scenarios.

The IOS TG organizes JEXs and JAs as the major drivers of its activities, which are carefully coordinated to pursue the tasks listed above. JEXs facilitate experiments across various devices and may also include some modeling and simulation activities. JAs, on the other hand, focus on coordinating modeling and simulation efforts, including benchmarking various codes and models. These JEXs and JAs are summarized in [appendix](#).

Since the publication of the progress of IPB in 2007 [11], significant progress has been made by the IOS TG as well as contributions from the broader tokamak community which will be addressed in this chapter. The major progress is highlighted briefly in the following sub-section.

1.3. Major progress from the previous special issues

The most outstanding achievements with respect to experiments include; (i) development of ITER demonstration discharges by matching several dimensionless parameters, (ii) development of scenarios in ITER-like tungsten environment in AUG and JET, with T and DT operation at JET, and (iii) the development of scenarios in superconducting

tokamaks, enabling long-pulse operations with a similar coil constraints to ITER. Many devices have attempted to establish ITER-relevant or ITER-like conditions by matching some key dimensionless parameters to explore ITER demonstration discharges. These efforts include replicating the ITER plasma shape, operating with low external torque, utilizing predominant electron heating, achieving ELM suppression, implementing a radiative divertor, employing EC-assisted plasma initiation, sustaining long-pulse operation, conducting T&DT operation, and so on.

AUG progressively installed tungsten-coated tiles over several years, achieving 100% tungsten coverage by 2007, and JET installed an all-metal wall in 2011, with beryllium in the main chamber and tungsten in the divertor, described as ‘ITER-like wall (ILW)’ [26], though it is no longer ITER-like, as ITER decided in 2023 to modify the planned FW armor material, replacing beryllium with tungsten. The superconducting tokamaks, EAST and KSTAR, have begun operations and have demonstrated long-pulse, high performance operations.

There have been significant achievements in modeling and simulations which encompass; (i) laying the foundation for predictive simulations for ITER through benchmark studies, (ii) developing and employing more advanced anomalous transport and pedestal models for integrated modeling, and (iii) exploring simulations of the entire discharge process, from plasma initiation to termination, including simulations with free-boundary codes. Benchmark studies have been conducted for plasma burn-through, energy and particle transport, and various H&CD schemes. More advanced anomalous transport models, such as TGLF and QuaLiKiz, and a pedestal model, such as EPED1, have been developed and implemented for integrated modeling of ITER scenarios. Free-boundary equilibrium codes, including CORSICA, DINA, and TSC, have been utilized to simulate the controllability of ITER plasmas within the constraints imposed by the voltage capabilities of the power supply systems and the FW geometry, covering the current ramp-up to ramp-down phases.

The rest of this section summarizes major progress since 2007 in the following areas: Optimizing plasma initiation in ITER; preparing for the low activation phase to fully commission all tokamak systems and establish and validate physics and scenario conditions in preparation for DT operation; developing and preparing baseline and hybrid scenarios to demonstrate the feasibility of achieving these regimes within device constraints; exploring steady-state scenarios to meet ITER’s steady-state goals; evaluating and preparing actuators for ITER; and developing integrated control solutions using shared actuators.

1.3.1. Plasma initiation. Due to the superconducting coils, large major radius, and thick double vacuum vessel layers, the toroidal electric field available for plasma breakdown and burn-through (i.e. plasma initiation) in ITER is limited to $\sim 0.30\text{--}0.35\text{ V m}^{-1}$, which is quite lower than the values used in most operating tokamaks. On the other hand, the heating power required for successful plasma burn-through in ITER

will be much higher than in current devices due to the very large vacuum volume. Therefore, careful optimization of the plasma initiation phase is crucial for successful plasma initiation in ITER.

Outstanding improvements have been made in plasma initiation modeling and code validation using experimental data. The major plasma initiation codes (DYON, SCENPLINT, and BKD0) have been verified through detailed comparisons with one another. According to predictive simulations of ITER plasma initiation, it was found that beryllium (Be) content below 5% would not affect the plasma burn-through, but modest carbon (C) content of $\sim 1\%$ – 2% can considerably increase radiated power. Therefore, it is recommended to keep the carbon concentration below 0.5% during the plasma initiation phase. Ohmic plasma burn-through in ITER will only be possible at a pre-fill gas pressure of less than 1 mPa, which is much lower than the typical pre-fill gas pressure (~ 5 mPa) used in current devices. Simulations also predict that 3–4 MW of additional heating, such as EC waves, could enable successful plasma burn-through at 5 mPa. However, the power absorption efficiency of the planned ITER EC wave scheme (i.e. O1 and X2 modes) would be low at the low electron temperature (T_e) and density (n_e) in the plasma initiation phase, meaning that higher EC wave power would be required. Plasma initiation at a lower pre-fill gas pressure would reduce the required EC wave power. For example, assuming a pre-fill gas pressure of 0.5 mPa, SCENPLINT-DINA simulations indicated successful plasma initiation with 2 MW of EC wave injection as a reference for plasma initiation in ITER. However, it should be noted that such low-pressure operation increases the risk of runaway electron generation, which could be further exacerbated by EC injection. In FTU experiments, runaway electrons were generated even at electric fields lower than the Dreicer electric field when EC power was injected.

Experiments on various devices have shown that ohmic plasma initiation with an electric field comparable to that of ITER is marginally possible with optimized magnetic configurations. The electric field required for plasma breakdown does not significantly depend on device size, confirming that the main challenge for plasma initiation is burn-through. No outstanding issues with plasma initiation were reported for Be or high-Z walls, as predicted by the simulations. Dedicated experiments in AUG (X2), DIII-D (X2), FTU (O1), HL-2A (O1), KSTAR (X2), Tore Supra (O1, X2), and the stellarator TJ-II (X2) have demonstrated successful ECH-assisted plasma initiation in ITER-like modes. However, due to the low absorption efficiency of the O1 and X2 modes, ITER will require much higher EC wave power injection. Mode conversion from O1 to X1 using an in-vessel mirror could increase the efficiency of wave power absorption.

1.3.2. Low activation phase scenarios. The ITER low activation phase, comprising the SRO and early FPO phases in the IRP, represents crucial stages for system commissioning, validating and refining the physics basis and control schemes for ITER plasma operation, and advancing the project towards demonstrating the viability of fusion energy at a reactor scale.

For the first time, these phases will allow for experimental demonstration and exploration of plasma conditions on a fusion reactor scale.

The SRO phase begins with plasma operations using hydrogen as non-fusion fuel or deuterium for a limited period to ensure a low activation. This phase is essential for validating and commissioning ITER's core systems, including coils, H&CD, fueling and pumping systems, control and disruption mitigation schemes, and diagnostics at low neutron activation, enabling maintenance, repair, and upgrade activities.

Key activities during SRO include systems and plasma commissioning and routine operation up to 15 MA/5.3 T in hydrogen in L-mode and approximately 7.5 MA/2.65 T in deuterium in H-mode. A gradual, stepwise increase in current at constant field is planned considering the operational window for wave heating. While SRO primarily uses hydrogen plasmas, deuterium plasmas are foreseen to be used as accessing H-mode proves challenging with hydrogen. Early H-mode access is planned at half field (5–7.5 MA/2.65 T) to commission H-mode operation and associated ELM mitigation techniques as soon as possible. Heating power of up to 40 MW from ECRH and 10 MW of ICRH will be used early in SRO, later supplemented at the start of FPO by a maximum of 33 MW of NBI heating and possibly up to 20 MW of ICRH.

Further operational constraints and dependencies require careful consideration, including coil current and force limits, MHD stability limits, field and density constraints for waves, NBI shine-through limits, density limits, the L–H transition power threshold, ELM mitigation, core-edge-divertor integration constraints, and the dependence of the selected main ion isotope(s) on core confinement, edge stability, divertor conditions, and tungsten sputtering.

Early FPO will involve scenario operations similar to SRO but with additional heating power (33 MW NBI + increased ECRH/ICRH power), operation with tritium (H+T plasmas) and new components becoming available (W water-cooled FW, TBMs, etc). Benefiting from these later system upgrades, preparations will be made for high performance DT operation in later FPO phases.

Specific physics questions relevant to the low activation phase, such as the isotope dependence of plasma properties, H-mode operation near the L–H transition power threshold, and the feasibility of early H-mode at low current and field, have been addressed and resolved through experimental and modeling studies, including as part of the ITPA IOS activities.

Experiments across multiple devices have provided important insights into H-mode access conditions that are relevant to ITER's low activation phase. In dedicated experiments at JET with helium plasmas, which were considered in earlier versions of the IRP to enable the investigation of H-mode operation at an early stage in SRO, it was demonstrated that the L–H power threshold in helium follows similar dependencies as in hydrogenic plasmas. It is $\sim 40\%$ higher than in deuterium but $\sim 30\%$ lower than in hydrogen. The experiments confirmed that helium operation could also be more favorable for achieving H-mode compared to hydrogen due to reduced NB shine-through losses and possibly lower power requirements at a given density. Experiments in recent JET-ILW and AUG

^4He campaigns found however that achieving type-I ELMy H-mode required significantly higher power than the L–H transition, sometimes making it inaccessible. These findings suggest no clear advantage of using ^4He over hydrogen for the purpose of ELM control commissioning in SRO. Studies further indicated that divertor configuration, impurities, and plasma rotation could highly impact the L–H threshold which may be relevant for H-mode operation in SRO.

H&CD experiments focused on questions related to the establishment of viable H-mode scenarios at reduced power levels in ITER's low activation phases. One challenge identified was the use of ECH at low toroidal field, where third-harmonic EC absorption would need to be considered which may be initially inefficient at low plasma temperatures. Experiments at AUG and DIII-D have however validated the feasibility of this approach, showing that once ECH initiates, it rapidly enhances power absorption through a self-reinforcing process.

Fuelling requirements in ITER low activation plasmas and the effectiveness of pellet injection and gas puffing have been assessed. While pellet fuelling is considered to be highly efficient for HFS injection in the foreseen low activation scenarios, its use may be limited to specific conditions, requiring gas puffing as main actuator to maintain appropriate density levels. Fuelling in helium plasmas may be more delicate i.e. with respect to divertor detachment control.

Comprehensive simulations with integrated modeling codes have helped refine ITER's operational scenarios for the low activation phase. Modeling of H-mode access in hydrogen or helium at 5 MA/1.8 T which was considered in earlier versions of the IRP confirmed that ECH could successfully drive early H-mode transitions, despite the challenges posed by third-harmonic absorption. Predictions also indicated that increasing heating power to $\geq \sim 30$ MW would significantly expand the H-mode operational space, particularly in helium plasmas.

Studies on isotope dependent transport effects showed deviations from simple gyroBohm scaling i.e. due to turbulence suppression mechanisms like $E \times B$ shear and zonal flows and a strong edge-driven favorable dependence of confinement on the isotope mass. These effects suggested that while hydrogen operation might struggle to achieve type-I ELMy H-mode, helium and deuterium plasmas could support robust H-mode confinement.

Neutron yield modeling in low activation phases in hydrogen and deuterium plasmas estimated production rates to be limited enabling the installation of in-vessel components foreseen in the Post-SRO assembly phase, while respecting shut-down dose rate requirements for workers involved in these activities.

Finally, integrated core-SOL/divertor modeling revealed that the envisaged low activation phase scenarios should be compatible with foreseen divertor operation and control schemes. Deep detachment may need to be avoided for successful density control, while pellet fuelling might be essential for maintaining L-mode plasmas at 15 MA/5.3 T at reduced input power. In helium plasmas, gas fuelling alone was found to be sufficient for achieving H-mode at 7.5 MA/2.65 T.

These findings provide crucial guidance for ITER's early operational phases, ensuring that H-mode access, plasma control, and heating schemes are optimized for efficient performance in both low activation and later DT phases.

1.3.3. ITER baseline scenarios. The description of the IBL scenario also falls into the category of predictions made before the completion of the device's construction. The approaches taken between 2008 and 2022 can be categorized into; (i) theoretical modeling, based on established physics models, to design and simulate expectations for a $Q = 10$ plasma, and (ii) extrapolations through JEXs on scaled baseline discharges from existing devices.

In the early 2010s, simulation efforts focused on numerically producing the entire span of a plasma discharge, including start-up, current ramp-up, current flattop to burn, and scheduled ramp-down/termination, in order to identify engineering limits. Free-boundary codes such as TSC, CORSICA, and DINA were utilized. Later in the decade, more complex physics models and integrated modeling suites were developed, contributing to studies such as (i) the entry to burn, (ii) simulations for particle control and fueling scenarios, and (iii) evaluations of tungsten production and accumulation. Based on this progress, time-dependent predictive simulations of the IBL scenario have been performed, with target coil currents guided by parameter variations in static equilibria. These simulations represent the current state-of-the-art in producing scenarios representative of experimental conditions. It has been shown that the requirements for the PF coils are consistent with the operating space boundaries set by the maximum allowed coil currents and forces.

Core particle transport simulations in the IBL scenario were benchmarked using a collection of ten codes widely used in tokamak transport modeling. This benchmark included the formulation of particle transport equations, differences between electron and ion solvers, and comparisons of pellet and edge gas fueling modules in both stationary and time-dependent simulations. The differences between simulation results were investigated, and it was found that the selection of the radial coordinate was a key reason for the discrepancies, mainly due to variations in edge grid distribution. Modeling efforts on neutron yield predictions for JET DT campaigns under JET-ILW were benchmarked with the DTE1 experiments, yielding reasonable results for the P_{fus} predictions in the DTE2 campaign conducted in 2019–2020. This highlighted the importance of validating physics models against experimental data.

The IBL scenario has been explored in experiments on AUG, Alcator C-Mod, DIII-D, JET, KSTAR, and TCV. Since the flux consumption of the scenario is important as ITER will operate close to the limitations of the power supplies and magnetic coil systems, the whole discharge from the current ramp-up, flattop, and ramp-down phases of the ITER discharge has been examined by matching dimensionless parameters such as q_{95} , β_N , $f_{GW} = n_e/n_{GW}$, t_{ramp}/τ_{CR} , and plasma shape as far as possible, where n_{GW} and f_{GW} are the Greenwald density ($n_{GW} \equiv I_p/(\pi a^2)$) [27, 28] and its fraction ($f_{GW} \sim \langle n_e \rangle / n_{GW}$),

respectively, t_{ramp} is the duration of the plasma ramp phase and τ_{CR} is the current relaxation or diffusion time which is proportional to the effective ion charge and electron temperature, T_e ($\tau_{\text{CR}} \sim Z_{\text{eff}} T_e^{1.5}$). It was found to be challenging to achieve stable, stationary conditions when the loss power is just above the threshold for entering H-mode. The primary effect is that the ELMs become infrequent, leading to a rise in electron density and, in devices with metal walls, an increase in impurity density and core radiation. Confinement often degrades as the density approaches the empirical limit, n_{GW} . An increase in the L–H power threshold contributes to this degradation as the density rises.

Experiments with metal walls over the past decades have shown that wall materials play a important role in IBL performance by setting the edge boundary conditions. The main source of high-Z impurities is the main chamber wall, rather than the divertor tiles. With a tungsten wall, discharges with $\beta_N < 2.0$ due to lower heating power, suffered from reduced ELM frequency, resulting in central density peaking and impurity accumulation. Gas puffing helps to avoid impurity accumulation but leads to a decrease in H-factors. Boronization can restore H-factors, but large type-I ELMs can occur, which are difficult to mitigate at $q_{95} \sim 3$. In the AUG experiments, an alternative scenario with larger q_{95} was suggested to maintain $\beta_N \sim 2.0$ and $H_{98y2} \sim 1.0$. Material changes also caused significant deviations from empirical scaling laws for energy confinement time in H-mode, which were developed under carbon wall conditions.

Reduction in applied torque was found to narrow the stable operating range in DIII-D due to the onset of an $n = 1$ tearing mode just after the transition from ohmic to H-mode for a range of $\beta_N = 1.5$ – 2.0 .

1.3.4. Hybrid scenarios. Since 2007, hybrid scenarios have been further studied across the wide range of operation windows, achieving high performance with $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$ surpassing the IBL scenario in stationary conditions lasting longer than $5\tau_E$ in AUG, DIII-D, EAST, JET, JT-60U, and KSTAR. These experiments cover the range of dimensionless parameters: $q_{95} = 2.8$ – 6.5 , $\rho^* = 0.04$ – 0.14 , $\nu^\dagger = 1 \times 10^5$ – 9×10^6 , and $f_{\text{GW}} = 0.30$ – 0.95 , where proxies are used for $\rho^* = (W_{\text{th}}/V)^{1/2}/Ba$, $\nu^\dagger = (n^3 V^2/W_{\text{th}}^2)(R^5/a^3)^{1/2}$, with W_{th} in MJ and n in 10^{19} m^{-3} .

Various experimental approaches have been attempted to access the hybrid regime; early heating during the current ramp-up phase, optimization of the plasma current ramp-up including current overshoot, late heating during the current flattop phase, and off-axis H&CD during the flattop phase. Among these, the early heating approach is routinely applied in many devices, while the current overshoot approach was primarily used in JET to improve confinement. These approaches have been established to tailor the q -profile with a large region of low magnetic shear in the core and $q_{\text{min}} \sim 1$. This ‘hybrid’ q -profile could be maintained afterwards, although the mechanism behind sustaining this profile still requires further investigation.

The development of integrated hybrid scenarios has been actively explored in ITER-relevant conditions, such as low external torque, ELM suppression with RMP, radiative divertor operation with ILW, long-pulse operations, and T&DT operation. DIII-D successfully developed hybrid scenarios with external torque of $\sim 1 \text{ Nm}$, which is expected to drive a similar amount of rotation as NBI in ITER. However, energy confinement was found to decrease as torque was reduced. ELM suppression using RMP has been attempted in DIII-D and KSTAR hybrid scenarios. In DIII-D, type-I ELM suppression was achieved with $n = 3$ RMPs in plasmas up to $\beta_N \sim 2.5$, with minimal confinement degradation. In KSTAR, $n = 1$ and 2 were used to stabilize ELMs, and suppression was sustained for up to 45 s, although some confinement degradation was observed.

Radiative divertor operation using nitrogen was implemented in AUG and JET, where the heat flux to the divertor was reduced by a factor of 4. While energy confinement degradation was observed in JET, confinement was improved in AUG due to an increase in pedestal temperature. Neon or argon was used in DIII-D, where the heat flux was reduced by a factor of 2.5, and energy confinement was sustained.

Long-pulse operation has been achieved in JT-60U up to 28 s with β_N of 2.6 and $H_{98y2} > 1$ and in JET for up to 20 s with $\beta_N \sim 2.5$ and $H_{89} \sim 2$. KSTAR also achieved long-pulse of 20 s with $\beta_N > 2.4$, though gradual confinement degradation was observed over time. JET established hybrid scenarios in T and DT by optimizing the gas waveforms to avoid impurity accumulation and set new records of the fusion power, exceeding 12 MW for 3 s and 10 MW for 5 s.

Energy confinement scaling studies have been performed using data collected from various devices and dedicated experiments. These studies found a weaker scaling with normalized gyroradius and a stronger scaling with collision frequency than in the standard H-mode scaling.

Integrated modeling has been conducted to project hybrid scenarios for ITER. Energy and current transport simulations were first benchmarked using various integrated suites such as ASTRA, CORSICA, CRONOS, METIS, ONETWO, TOPICS, and TSC/TRANSP. Some differences were identified between the codes, primarily due to the H&CD source models, the specification of input parameters in the GLF23 model, and the treatment of equilibrium. Subsequent modeling efforts using integrated suites have shown that a $Q > 5$ hybrid scenario with $q \geq 1$ is feasible using 33 MW of NBI, 20 MW of IC, and/or 20 MW of EC, under certain assumptions, including the pedestal temperature. Additionally, it was found that lowering the plasma density could extend the duration of the burn phase to $\sim 1000 \text{ s}$ with $Q \geq 5$ for $I_p \geq 13 \text{ MA}$ even without hybrid-like q -profile. This could be a potential candidate for long-pulse operation for TBM tests in ITER.

1.3.5. Steady-state scenarios. Steady-state scenario research in ITER aims to optimize performance with a target $Q = 5$ in stationary discharges where the current is driven fully non-inductively for a duration of up to 3000 s, i.e. more than at least current relaxation times. Theory-based modeling

suggests that plasma confinement enhancement $H_{98y2} > 1.5$ is likely required to achieve fully non-inductive operation with $Q \sim 5$ ($I_p \sim 8\text{--}9$ MA, $\beta_N \sim 2.8\text{--}3.1$, $\beta_t \sim 2.2\text{--}2.5$) using ITER's FPO phase H&CD tools [10] and their possible upgrade options. Plasma density plays a crucial role, and core density exceeding the Greenwald fraction of 1 may relax the confinement enhancement requirement.

Two main options for reaching ITER's steady-state goals are; (i) scenarios with strong reversed shear and high bootstrap current (high bootstrap fraction, f_{BS} , scenarios) and (ii) scenarios with weak reversed or low magnetic shear (high auxiliary power fraction, f_{Aux} , scenarios).

Significant progress has been made in experiments toward demonstrating high normalized fusion performance in steady-state operation. However, simultaneously achieving both high performance and steady-state conditions remains very challenging. Fully non-inductive discharges tend to fall somewhat short of the normalized performance goals relevant to $Q \sim 5$ in ITER, while high performance discharges tend to fall short of the fully non-inductive goal of $f_{NI} \sim 1$. Additionally, discharges with ITER-like shapes tend to exhibit slightly lower performance than those with stronger shaping. It is also worth noting that some of the very high plasma confinement enhancements observed in present-day tokamaks may be due to turbulence stabilization effects that are not expected to play a dominant role in ITER.

Since 2007, a major development has been the start of high power plasma experiments in the two superconducting tokamaks, EAST and KSTAR. EAST achieved a reproducible, fully non-inductive H-mode duration of 403 s, with high normalized fusion performance; $H_{98y2} \sim 1.3$, $\beta_N \sim 1.6$, $\beta_p \sim 2.5$, $f_{BS} > 50\%$, $n_e/n_{GW} \sim 0.7$ using wave-only H&CD and tungsten divertor. KSTAR achieved high $\beta_N \sim 3$ in the high l_i scenario ($l_i \sim 1$) sustained for 12 s at $q_{95} = 4$ and $q_{min} \sim 1$. More recently, KSTAR developed a new ITB scenario regulated by fast ions, known as FIRE mode, with central ion temperature $T_i \sim 10$ keV, sustained for up to 50 s. This scenario exhibited no ELMs due to an L-mode or I-mode edge, no notable core impurity accumulations, and low loop voltage.

Another outstanding development has been the operation of advanced scenarios in a metal wall environment in both AUG and JET. On AUG, 100% tungsten coverage was reached in 2007, and advanced scenario development began around 2012. Work on plasmas with q_{min} above unity led to a scenario with nearly fully non-inductive CD at a current of 600 kA, with $q_{95} \sim 7.1$, $\beta_N \sim 2.2$, $H_{98y2} \leq 0.9$, and low line averaged central density of $n_e \sim 4 \times 10^{19} \text{ m}^{-3}$. Higher performance with $\beta_N \sim 2.7$, $H_{98y2} \sim 1.2$, and $q_{95} \sim 5.3$ was achieved at 800 kA, though with a lower non-inductive fraction. In AUG, boronization is an important tool to reduce and control tungsten sources, allowing reliable operation in high confinement regimes.

In JET, experiments conducted with a carbon wall achieved plasmas with $\beta_N \sim 2.7$, $H_{98y2} = 1.2$, $q_{95} = 4.7$, and $q_{min} \sim 1.8$ for the first time at global $\rho^*/\rho_{ITER}^* \approx 2$, $\nu^*/\nu_{ITER}^* \approx 4$, and sustained for times longer than $10\tau_E$. The improved confinement, leading to $H_{98y2} > 1$, was attributed to the presence of an ITB in the ion temperature profile. After the installation of

a beryllium wall and tungsten divertor in 2011, a scenario was developed to achieve elevated q -profile plasmas with the ILW. ITBs have been clearly observed in several discharges during this experimental effort.

DIII-D and JT-60U have pursued both high f_{Aux} and high f_{BS} approaches toward achieving $Q = 5$ in ITER. On DIII-D, the high f_{Aux} approach produced nearly stationary plasmas with $H_{98y2} \geq 1.3$ and $q_{min} \sim 1.4$, sustained at $\beta_N \sim 3.5$ and $\beta_t \sim 3.6\%$ for two current relaxation time scales (~ 3 s), although with $f_{NI} \sim 70\%\text{--}75\%$. In the high f_{BS} approach, fully non-inductive conditions were established at reduced plasma current ($\beta_N \sim 3.3$, $H_{98y2} \sim 1.5$, $q_{95} \geq 10$). This scenario was then extended to lower q_{95} ($\sim 6\text{--}8$), showing compatibility of excellent core confinement with low edge pedestal (for small ELMs) and detached divertor operation, albeit at $f_{NI} < 1$.

Before ending operations in 2008, JT-60U focused on expanding the operational space of advanced tokamak plasmas towards higher β_N . In weak magnetic shear plasmas (high f_{Aux} scenario), high $\beta_N \sim 3$ exceeding the $n = 1$ ideal MHD limit without a conducting wall ($\beta_N^{no-wall}$) was sustained for ~ 5 s. In reversed shear plasmas (high f_{BS} scenario), utilizing wall stabilization enabled expanding the operational regime towards lower q_{95} and higher β_N . Transiently, $\beta_N \sim 2.7$ was achieved with $H_{98y2} \sim 1.7$ at $q_{95} \sim 5.3$ ($f_{BS} \sim 0.9$), and with both internal and edge transport barriers, even under conditions of very low toroidal rotation.

1.3.6. Actuators for integrated scenario operation.

Actuators play a crucial role in creating the plasma conditions necessary for successful tokamak fusion reactors. These conditions include the sustained production of large amounts of fusion energy in burning DT plasmas. The IOS TG has focused on the development and application of computer codes for modeling and predicting actuator effects, benchmarking these codes, and conducting experiments to quantify actuator effects and validate predictions.

For NBI, benchmarking between Fokker–Planck and Monte Carlo calculations has been conducted, and the results converged to approximate agreement. Experiments to validate NBCD have been performed on AUG, DIII-D, JT-60U, and MAST. The current profile broadened by off-axis NBCD was evaluated using MSE diagnostics, showing good agreement between the measured and calculated NBCD profiles. Comparisons of the measured and calculated CD profiles suggested the presence of anomalous fast ion transport, likely caused by turbulence and certain MHD instabilities.

For ICRF heating, benchmarking of predicted ITER ICRF performance has been carried out. Results of using TRANSP [29] TORIC [30] wave solver predictions with those of AORSA, CYRANO, EVE, PSTELION, and TASK/WM revealed disagreements by a factor of two or more in heating power partitions, particularly for cases involving second harmonic ^3He heating in bulk hydrogen plasmas. The modeling indicated that mode conversion and parasitic slow mode absorption near the peripheral LH resonance could be optimized by steepening the density profiles locally near the resonance, affecting both the fast and slow modes in the SOL. In

experiments, fundamental ($n = 1$) deuterium heating with D-NBI has proven to be the most fusion-power efficient method in JET's DTE2 experiments.

For EC H&CD, numerical codes were benchmarked. The study compared predictions for several representative cases using TORAY-GA, GENRAY, CQL3D, and BANDIT-3D, as well as ray tracing with TORBEAM, OGRAY, and GRAY. The agreement between codes for off-axis applications was excellent.

Although LHCD is not currently being considered for ITER, it has very attractive properties, including CD capability and steerability. Benchmarking of LHCD codes and predictions for LH absorption by fusion alpha particles were addressed. Predictions from CQL3D-GENRAY, DELPHINE, ACCOME, SPOT, DKE, and LSC were discussed and compared for an ITER-like steady-state scenario plasma. Differences in the magnitude of the driven current were identified, with the more comprehensive 2D (velocity space) Fokker–Planck ray tracing models predicting the highest values, while the response function codes predicted lower values. Various experiments were conducted on C-Mod, EAST, FTU, JET, and Tore Supra.

Synergies were found between heating and other actuators, with LH waves providing the highest CD efficiency (defined as the ratio of driven current to injected wave power), though with limited control capability. In contrast, EC waves drive highly localized currents, making them particularly well-suited for control purposes, but their CD efficiency is much lower than that of LH waves (typically an order of magnitude lower in present-day experiments, but predicted to be of the same order of magnitude at reactor-relevant temperatures).

Pellet, gas, and NB fueling are considered actuators for density control, which is crucial for plasma performance. PFCs also need to be considered as a source and sink of fuel, ash, and other impurities to accurately predict plasma performance. The replacement of the JET carbon wall with the ILW reduced the degradation of thermal plasma energy confinement with heating power. However, the accumulation of tungsten in the plasma core shortened the duration of high-power discharges in JET during the DTE2 campaign. Methods for controlling this were developed during the recent JET DTE3 campaign.

1.3.7. Scenario control. The mission of the IOS TG includes the development of integrated control solutions for robust scenario realization and sustainment. The efforts on scenario control include; (i) kinetic control, from density and beta control to burn control, (ii) kinetic and magnetic profile control, (iii) initiation and termination control, (iv) control of proximity to stability boundaries, (v) exception handling, (vi) actuator-sharing and control integration, and (vii) core-edge control integration.

Outstanding progress has been made over the last two decades toward developing the control solutions necessary for reactor operations. One of the greatest achievements has been recognizing the importance of a model-based approach to control design. A model-based approach facilitates the

extrapolation of control solutions tested in current devices to future fusion reactors, including ITER. Moreover, it allows for the design of the necessary control architecture and actuator-sharing strategies, embedding the physical dynamics of the controlled system into the control synthesis for both the feed-forward and feedback components. This approach also enables testing control solutions in simulations before implementation and experimentation, maximizing the readiness of control algorithms for first-day operation.

This new focus on model-based control has led to the development of a new class of physics-based predictive codes, emphasizing their use in control applications. Codes such as METIS, COTSIM, and RAPTOR enabled control-oriented simulations and provided, after reduction, the control-oriented models needed to address control problems beyond just equilibrium control, as outlined in the previous paragraph.

Several examples of the progress in plasma control enabled by control-oriented modeling can be highlighted. First, the development and experimental testing of the first profile controllers on JET, DIII-D, EAST, and KSTAR, which tackled not only current density control but also rotation and temperature control. Second, the first actuator-sharing algorithms were developed and tested experimentally on DIII-D, AUG, and TCV. Third, control algorithms to regulate burn conditions were developed while considering the highly non-linear plasma-actuator dynamics. Current efforts on machines like DIII-D focus on emulating burn dynamics to test these burn controllers before ITER's operation.

The major progress in IOS is briefly highlighted above, and the details of these achievements are reported in the following sections; plasma initiation (section 2), low activation phase scenarios (section 3), baseline scenario (section 4), hybrid scenario (section 5), steady-state scenario (section 6), actuators for integrated scenario operation (section 7), and scenario control (section 8). The summary will be provided with remaining outstanding issues in section 9. Plasma initiation, actuators for IOS, and scenario control are extensively covered in this issue, as they have shown significant progress since IPB 2007. In contrast, the low activation phase is relatively short, as ITER has recently changed its plan.

It is noteworthy that there are several common issues that are addressed separately in different sections to provide various insights. Flux pumping is one notable example, discussed in sections 5 and 7. Integrated modeling serves as a common theme throughout all sections and is explored in more detail in section 7.4.

This chapter is also closely linked to other chapters in this special issue. The validation of transport models against experiments is extensively covered in chapter 2, while pedestal model validation is discussed in chapter 3. Disruption avoidance in operation is addressed in chapter 4, and divertor and SOL dynamics for integrated modeling are examined in chapter 5. The impact of energetic particles on transport and stability is analyzed in chapter 7, and the diagnostics required for scenario control are detailed in chapter 8.

Given the chapter's length and complexity, with multiple references to the same topics appearing in different

sections, search tools can be particularly useful for readers. Additionally, a list of acronyms has been included in the [appendix](#).

2. Plasma initiation

Coordinated by Hyun-Tae Kim

2.1. Background

Plasma initiation consists of two phases; the plasma breakdown phase and the plasma burn-through phase. Seed electrons in a vacuum vessel are accelerated by a toroidal induced electric field or a wave and generate an electron avalanche, causing collisional ionizations of pre-filled neutrals. This phase is also known as the Townsend breakdown phase because the required electric field is usually calculated using the Townsend electron avalanche theory. The plasma breakdown phase lasts up to a few percent of the ionization level. At this point, the Coulomb collisions become dominant over the electron-neutral collisions. The next phase is defined as the plasma burn-through phase. Electrons and ions lose energy through atomic reactions with neutral particles such as ionization, radiation, and charge exchange in this phase. The atomic reactions inhibit the increase in T_e . Increasing T_e is essential for the I_p ramp-up, as the plasma resistance R_p is proportional to $T_e^{-1.5}$. This requires high heating power to overcome the energy loss until sufficient ionization is achieved. Once the main fusion fuel and impurities are sufficiently ionized, the electron energy loss is significantly reduced. This allows a further increase in T_e and therefore I_p ramp-up with reduced R_p . A detailed description of the physics of plasma breakdown and plasma burn-through can be found in [31, 32].

Plasma breakdown and burn-through in ITER will be challenging. The inducible toroidal electric field will be limited to 0.3 V m^{-1} due to the superconducting CS and the large major radius ($\sim 6 \text{ m}$). Though this is a much lower value than that typically used in present tokamaks (e.g. 1 V m^{-1} in JET), the vacuum volume in ITER is larger, indicating that the heating power required for plasma burn-through would be higher than in present tokamaks. In addition, ITER has two thick stainless steel vacuum vessel layers (6 cm each). It creates a high toroidal electrical conductivity in the vacuum vessel, which could induce high eddy currents. Eddy currents shorten the effective connection length and reduce the toroidal electric field even further. According to the predictive modeling of ITER plasma burn-through presented in this section, ohmic plasma burn-through would only be possible at very low pre-fill gas pressures ($< 1 \text{ mPa}$). However, operation at such a low pre-fill gas pressure has the risk of plasma breakdown failure and/or runaway electron generation. To avoid runaway electrons, it is necessary to initiate a plasma at a reasonably high pre-fill gas pressure. This means that ECH assistance is required for plasma initiation in ITER. In order to limit the ECH injected power to $\sim 3 \text{ MW}$ (given its low absorption efficiency during the plasma initiation phase and the need to avoid damage to PFCs), it is necessary to carefully optimize the plasma

initiation operating scenario, which includes the coil current waveform, gas puffing, and ECH injection.

In addition, feedback plasma control will not be easy in ITER. The thick vacuum vessel results in a long decay time of the eddy currents, and the position of the PF coils from the plasma is more distant than in other tokamaks. It is also expected that the stray field from the surrounding ferromagnetic structure will affect the magnetic field in the plasma region. Therefore, its control signals should be optimized very carefully. Section 2.3 presents an example of the scenario optimization to address the issues foreseen for ITER plasma initiation. However, the issues are not limited to ITER and could be generic for any tokamak moving towards a large size and superconductivity compared to today's devices.

This section summarizes the progress made in plasma initiation modeling and experiments since the IPB 2007 [33]. Section 2.2 presents the progress in plasma initiation modeling and validation. Section 2.3 describes the ITER issues and the scenario optimization for ITER plasma initiation using plasma initiation modeling. Section 2.4 reviews the progress of plasma initiation experiments in current tokamaks and discusses the physics of runaway electron generation and its observation.

2.2. Plasma initiation modeling and validation

2.2.1. Plasma breakdown modeling. The electric field for Townsend breakdown in a hydrogen or deuterium gas can be derived from the condition: $\alpha L_f > 1$ [34], where α is the expected number of collisional ionizations while an electron travels one meter along a magnetic field line, and L_f is the mean length of open magnetic field lines between the two contact points at the wall (i.e. the effective connection length). $\alpha L_f > 1$ implies that an electron generates more than one electron via collisional ionization before it disappears at the wall. α is the product of the average number of the electron-neutral inelastic collisions in a unit length (i.e. $\frac{1}{\lambda_{en}}$) and the probability of an electron having sufficient energy for ionization (i.e. $\exp\left(-\frac{\delta_{iz}}{\lambda_{en}}\right)$):

$$\alpha = \frac{1}{\lambda_{en}} \times \exp\left(-\frac{\delta_{iz}}{\lambda_{en}}\right) \quad (1)$$

Here, λ_{en} is the mean-free-path of an inelastic collision of an electron with a neutral particle and $\delta_{iz} \equiv \frac{V_l}{E}$ is the required travel length of an electron to be accelerated for a collisional ionization. E is an electric field, and V_l is the ionization energy (i.e. 13.6 eV for hydrogen ionization). λ_{en} is inversely proportional to the gas pressure p . Two constants are defined as $A \equiv \frac{1}{p\lambda_{e-n}}$ and $B \equiv \frac{V_l}{p\lambda_{e-n}}$, which are specific to the type of gas. For hydrogen isotopes, $A = 510 \text{ m}^{-1} \text{ Torr}^{-1}$ and $B = 1.25 \times 10^4 \text{ V m}^{-1} \text{ Torr}^{-1}$. α can be formulated as $\alpha \equiv pA \exp\left(-\frac{B}{E/p}\right)$. Replacing α in $\alpha L_f > 1$ gives the formula to calculate the electric field required for the breakdown of hydrogen isotopes gas in tokamaks:

$$E [\text{V m}^{-1}] \geq \frac{1.25 \times 10^4 p [\text{Torr}]}{\ln(510 \times p [\text{Torr}] \times L_f [\text{m}])}. \quad (2)$$

The Townsend criteria were used in the previous IPB to assess the plasma breakdown condition in ITER [35]. The Townsend criteria have also been used in the DYON code [36] to evaluate the plasma breakdown in individual open field lines, and the operating space of the pre-fill gas pressure and the loop voltage for ohmic plasma breakdown in MAST-U were well reproduced [37].

The Townsend criteria alone may not be sufficient to accurately calculate the time window of the electron avalanche phase. In the KSTAR experiments, the actual electron density growth rate during the electron avalanche phase was found to be much lower than that calculated using the Townsend theory. It has been reported that there could be a perpendicular electric field due to the space charge effects during the avalanche phase [38]. A 2D/3D particle-in-cell simulation with the BREAK code [39, 40] has shown that the space charge electric field could explain the delayed time scale of the electron avalanche process. When the electron density reaches a certain threshold value, the space charge could produce strong self-generated electric fields that cancel the externally induced electric fields. As a result, the effective parallel electric field is significantly reduced, decreasing the collisional ionization rate along the magnetic field lines. In addition, the self-generated electric field could induce the turbulent transport driven by the $E \times B$ drift, increasing the particle loss rate. The BREAK simulation has also shown that there could be turbulent plasma mixing due to the $E \times B$ drift. The $E \times B$ drift in the vertical plane of (R, Z) causes the electrons and ions to flow from upstream to downstream across the magnetic field lines (see figure 3). This implies that the plasma, during the electron avalanche phase, would be mainly in the most downstream region, rather than in the field-null region as conventionally assumed [41].

2.2.2. Plasma burn-through modeling. The loop voltage or ECH power required for successful plasma burn-through at typical operating pre-fill gas pressures (a few mPa) is higher than that required for plasma breakdown. The operating window of the loop voltage, ECH power, and pre-fill gas pressure is therefore mainly determined by the plasma burn-through feasibility. This section reviews the modeling of plasma burn-through based on a 0D plasma transport model. Further details can be found in [24] and the reference therein. The modeling of the ITER plasma initiation operating window and scenario optimization follows in section 2.3.

2.2.2.1. Assumptions, input, and output. Assumptions. Without plasma current, the magnetic field lines are fully open and connected to the FW. As the plasma current increases, the magnetic field configuration evolves from fully open field lines to closed flux surfaces. Heat and particle transport in partially open field lines dominates in the plasma burn-through phase, and any strong spatial profile of T_e and n_e is prevented. Based on this, current plasma burn-through modeling codes such as DYON [32, 42, 43], SCENPLINT [44, 45], and BKD0 [46–48] assume uniform profiles of electron and ion temperatures,

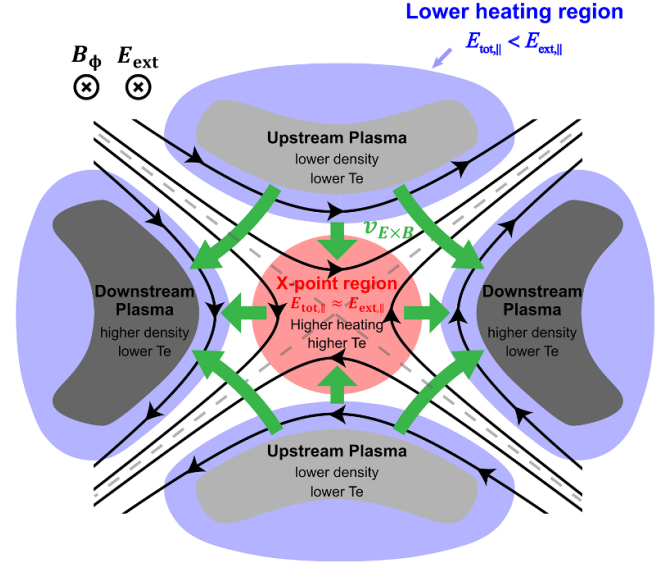


Figure 3. Topology analysis of the quadrupole electromagnetic structure. The magnitude of the total parallel electric field in the X-point region (red) is larger than that in the other regions (blue). The mean $E \times B$ drifts (green) corresponding to the quadrupole magnetic structure yield two inflows and two outflows around the X-point. The plasma density of the downstream region (dark gray) is higher than that of the upstream region (pale gray). Reproduced from [40]. CC BY 4.0.

electron and ion densities, and current densities, i.e. 0D plasma transport model.

In the plasma breakdown phase, collisions between electrons and neutral particles dominate. At a few percentages of the degree of ionization, Coulomb collisions become much more frequent than electron-neutral particle collisions, and the Spitzer resistivity is valid to calculate the plasma resistance. Based on this, the plasma burn-through modeling codes start from a few % of ionization.

A constant plasma shape and position was assumed by estimating the size of the magnetic field null and the ECH resonance position, as was done in the IPB 2007 [33]. Although the plasma volume model has been developed in the recent years [36], it was not used in the ITER plasma initiation modeling in section 2.3.

Sputtering at the FW by incident fuel or impurity ions is the main source of the impurities. For a beryllium wall in ITER, physical sputtering is assumed to be dominant, and the sputtering yield is calculated using the Bohdansky formula, which calculates the sputtering yield as a function of the incident ion energy [32].

The pre-filled fuel gas is assumed to be hydrogen isotope molecules (i.e. H_2 or D_2) at room temperature (i.e. 0.026 eV), and the initial hydrogen isotope atom density is calculated as:

$$n_H(0) [\text{m}^{-3}] = \frac{2 \times p_0 [\text{Pa}]}{0.026 \times 1.6 \times 10^{-19}} = 4.8 \times 10^{20} \times p_0 [\text{Pa}] \quad (3)$$

In the modeling, fully dissociated D_2 molecules are assumed, since the dissociation energy is much smaller than the electron energy loss for ionization and radiation of deuterium atoms.

Input data to modeling. The followings are input data of device parameters or operating scenarios.

- Pre-filled gas pressure and external gas fueling.
- Vacuum space volume.
- Loop voltage at the plasma region.
- ECH hardware setting: Injection power, frequency, injection angles, number of harmonics, fraction of O or X modes, and the number of plasma and EC beam interactions
- Toroidal magnetic fields.
- Stray magnetic fields at the plasma region from coils and vessel eddy currents.

Output data from modeling. The followings are output data from simulations by the plasma burn-through modeling.

- Time evolution of the plasma parameters: $I_p(t)$, $T_e(t)$, $T_i(t)$, $n_e(t)$, $n_D^0(t)$, $n_D^{1+}(t)$, and impurity densities $n_{\text{imp}}^{z+}(t)$, e.g. beryllium, carbon, oxygen, nitrogen, helium, and neon.
- Energy balance: ohmic heating power, ECH heating power, ionization electron energy loss, radiated electron energy loss rate, equilibration heat exchange rate, charge exchange ion energy loss rate, transport energy loss rate.
- Particle balance: ionization rate, recombination rate, and charge exchange rate of main fuel gas and impurities in each charge state, transport particle loss rate
- Synthetic data: line emission data (e.g. D_α , C^{2+} , and O^{3+}), etc.

2.2.2.2. Model improvement. With the above assumed plasma parameters and the input data, the plasma burn-through modeling codes solve the circuit equation of a plasma current and a major eddy current (i.e. two ring model), and the time-dependent 0D transport equation (i.e. energy balance equations for electrons and ions, and particle balance equations for fuel ions and impurities). Full details of the plasma burn-through models can be found in [32, 42, 49]. The following description will focus only on the new physics models that have been developed since the publication of the IPB in 2007 [33]. The full electromagnetic plasma burn-through model, taking into account the eddy currents in all the passive structures, has been recently developed in [36] and validated in [37], but it is not included in the discussion in this section.

Confinement time model. In the previous IPB in 1999 [35] and in 2007 [33], the effective connection length was used only to calculate the required toroidal electric field for Townsend breakdown, and the energy confinement time was estimated in plasma burn-through modeling with a simple constant [49] or scaling law [33]. In [42], the following improved model for the effective connection length L_f was proposed, and used to calculate the particle confinement time,

$$L_f = 0.25a_{\text{eff}} \frac{B_\phi}{B_\perp} \exp\left(\frac{I_p}{I_{\text{ref}}}\right) \quad (4)$$

Here, a_{eff} is the effective minor radius ($\sim$ minor radius of the vacuum chamber). B_ϕ and $B_\perp$ are the toroidal field and stray field, respectively. The exponential term has been added to describe the evolving L_f which approaches infinity as closed flux surfaces are formed when I_p is sufficiently higher than the vessel eddy currents resulting in the stray field. I_{ref} is the reference plasma current when the field lines are fully closed, and the values depend on the device (e.g. $I_{\text{ref}} = 100$ kA in JET).

The particle confinement time for parallel and perpendicular transports are calculated, respectively,

$$\tau_{\parallel} = \frac{L_f}{C_s} \quad (5)$$

$$\tau_{\perp} = \frac{a_{\text{eff}}}{v_{\text{Bohm}}} \quad (6)$$

where sound speed $C_s = \sqrt{\frac{T_e + T_i}{m_i}}$ and $v_{\text{Bohm}} = \frac{1}{8} \frac{T_e [\text{eV}]}{a_{\text{eff}} [\text{m}] B_\phi [T]}$. Then, the parallel and perpendicular confinement times are combined to yield the particle confinement time

$$\frac{1}{\tau_p} = \frac{1}{\tau_{\parallel}} + \frac{1}{\tau_{\perp}}. \quad (7)$$

Since the plasma initiation modeling assumes uniform temperature and density profiles, the transport energy loss results only from particle convection. Therefore, the energy confinement time is assumed to be equal to the particle confinement time, $\tau_E = \tau_p$, and the global transport energy loss is calculated as

$$P_{\text{econ}} = 1.5 \frac{n_e T_e}{\tau_p} \quad (8)$$

$$P_{\text{icon}} = 1.5 \frac{n_i T_i}{\tau_p}. \quad (9)$$

Impurity density evolution model. In the previous plasma burn-through simulation [33], the impurity content was prescribed as a function of time. Since the impurity content is determined by the PWI, modeling of the impurity content as a function of plasma parameters is required. [42] has developed a new model that calculates the evolving impurity densities in each charge state by solving the impurity ion particle balances and modeling the PWI. In the new model the impurity ion outflux to the wall $\Gamma_{\text{I,out}}^{Z+}$ and the impurity neutral influx from the wall $\Gamma_{\text{I,in}}^0$ are calculated as

$$\Gamma_{\text{I,out}}^{Z+} = V_p \frac{n_I^{Z+}}{\tau_p} \quad (10)$$

$$\Gamma_{\text{I,in}}^0 = V_p \sum_I \sum_Z \frac{Y_A^I n_I^{Z+}}{\tau_p} \quad (11)$$

where Y_A^I is the recycling or sputtering yield, the superscript I is the recycled or sputtered impurity neutral species, and A is the incident ion resulting in the recycling or sputtering at

the wall. The new model was used in the ITER modeling in section 2.3.

ECH model. In the previous burn-through simulation [33, 49], a few MW constant absorbed ECH power was prescribed. However, the ECH power absorption during the burn-through phase is inefficient as T_e and n_e are low, so it is significantly lower than the injected ECH power. To address this uncertainty, the ECH absorbed power is now calculated as a function of plasma parameters and hardware settings (e.g. injected power, frequency, injection angle, # of harmonics, etc.) in current plasma modeling codes [24]. For example, the ECH model in DYON is

$$P_{\text{ECH}} = P_{\text{inj}} \{1 - (1 - \varepsilon)^{n_{\text{int}}}\} \quad (12)$$

$$\varepsilon = (1 - f_O e^{-\eta_O} - f_X e^{-\eta_X}) \quad (13)$$

where ε is the absorption efficiency for the single path of the EC beam, n_{int} is the number of EC beams and plasma interactions (i.e. $n_{\text{int}} = 2$ for double path) and f_O and f_X are the fraction of O-mode and X-mode, respectively. η_O and η_X are optical thickness of O-mode and X-mode, respectively [50].

2.2.2.3. Code benchmark. As a JA in the ITPA IOS TG, the mainstream plasma burn-through modeling codes used in several devices (SCENPLINT in 2007 IPB, DYON in JET, MAST, VEST, and KSTAR, and BKD0 in JT60-SA) were compared for the first time [24]. Prior to any reconciliation work, all three codes consistently predicted that the pre-filled gas pressure required for ohmic plasma burn-through in ITER will be lower than 1 mPa. A detailed source code comparison allowed the developers to identify the minor differences between the codes. After reconciliation work to remove the identified differences in the codes, the three codes produced identical simulation results and confirmed the consistency.

In this code benchmark, it was found that the evolution of the plasma volume can significantly affect the evolution of the plasma parameters and therefore a consistent physics model needs to be developed to improve the prediction capability. This motivated the development of a full circuit equation solver and Townsend breakdown modeling in DYON, which allows the modeling of the evolving plasma volume during the plasma initiation phase [36, 37]. The plasma volume model successfully reproduced data from the MAST and MAST-U experiments.

2.2.2.4. Model validation against experiments. The DYON simulation with the new models for confinement time and impurity evolution was compared with JET ohmic burn-through discharges with carbon wall [42] and beryllium wall [32]. The line emission data of deuterium and impurities were well reproduced by DYON. Since the synthetic line emission data is a function of T_e and n_e , the good temporal agreement with the measured data during the burn-through phase (i.e. 0–100 ms) implies that the simulated T_e and n_e could be in agreement with the experiment.

Following the lessons learned from the code benchmark [24], the full circuit equation system describing all active and

passive structures has been integrated in DYON. It allows the prediction of plasma initiation success or failure in individual discharges, using the control room data such as the CS and PF current waveform and the measured pre-fill gas pressure. It allows the validation of the prediction capability of the full electromagnetic DYON with dedicated MAST-U plasma initiation experiments. Figure 4 shows that the operation space of the pre-fill gas pressure and the loop voltage for ohmic plasma initiation in experiments are correctly reproduced in DYON. A few cases of prediction failures are due to the input data error bars in the modeling as they are found near the boundary of the operation space [37].

The DYON simulation with the ECH model was compared with the KSTAR experiments, which have the similar ECH configuration as the ITER first plasma; 170 GHz frequency, second harmonics, 100% X-mode [50]. The carbon sputtering yield was prescribed to reproduce the ohmic burn-through experimental data, and the same PWI model was used in the EC-assisted burn-through simulation. The EC-assisted burn-through simulation reproduced the synthetic interferometry data and the line emission data of deuterium, carbon, and oxygen, showing good agreement with the experimental measurement. This implies that the absorbed ECH power during the plasma burn-through phase was calculated with good accuracy. The calculated efficiency of the ECH power absorption is less than 10% during the burn-through phase.

2.2.3. Codes used for design and simulation of ITER plasma initiation scenarios. The following three codes have been used for the design and simulation of ITER plasma initiation scenarios:

- (i) SCENPLINT: plasma burn-through simulation code based on 0D plasma transport model.
- (ii) TRANSMAX: electromagnetic scenario optimization code used to design the ITER circuit system operation scenarios. The plasma resistance is calculated with SCENPLINT.
- (iii) DINA: time-dependent free-boundary equilibrium simulation code used for simulation of plasma initiation scenarios designed with TRANSMAX.

SCENPLINT. SCENPLINT [44, 45] is a 0D plasma transport code used to calculate electron and ion temperatures, neutral atom, electron and ion densities, plasma resistance and current during the plasma burn-through phase for given plasma geometry parameters (i.e. major radius and minor radius), and a prescribed waveform of voltage, $V_{\text{ext}}(t)$, induced on the plasma by variation of currents in the CS and PF coils and eddy currents. The SCENPLINT code is similar to DYON and BKD0, and the code benchmark confirmed their consistency [24].

The present version of SCENPLINT includes:

- The models of physical sputtering of *Fe* (for first plasma operation with temporary steel limiter) and *Be* (for other phases of operation) of the PFCs, allowing to obtain the impurity content in plasma

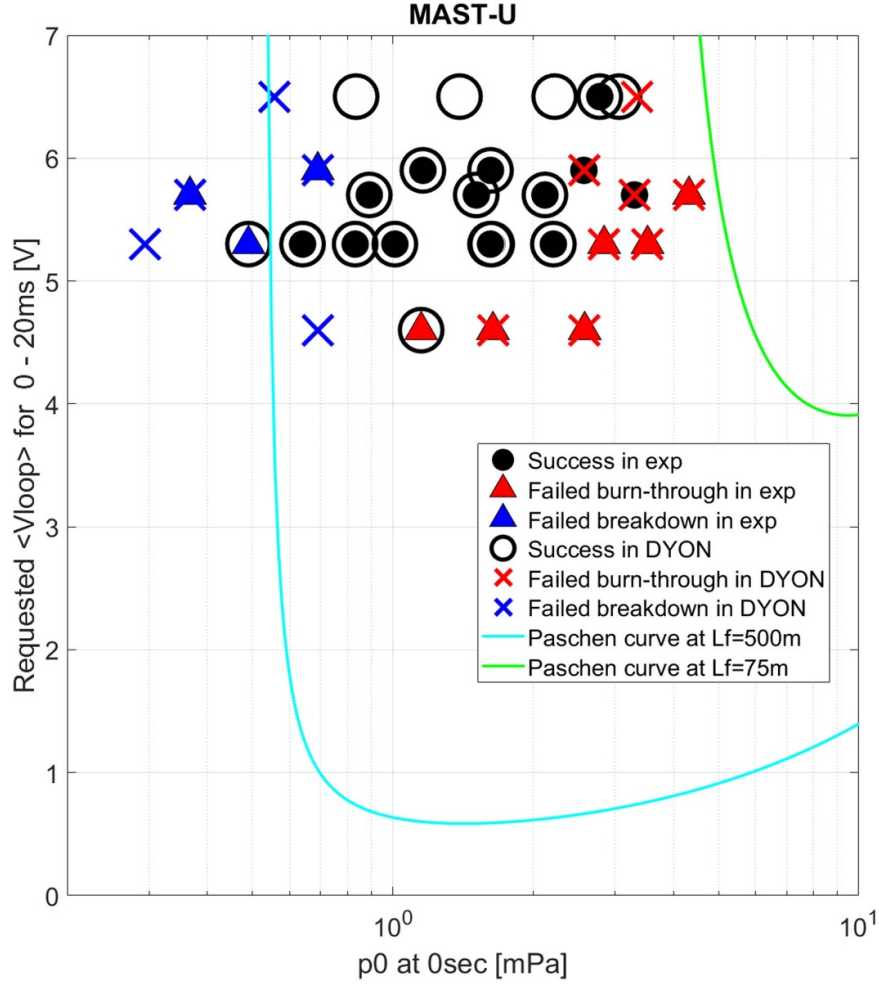


Figure 4. Comparison of the operation space for ohmic plasma initiation in MAST-U and the prediction results with the full electromagnetic DYON. Reproduced from [37]. © 2024 The Author(s). Published by IOP Publishing Ltd on behalf of the IAEA. CC BY 4.0.

- The model for runaway electrons generation and evolution
- The model of ECH power absorption at the main and second harmonics
- The plasma resistance with effects of electron-neutrals collisions.

TRANSMAK. Scenarios of CS and PF system operation during the plasma initiation are designed using the TRANSMAK code [44]. The aim is to keep the stray magnetic field value lower than a given value (typically 2 mT) in the breakdown region (typically $R \approx 5.7$ m, $Z \approx 0$, $a \approx 1.6$ m) at the time of plasma breakdown, defined when $E = 0.3$ V m⁻¹. After the plasma breakdown, the operating scenario of the CS and PF system is optimized to achieve the following goals:

- To ensure a sufficiently long effective connection length, the absolute value of the radial component of the magnetic field B_r should be less than about ± 2 mT
- To ensure the radial force balance of the plasma current, the vertical component of the external magnetic field B_z should be as close as possible to the ‘Shafranov’ field $B_{z,S}$, expressed with plasma parameters R , a , I_p , β_p defining the value of [51]:

$$B_{z,S} = -\frac{\mu_0 I_p}{4\pi R} \left[\log \left(\frac{8R}{a} \right) + \beta_p + \frac{l_i}{2} - 1.5 \right] \quad (14)$$

where the evolving I_p and β_p are calculated with the coupled SCENPLINT code.

- To ensure the VS and formation of plasma with circular cross section having major and minor radii $R \approx 5.9$ m and $a \approx 1.8$ m, the decay index n of B_z , averaged over the prescribed plasma boundary, should be somewhere between 0 and 0.7, i.e.

$$0 < n \left(= -\frac{R}{\langle B_z \rangle} \left\langle \frac{dB_z}{dr} \right\rangle \right) < 0.7 \quad (15)$$

where $\langle \dots \rangle$ means averaging over the plasma boundary prescribed with the minor radius a .

- To minimize the total power required from the power supply (i.e. electric grid).

The outputs of the TRANSMAK calculation are: (i) 11 values of the initial currents in the CS and PF coils, (ii) 7 resistances of the switching network units (SNUs), and (iii) 11 pre-programmed waveforms of the voltages of the CS and PF coil converters.

In order to take into account the stray magnetic fields resulting from the 3D vacuum vessel structure and the ferromagnetic materials in the ITER building, TRANSMAX includes additional 2D conducting elements. The operating scenario of the CS and PF circuit system is optimized in TRANSMAX with the constraint that all engineering parameters (coil currents, voltages, magnetic field on conductors, and forces) are within engineering limits.

DINA. Scenarios of CS and PF system operation during the plasma initiation obtained with the TRANSMAX code are used in the DINA code simulations. The DINA code [52] consists of:

- 2D elements of the Tokamak Electromagnetic Model (conducting structures, ferromagnetic elements of the building, etc similar to that in the TRANSMAX code),
- 2D plasma free-boundary equilibrium solver implementing the feedback control of the plasma current, position and shape taking into account eddy currents in the vacuum vessel, and models of the coil power supplies,
- 0D plasma transport model (similar to that in SCENPLINT) at the beginning of plasma initiation and 1D plasma transport model for the later phases, 1D model describing the diffusion of the poloidal magnetic flux (evolution of I_i).

In addition, DINA simulations took into account numerous engineering constraints on the coils and their power supplies.

2.3. Plasma initiation in ITER

2.3.1. Circuit system. The ITER PF system consists of a segmented CS and six outer superconducting PF coils, as shown in figure 5. All PF coils and all CS modules, except for the two central modules (CS1U and CS1L, connected in series), have independent power supplies. The currents in all 11 circuits of the CS and PF coils are varied by AC/DC converters. Each converter produces a no-load voltage ± 1.35 kV [53]. The CS circuits (except for CS1), PF1 and PF6 have two converters; there are four converters in the CS1 circuit, and three converters in the PF2, PF3, PF4, and PF5 circuits. Table 2 shows some CS and PF coil parameters; number of turns (N), maximum on-load voltage of converters used in simulations of plasma magnetic control ($V_{\max}$), maximum current in one turn ($I_{\max}$), maximum value of magnetic field on the coil conductor at 4.3 K (design limit). Dependence of the maximum current in one turn of each coil on the maximum magnetic field on the coil is obtained by the linear interpolation or extrapolation between or beyond pairs of values given in the table. Operational space (I , B) is below this lines.

In order to increase the loop voltage during the plasma initiation phase, high voltages are generated by the SNUs in five CS circuits and in the PF1 and PF6 circuits. At $t = 0$ s (the start of the CS current discharge) the currents in these coils are interrupted and then transferred to a resistor bank, inducing a maximum voltage of 8.5 kV across the coils, except for the CS1 coils which have a maximum voltage of 6 kV in each coil [53]. The SNU of all circuits, except CS1, which

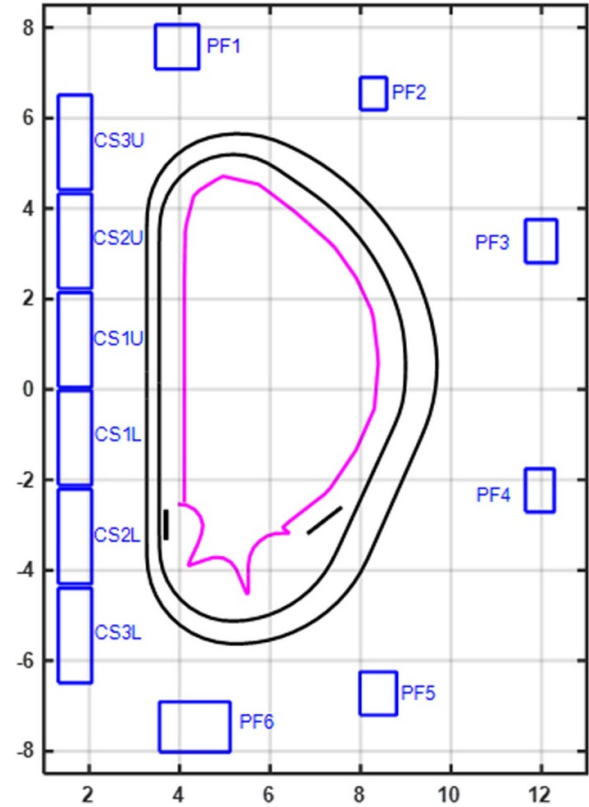


Figure 5. ITER CS and PF coils and their numbering (in blue), 2D model of vacuum vessel including toroidally continuous conducting triangular support of blanket modules #18 and divertor inboard rail (in black), and plasma facing line of the first wall and divertor (in magenta).

uses only one set of the resistors (different in different coils), are switched off at $t \approx 3$ s. In the CS1 SNU, at $t \approx 3$ s the initial set of resistors is switched to another set with approximately half the resistance, and then CS1 is switched off at $t \approx 13$ s.

At the beginning of the CS current swing, the maximum magnetic flux (116 Wb) is generated. Except for CS3U, the current in the CS modules is limited by the magnetic field on the conductor, which is 13 T, and the maximum value available is about 40 kA. The maximum current in CS3U is limited by the combination of vertical forces acting on the CS modules at the beginning of the CS current swing [54].

2.3.2. Electromagnetic structures. The design (TRANSMAX) and the simulation (DINA) of the plasma initiation were performed using the main elements of the Tokamak Electromagnetic Model (figure 6):

- 2D model of conducting structures—the vacuum vessel, including the toroidal continuous conducting triangular support of blanket modules #18 (vertical black object in figure 6(a)) and the inboard divertor rail (inclined black object in figure 6(a)), and the cryostat (with cryostat thermal shield).
- 2D model of the stray magnetic field generated by magnetized ferromagnetic elements of the tokamak building

Table 2. ITER CS and PF coil parameters; number of turns (N), maximum on-load voltage of convertors ($V_{\max}$), maximum current in one turn ($I_{\max}$), maximum value of magnetic field on the coil conductor at 4.3 K.

Coil	N	$V_{\max}$ (kV)	$I_{\max}$ (kA)	$B_{\max}$ (T)
CS3U	554	2.1	45	12.3
			40	13.0
CS2U	554	2.1	45	12.3
			40	13.0
CS1U	554	2.1	45	12.3
			40	13.0
CS2L	554	2.1	45	12.3
			40	13.0
CS3L	554	2.1	45	12.3
			40	13.0
PF1	248.6	2.1	48	6.4
			41	6.5
PF2	115.2	3.15	55	4.8
			50	5.0
PF3	185.9	3.15	55	4.8
			50	5.0
PF4	169.9	3.15	55	4.8
			50	5.0
PF5	216.8	3.15	52	5.7
			33	6.0
PF6	459.4	2.1	48	6.4
			41	6.5

(e.g. rebar in concrete) modeled as two virtual coils with given currents.

The 3D objects such as ports in the vacuum vessel (see figure 6(b)) were taken into account by increasing the resistance in 2D conducting models (see figure 6(a)). The axisymmetric component of the magnetic field in the plasma region resulting from the 3D ferromagnetic structures (figure 6(c)) were also modeled with the virtual coils.

Other magnetic field sources make a small contribution to the toroidally averaged value of the stray magnetic field at plasma initiation. For example, the ferromagnetic inserts (which reduce TF ripple), located between the inner and the outer walls of the vacuum vessel in the poloidal plane of the TF coils, produce a vertical magnetic field (opposite to the ‘Shafranov’ field) lower than 0.2 mT in the plasma region. TF coil busbars produce a vertical magnetic field (unidirectional to ‘Shafranov’ field) of less than 0.1 mT in the plasma region.

2.3.3. Plasma initiation scenario. In the IPB 2007 [33], outboard plasma initiation was planned, and the plasma volume was limited by the outboard limiter. To avoid a temporary outboard limiter and an additional gyrotron for

plasma initiation, the operating scenario was changed to inboard plasma initiation. In the present plan, the plasma with a minor radius $a \approx 1.6$ m touches the FW center column at $R \approx 5.7$ m in the horizontal plane i.e. $Z \approx 0$. Plasma breakdown is assumed when the toroidal electric field, induced by the CS current discharge, increases to $E_t \approx 0.3$ V m⁻¹ at $R \approx 5.7$ m (loop voltage of 10.7 V). Due to the low toroidal resistance of the vacuum vessel ($\approx 8 \mu\Omega$), the penetration of E_t into the vacuum vessel takes about 1.1 s.

At the plasma breakdown (when $E_t \approx 0.3$ V m⁻¹), the stray poloidal magnetic field should be less than 2 mT in the breakdown region ($R \approx 5.7$ m, $Z \approx 0$, $a \approx 1.6$ m). After plasma breakdown, the magnetic field should produce plasma equilibrium with $R \approx 5.7$ m, $Z \approx 0$, $a \approx 1.6$ m, $k \approx 1$ (i.e. circular shape). Due to the large volume of the pre-fill gas (≈ 1000 m³), the gas molecular pressure should be less than 0.5 mPa and ECH assistance is required for pre-ionization and for the further reliable burn-through.

2.3.4. Plasma initiation modeling

2.3.4.1. Operating window for plasma burn-through. The electron heating power required for plasma burn-through increases with the pre-filled neutral density. DYON predicts that the ohmic plasma burn-through at a 12 V peak loop voltage and a 5.3 T toroidal field at $R = 5.65$ m in ITER would require less than 1 mPa ($\sim 7.2 \times 10^{-6}$ Torr) [43]. Operation at a 2.65 T toroidal field, which is foreseen in the ITER first plasma, has also been consistently predicted by DYON, SCENPLINT, and BKD0 (see figure 7 [24]). Ohmic plasma burn-through would not be possible at the pre-fill gas pressures above 1 mPa in ITER, and that ECH heating would be required.

Since the prediction made in figure 7 was only for pure hydrogen burn-through, i.e. no impurities, the required pre-filled gas pressure could be even lower than 1 mPa with impurities. Low levels of carbon or oxygen ($\sim 1\%$ – 2%) can significantly increase the radiated energy loss. The simulations indicate that ITER should keep the concentration of low-Z impurities below 0.5% at plasma initiation [32].

DYON predicted that the required ECH power absorption is about 3–4 MW at 6.6 mPa ($\sim 5 \times 10^{-5}$ Torr), which is a common pre-filled gas pressure in current tokamaks, e.g. JET [43]. According to the calculation with the ECH model described above, the efficiency of the 2nd harmonic X-mode single-pass ECH power absorption in ITER is likely to be less than 10% due to the low T_e and n_e during the plasma burn-through phase [24], implying that much higher ECH injected power would be required for successful plasma burn-through at 6.6 mPa. BKD0 predicted that the 1st harmonic O-mode multi-pass ECH absorption is higher, extending the operational pre-fill gas pressure by 0.8 mPa MW⁻¹ (i.e. successful plasma burn-through at 6.6 mPa with 7–8 MW injected power) [55].

2.3.4.2. Modeling of plasma initiation scenario. This section presents an example of an ITER plasma initiation scenario with the following assumptions;

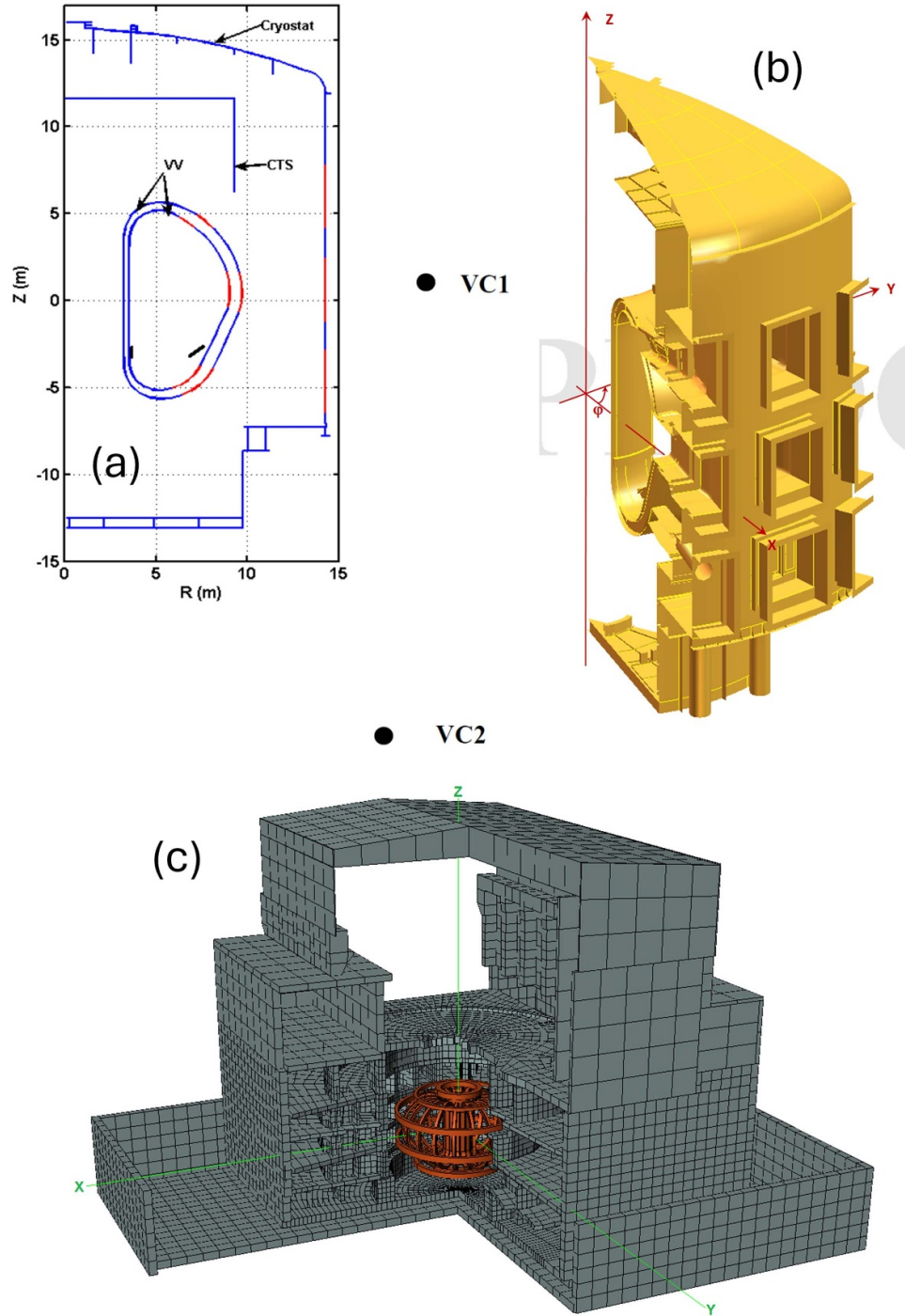


Figure 6. Main elements of Tokamak Electromagnetic Model: (a) 2D models of the vacuum vessel (VV), including toroidally continuous conducting triangular support of blanket modules #18 (vertical black object) and divertor inboard rail (inclined black object), cryostat and cryostat thermal shield (CTS), and virtual coils (VC) with currents reproducing axisymmetric component of magnetic field from ferromagnetic elements of the tokamak buildings, (b) 3D model of the VV, cryostat and CTS, (c) 3D model of ferromagnetic elements of the tokamak building.

- starting from a fully charged CS
- nominal toroidal magnetic field (5.3 T at 6.2 m)
- 2 MW of ECH-assist
- 1000 m³ of vacuum volume
- full beryllium FW.

In this scenario, at the time of plasma breakdown (i.e. 1.1 s after the start of the CS current discharge), eddy currents induced in the vacuum vessel generate a magnetic field with $B_z \approx -28$ mT, $B_r \approx -4$ mT, eddy currents induced in the cryostat generate a magnetic field with $B_z \approx -15$ mT,

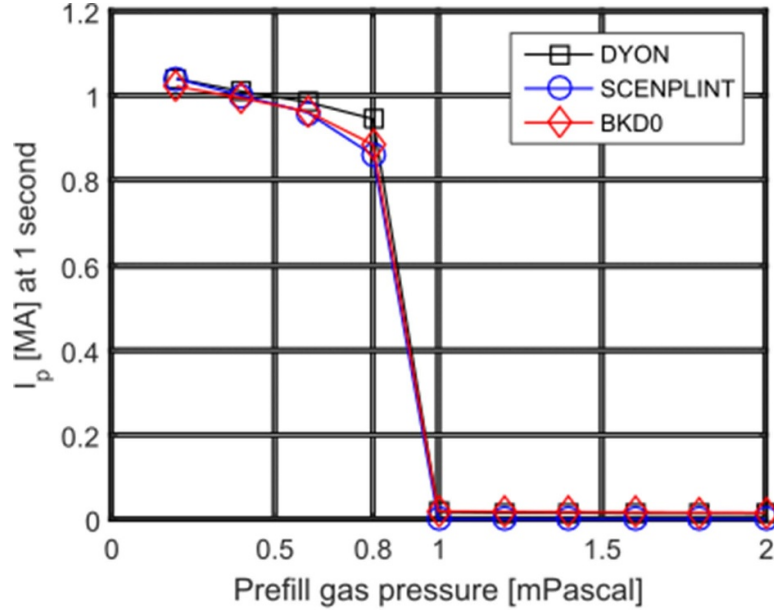


Figure 7. Predicted required pre-filled gas pressure for ohmic plasma burn-through in ITER (assuming minor radius of 1.6 m and major radius of 5.65 m). Reproduced from [24]. © EURATOM 2020. All rights reserved.

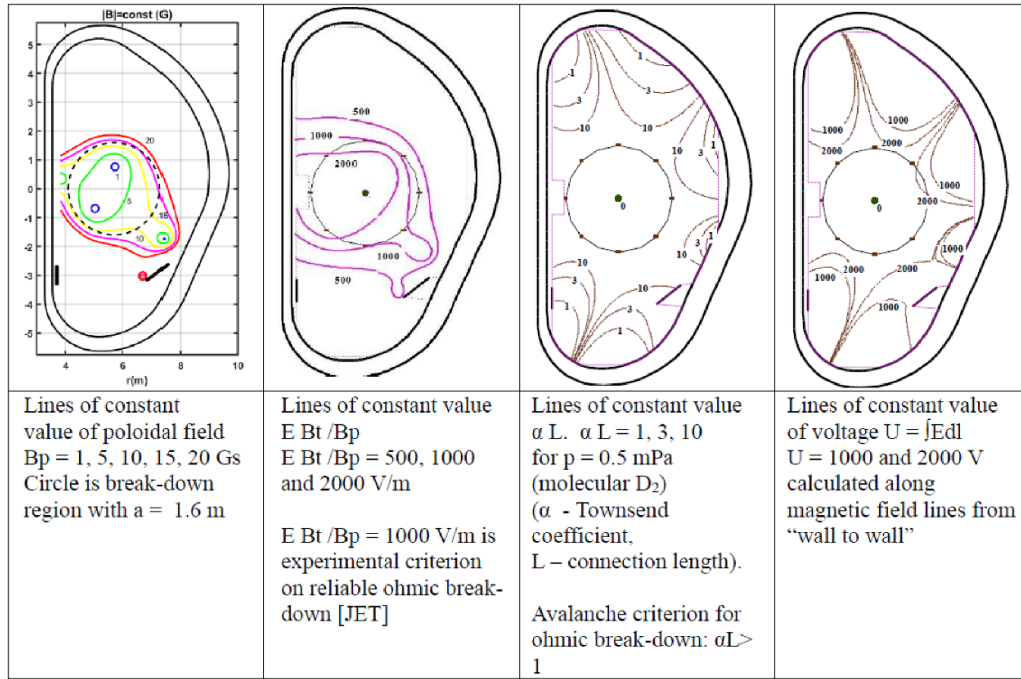


Figure 8. Various characteristics of the magnetic configuration at the breakdown (TRANSMAX).

$B_r \approx -2$ mT, ferromagnetic elements of the building generate magnetic field with $B_z \approx -11$ mT, $B_r \approx -1$ mT (the 'Shafranov' field is positive) in the plasma breakdown region. Figure 8 shows various characteristics of the magnetic configuration at the time of plasma breakdown ($t = 1.1$ s) from TRANSMAX.

The results of the plasma initiation simulations carried out using the DINA code are shown in figures 9–12. Figure 9 shows the plasma current, the voltages in the circuits of the CS and PF coils generated by the converters, and the currents

in the circuits of the CS and PF coils. The voltages in all circuits are within the design limits given in table 2. The initial currents in all CS coils are around 40 kA, except for CS3U, where they are limited by a combination of vertical forces acting on the CS modules at the start of the CS current discharge [54].

Figure 10 shows the maximum value of the magnetic field on the CS and PF conductors, the plasma current (I_p) and the total power requested from the electric grid (P_{grid}). The magnetic fields on all conductors are within the design limits

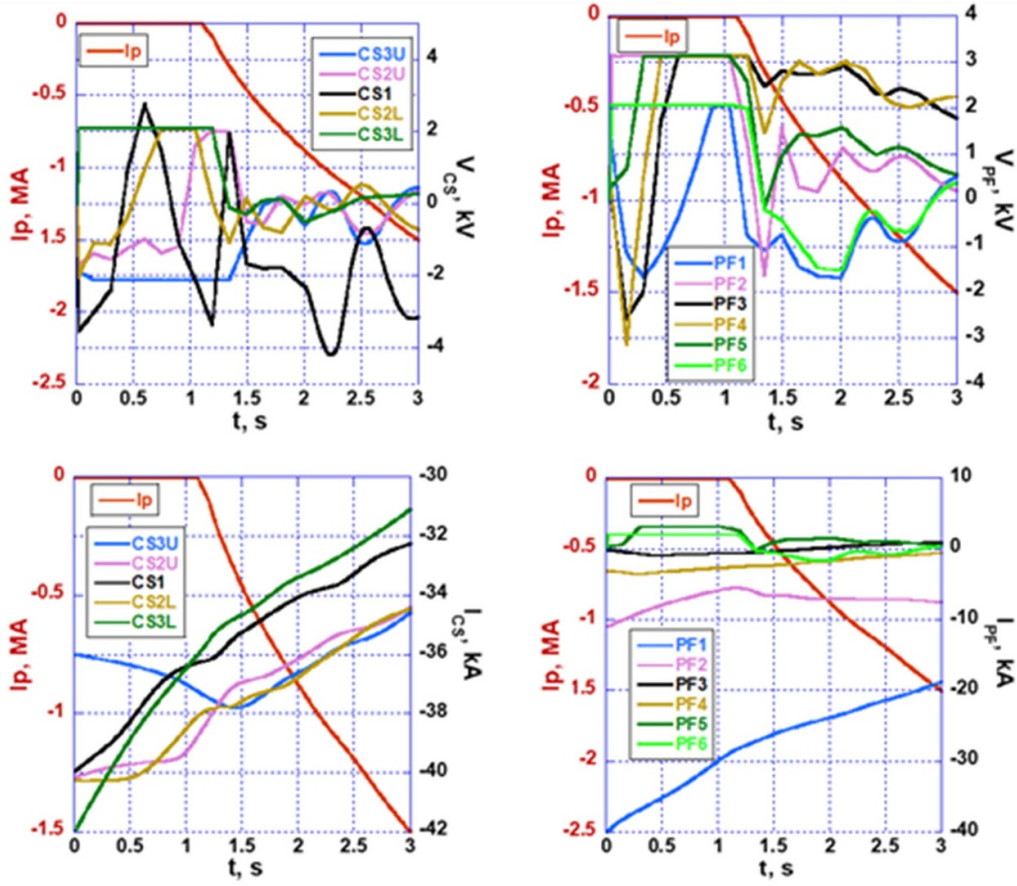


Figure 9. Plasma current (I_p), voltages in the circuits of CS and PF coils produced by converters (V_{CS} and V_{PF}), currents in the circuits of CS and PF coils (I_{CS} and I_{PF}). Breakdown at $t = 1.1$ s.

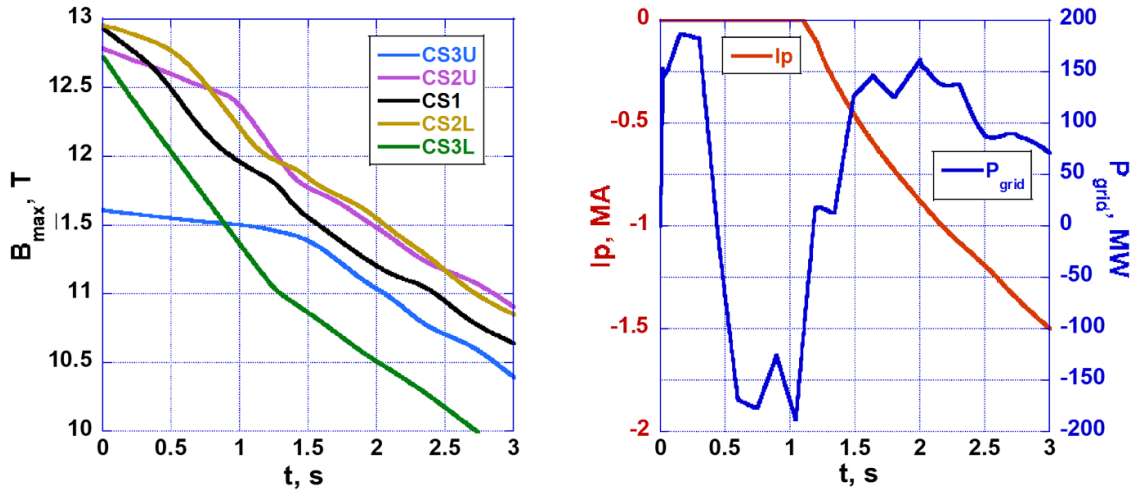


Figure 10. Maximum value of magnetic field on the CS and PF conductors, plasma current (I_p) and total power requested from the electric grid (P_{grid}). Breakdown at $t = 1.1$ s.

(table 2). The total power requested is within ± 200 MW. The limit on total power requested from the grid is ± 500 MW.

Figure 11 shows the plasma current (I_p), minor radius (a), elongation (k), resistive loop voltage (V_{res}), energy confinement time (τ_E), averaged over the plasma cross-section poloidal magnetic flux, Ψ_{ext} , generated by all external currents

(CS and PF coils, and eddy currents), $\langle \Psi_{ext} \rangle = \int \Psi_{ext} j_p dS_p / I_p$, and the averaged poloidal magnetic flux from the coils only, ($\langle \Psi_{coils} \rangle$). At the time of plasma breakdown, the magnetic flux generated by the external currents is $\Psi_{ext} \approx -109$ Wb. The eddy currents induced in the conducting structures generate about -6 Wb of this flux.

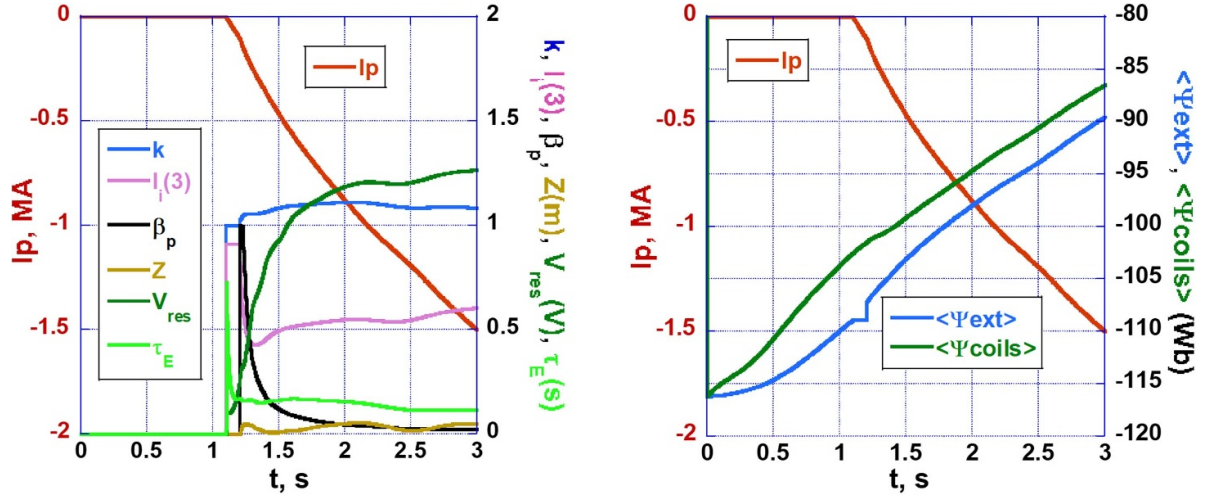


Figure 11. Plasma current (I_p), elongation (k), internal inductance (l_i), poloidal beta (β_p), Z-coordinate of the plasma current center, resistive loop voltage (V_{res}), energy confinement time (τ_E), averaged over the plasma cross-section poloidal magnetic flux, Ψ_{ext} , produced by all external currents (CS and PF coils, and eddy currents) $\langle \Psi_{ext} \rangle = \int \Psi_{ext} j_p dS_p / I_p$, and the averaged poloidal magnetic flux from the coils only, ($\langle \Psi_{coils} \rangle$). Breakdown at $t = 1.1$ s.

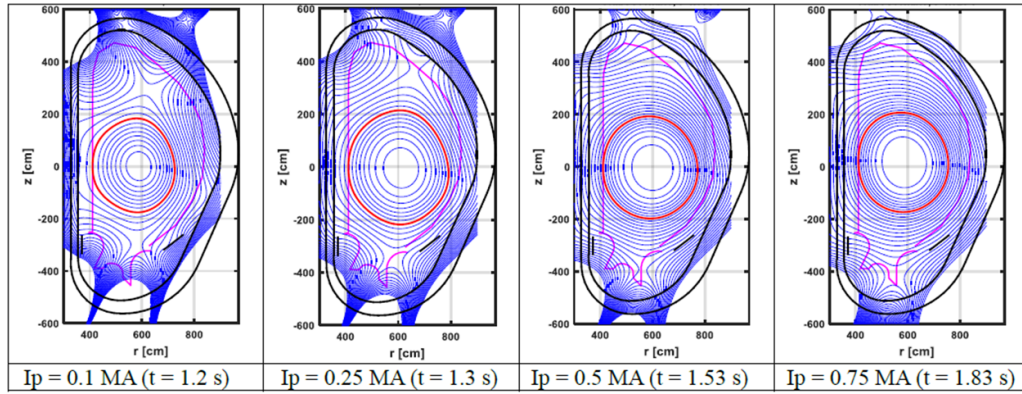


Figure 12. Magnetic configurations at $I_p = 0.1, 0.25, 0.5, 0.75$ MA.

Figure 12 shows magnetic configurations at $I_p = 0.1, 0.25, 0.5, 0.75$ MA.

2.4. Experiments in support of ITER

2.4.1. Inductive plasma initiation at low E_t . The feasibility of the inductive plasma breakdown with $E_t \sim 0.3 \text{ V m}^{-1}$ in ITER is subject to the capability of minimizing the stray poloidal magnetic field, which is to achieve long connection lengths of open field lines in the vacuum chamber as aforementioned. Several device-specific or scenario-specific metrics and reduced models have been used to design new tokamaks or to improve the interpretation of plasma initiation results [34, 56]. One of the most commonly used metric is $EB_t/B_p > 1000 \text{ V m}^{-1}$ [57], which is an empirically derived threshold for reliable unassisted plasma initiation at JET and is also shown for TRANMAK calculations for ITER start-up in figure 8. The connection lengths are typically longer in a larger devices,

and it indicates that plasma breakdown might be easier in large devices. However, in the experiments in the current devices, it was found that the small devices with many PF coils located closely to the breakdown region can often achieve a more complete reduction of the stray poloidal magnetic field. JET [58], DIII-D [34, 59], KSTAR [60], and Tore Supra [61] have also demonstrated the inductive plasma breakdown with electric field values at or below the ITER design value (0.3 V m^{-1}) without any strong scaling of the minimum E field with device size [13].

The successful progression of the plasma initiation through the plasma burn-through phase is sensitive to the impurity content of the plasma. Encouragingly, successful inductive plasma burn-through with E_t below 0.3 V m^{-1} has been demonstrated in the current devices using various FW materials and wall conditioning methods. Tokamaks with metal FWs have demonstrated reliable plasma burn-through without inter-pulse glow discharge cleaning (GDC), even following plasma disruptions [13].

2.4.2. ECH-assisted plasma initiation

2.4.2.1. EC wave for plasma initiation. ECH extends the operational margin for successful plasma initiation at low E_t [62] and is often an important tool for initiating plasmas prior to optimizing the operating scenario and fully conditioning the PFCs [63]. ECH-assisted plasma initiation using toroidal injection of the EC beam has demonstrated successful plasma initiation below $E_t = 0.3 \text{ V m}^{-1}$ in AUG (X2), DIII-D (X2), FTU (O1), KSTAR (X2), Tore Supra (O1, X2), HL-2A (O1), and TJ-II (X2) [13, 64]. However, in some cases, the required ECH power was almost a factor of 2 higher than for perpendicular EC injection [13]. ECH can either create a low-temperature plasma prior to the induction of a loop voltage (i.e. pre-ionization), or in the presence of a loop voltage it can relax the requirements for the Townsend breakdown by additionally accelerating electrons that are otherwise subcritical in the energy distribution [49, 65, 66]. In the subsequent plasma burn-through phase and the I_p ramp-up phase, ECH can facilitate the overcoming of the radiation barrier and drive the plasma current. The use of ECH in ITER will extend the operating window for successful plasma initiation and could allow the development of scenarios with low E_t or with a finite vertical equilibrium field.

The ITER EC wave antennas are designed to operate at 170 GHz, with an injection angle of 20 degrees relative to the toroidal magnetic field to maximize the predicted efficiency of the ECH&CD at high plasma temperature. The EC system will also be used to assist plasma initiation with the perpendicular O- and X-mode waves. The efficiency of wave absorption is the highest at the first harmonic resonance. However, due to the presence of a cut-off layer, the first harmonic X-mode waves (X1) cannot be coupled when injected from the low-field side, as is the case in ITER. At the full toroidal field (i.e. $B_t = 5.3 \text{ T}$), only the first harmonic O-mode (O1) is available for wave coupling. At the half toroidal field (i.e. $B_t = 2.65 \text{ T}$), the second harmonic X-mode (X2) and O-mode (O2) are available for wave coupling, and the X2 wave absorption is much more efficient than the O2 wave absorption. The injection of the O1 wave at the full toroidal field and the X2 wave at half toroidal field are therefore planned for ITER plasma initiation. Experiments in current devices have successfully achieved pre-ionization and plasma burn-through within $E_t < 0.3 \text{ V m}^{-1}$ using both O1 and X2 injections [62], demonstrating the ECH scheme planned for ITER. Further optimization of EC wave absorption in ITER is planned by reflecting the O1 wave at the inboard PFCs and driving mode-conversion to the X1 waves, which is much more efficient for the wave absorption.

While ECH is the main strategy to support plasma initiation in ITER, other wave sources such as ICRH or LHCD also have a potential to be alternatives. Plasma breakdown using the standard ICRF antenna has been developed as an alternative wall conditioning technique in the presence of the steady high magnetic field. In preparation for use in ITER, the routine inter-pulse and overnight conditioning has been demonstrated in JET [67]. LHCD has shown a high potential for non-inductive plasma current start-up in JT-60U [68].

2.4.2.2. EC-assisted pre-ionization. The efficiency of EC wave absorption during the plasma breakdown phase will be low due to the low density and temperature. This will result in multiple reflections of the injected EC waves within the ITER vessel. Multi-pass absorption has been calculated for the ITER antenna design [69]. At the full toroidal field ($B = 5.3 \text{ T}$), it is predicted that a large fraction of the injected waves will be converted to an X1 wave by the Budden tunneling effect [70] at the R cut-off layer of the X1 wave, which is determined by the propagating EC polarization and the local magnetic configuration. The X1 wave has a much higher absorption efficiency than that of the O1 wave, so the mode conversion increases the effective absorption efficiency of the injected O1 wave. The effective absorption of the injected O1 wave increases rapidly with n_e and saturates to a certain high value for $n_e > 1 \times 10^{16} \text{ m}^{-3}$. As a result, the effective absorption efficiency of the injected O1 wave should be sufficient for reliable plasma breakdown in the density range of $2.5 \times 10^{15} - 10^{18} \text{ m}^{-3}$ (e.g. 2 MW absorption of 6.7 MW injection at $T_e = 5 \text{ eV}$, $n_e = 2.5 \times 10^{15} \text{ m}^{-3}$ and 6.2 MW absorption of 6.7 MW injection at $T_e = 5 \text{ eV}$, $n_e = 10^{18} \text{ m}^{-3}$). Note that the calculation does not take into account the effect of the ohmic loss of the EC wave energy on the PFCs. If such an ohmic loss is not significant, the single-pass absorption of the O1 wave in the ITER scenarios could be underestimated. In FTU experiments, the multi-pass effect for typical toroidal injection angle, $\beta = 20^\circ$ and 34° at $B_t = 5.3 \text{ T}$, $P_{EC} = 0.35 \text{ MW}$ was demonstrated [46]. In the case of $\beta = 20^\circ$, the O1 waves were reflected on the toroidal limiter at the inner wall after the first pass through the EC resonance (ECR) layer. Plasma current ramp-up (8.6 MA s^{-1}) for $\beta = 20^\circ$ was faster than that (3.5 MA s^{-1}) for $\beta = 34^\circ$. The calculated power deposition using experimental data clearly indicates that the second pass X1 is two orders of magnitude higher than the first pass O1. The polarization conversion process on PFCs is of great importance for plasma initiation and this result is promising for O1 plasma initiation to ITER.

Experimental demonstrations of X2 wave pre-ionization have also been successfully achieved in the current devices such as DIII-D [71, 72], KSTAR [63] AUG [62], EAST [73], and J-TEXT [74]. In DIII-D the X2 and O2 waves injected from the LFS were compared. As expected from the EC wave theory, no pre-ionization was observed with the O2 wave, and the faster initial plasma current ramp-up and higher density were obtained with the X2 wave [75]. In the first plasma operation of KSTAR, the effectiveness of X2 pre-ionization was demonstrated in a superconducting device for the first time [63]. According to the linear theory, especially for the X2 wave, the electron energy gain requires a finite electron Larmor radius, which is practically zero at the early phase of pre-ionization (see section 5 of [76]). At the early phase of X2 pre-ionization, the non-linear wave-particle interaction is suggested to be responsible for a seed electron energy gain [65, 77, 78]. Due to the low electron temperature, each electron stays within the beam illumination region long enough to be trapped by the wave. As a result, it attains a finite (mostly) perpendicular energy beyond the background neutral ionization potential.

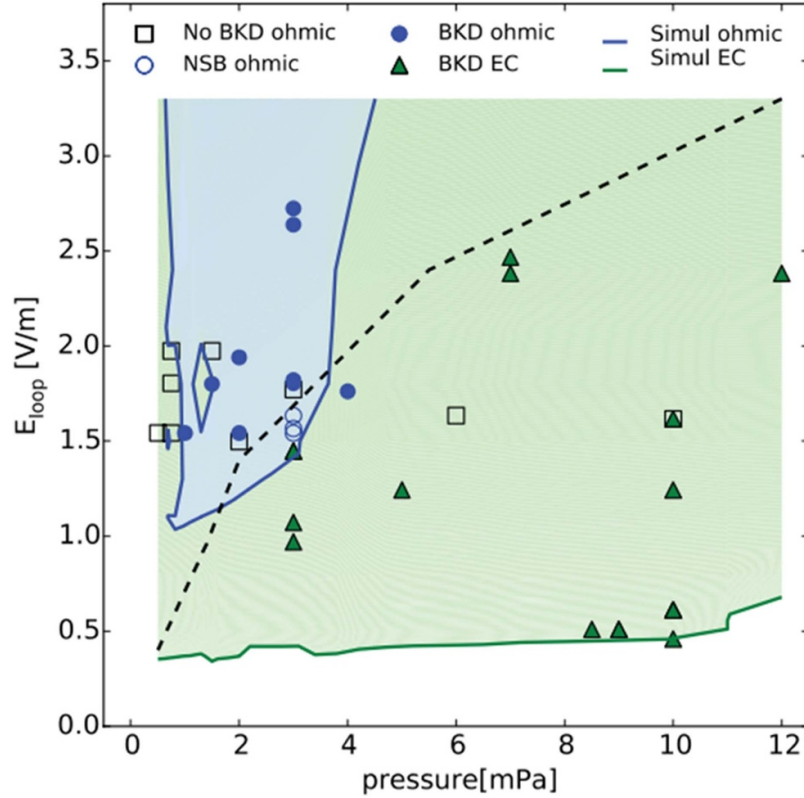


Figure 13. Operational region in terms of E_t (V m^{-1}) and p_0 (mPa) on FTU. The dashed line indicates the upper limitation to avoid to generate fast electron based on the Dreicer field. The dark blue and green areas bounded by solid line denotes the operation window based on BKD0 simulation for only ohmic and ECH assist, respectively. It should be noted that open symbols indicate no sustainable breakdown. Reproduced from [46]. © 2015 EURATOM. All rights reserved.

2.4.2.3. EC-assisted burn-through. In the DIII-D experiments, EC-assisted (X2) plasma burn-through was compared with the conventional ohmic plasma burn-through. Conventional ohmic plasma initiation has been optimized to produce an electron avalanche at a low toroidal electric field $> 0.3 \text{ V m}^{-1}$ (typically 0.9 V m^{-1}) in a FNC. The position of the initial maximum visible radiation was located at the X2 resonance layer ($R = 1.8 \text{ m}$) in the EC-assisted discharge, and current channel could appear in the different position, while the initial maximum visible occurred at $R \sim 1.18 \text{ m}$ near the inboard wall (at $R = 1.02 \text{ m}$) in the ohmic discharge. The position is consistent with the plasma current centroid ($R \sim 1.09$ to 1.20 m) and is also in the region of low PF ($< 9 \text{ G}$) and long connection length ($\sim 1000 \text{ m}$). This implies the initial current formation in EC-assisted discharge is different from that in ohmic discharge. The EC-assisted discharge showed much faster oxygen burn-through than the ohmic discharge [75].

The operating space required for successful plasma initiation (reaching 150 kA plasma current) in FTU is shown in figure 13. The experimental data were compared with the results of the BKD0 simulation, which included the wall reflection modeling of the EC wave. The power balance based on the FTU experiment data indicates that the multi-pass EC absorption played an important role in the plasma burn-through. The ohmic plasma burn-through was only successful at high E_t and low pre-fill gas pressure p_0 . The operating space required for successful plasma burn-through became significantly wider

with the EC wave, and the experiment data was well in agreement with the results of the BKD0 simulation, which assumed 2% O molecule content. The operating space was limited by the E_t threshold defined by the Dreicer electric field as shown by the dashed line in figure 13. In the FTU breakdown experiments with the EC wave, the generation of runaway electrons was unexpectedly observed at low E_t with $p_0 < 8 \text{ mPa}$. This is important for the ECH application, as such runaway electrons should be avoided during the initial phase of operation to protect the machine.

2.4.2.4. Plasma initiation without toroidal electric field E_t .

Plasma initiation without E_t would benefit the operation of ITER. It would not only save the volt-second in the CS but also reduce the risk of runaway electrons damaging the unprotected internal vessel components in ITER during the commissioning phase. Plasma initiation without E_t has been attempted in current devices such as TRIAM-1M (O1 $20 \text{ kA}/0.1 \text{ MW}$) [79], DIII-D (X2 $33 \text{ kA}/2.2 \text{ MW}$) [80], KSTAR (O1 $14.5 \text{ kA}/0.18 \text{ MW}$) [81] and QUEST (X2 $66 \text{ kA}/0.2 \text{ MW}$) [82]. In the DIII-D experiments, where the aspect ratio is similar to that of ITER (i.e. $1/\varepsilon_{\text{DIII-D}} = 2.5$ and $1/\varepsilon_{\text{ITER}} = 3.1$), a limited set of operating parameters were investigated; pre-filled gas pressure = $0.1 - 0.3 \text{ mTorr}$, $B_t = 1.5 - 2.2 \text{ T}$, $B_z = -60 - -40 \text{ G}$, and injected $P_{\text{EC}} = 0.8 - 2.8 \text{ MW}$. The increase in the non-inductive I_p was observed by increasing the pre-filled gas pressure, or reducing

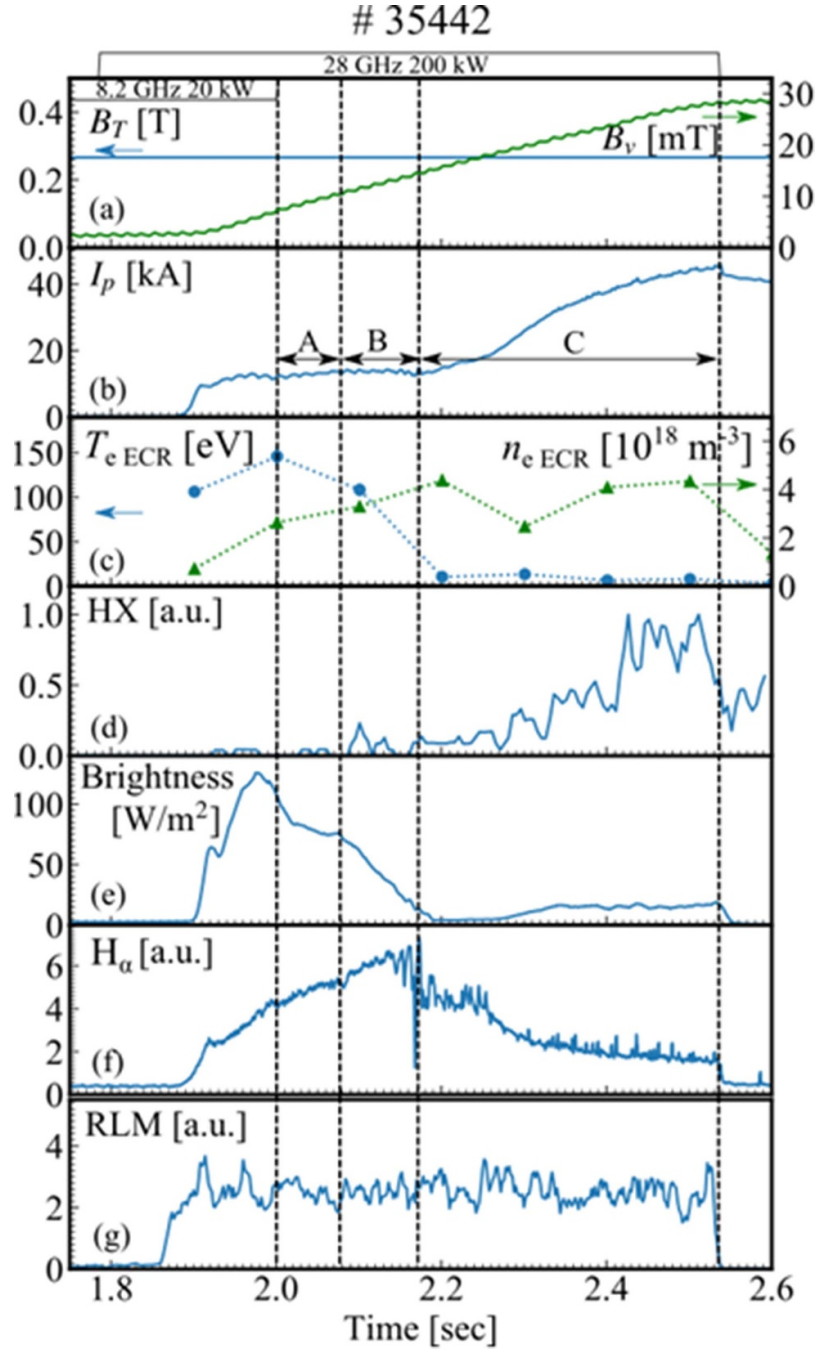


Figure 14. Time evolutions of B_t and B_p (a), plasma current (b), T_e and n_e (c), HXR emission (Energy of x-ray > 40 keV) (d), radiation loss (e), H alpha (f), and stray RF power (g) in plasma initiation without E_r on QUEST with X2. Reproduced from [83]. © IOP Publishing Ltd All rights reserved.

the B_t (i.e. moving the X2 resonance layer towards the HFS. In the scanned range of B_z and P_{EC} , the non-inductive I_p did not vary much. The maximum non-inductive I_p achieved was 33 kA, and the dominant driving mechanism was found to be the Pfirsch–Schlüter current on open field lines [80].

In the QUEST experiments, the non-inductive I_p was achieved up to 66 kA at 0.2 MW EC power. The EC wave had 60% O2 and 40% X2 modes at the plasma boundary and $N_{||} \sim 0.78$ at the second harmonic resonance layer. The non-inductive I_p was mainly driven by the bootstrap current

at the beginning (i.e. the phase A and B in figure 14), and subsequently by the runaway electrons. Figure 14 shows that in the phase C the measured I_p and the detected hard x-ray emission increase together [83]. When the runaway electrons are present, the effective EC energy absorption on the runaway electrons further increases the fast electron population and energy. This results in a decrease in the thermal electron temperature (down to 5 eV equivalent to the ion temperature, see figure 14(c)). The non-inductive I_p and the closed flux formation have been modeled and the experimental data

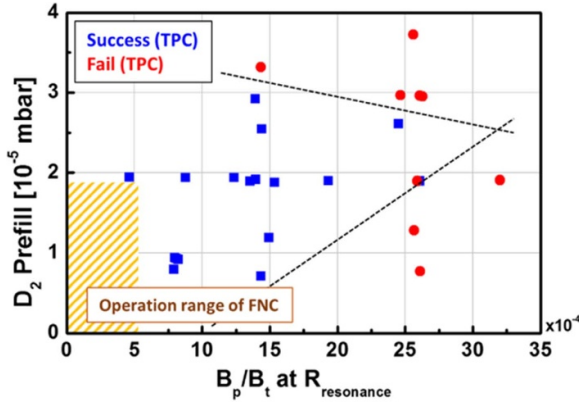


Figure 15. Operation region in terms of P_{pre} and B_p/B_t for TPC on KSTAR. The region for FNC is indicated by rectangle filled by diagonal lines. Reproduced from [85]. © 2017 IAEA, Vienna. All rights reserved.

have been quantitatively analyzed in detail [84]. A positive feedback between the runaway electrons and the EC wave absorption could significantly improve the non-inductive I_p . However, as it also carries the risk of damaging the device, careful optimization of the injected EC power with monitoring of the runaway electrons will be necessary.

2.4.2.5. Initiation with finite vertical magnetic field. The FNC has been conventionally used for plasma initiation in most tokamaks, because it has a long effective connection length, thereby reducing the loop voltage required for successful Townsend breakdown and plasma burn-through. On the other hand, a completely different magnetic configuration, the so-called TPC, has recently been demonstrated in current devices, including conventional tokamaks such as KSTAR [85] and spherical tokamaks such as VEST [86], LATE [87], TST-2 [88], and QUEST [83]. The TPC has a positive decay index and is similar to the vacuum field configuration during the limiter phase. It enhances the confinement of the trapped particles in the open field lines [89], and the confined electrons help plasma breakdown and CFSs formation [87]. In the KSTAR experiments, the TPC was found to allow the wider operating window of the pre-filled gas pressure (see figure 15 [85]). In the DIII-D experiments, the injected X2 EC wave power threshold for successful plasma initiation was lower than that in the FNC. In the JT-60SA first operation, the TPC demonstrated the enhanced resilience to the poloidal magnetic field errors. The TPC successfully sustained the plasma in the open magnetic fields, and it enabled a reliable plasma initiation, which was not achievable with the FNC [90]. In the TST-2 spherical tokamak experiments, the TPC and the FNC were directly compared using the same time trace of E_r . The EC wave was injected at the same time as E_r induction, and faster plasma breakdown and higher plasma current were achieved in the TPC [88].

The TPC also simplifies the feedforward PF circuit scenarios. A positive decay index is required for the VS of plasmas during the I_p ramp-up phase. Thus, the poloidal FNC requires additional feedforward voltage control in the PF circuits after

the plasma burn-through phase. As the TPC already has a positive decay index, the same magnetic field configuration can be maintained from the plasma breakdown phase to the I_p ramp-up phase. This would reduce the requirements for the power supply voltage connected to the superconducting coils and also for the EC power.

2.4.3. Runaway electron formation during plasma initiation

2.4.3.1. Impact of runaway electrons. First of all, the definition of *suprathermal electrons* and *runaway electrons* is clarified. Electrons in high tails that deviate from a Maxwellian distribution are defined as *suprathermal electrons*, but only those that are continuously accelerated (because the acceleration exceeds the collisional drag force) are defined as *runaway electrons*. At plasma initiation, the average energy of the suprathermal electrons is in the range 1–100 keV, while average energies above 1 MeV up to tens of MeV usually characterize the runaway electrons [91].

Runaway electrons can occur at different stages of a plasma discharge, during the I_p ramp-up, during the I_p flattop (when electron density is low), or during a plasma disruption. In general, runaway electrons generated during a plasma disruption are considered to be a serious threat as they can damage the PFCs. In practice, however, the presence of runaway electrons is always potentially dangerous, even when they occur during the plasma initiation phase, because their sudden loss can lead to a disruption event if not properly controlled or suppressed.

Damage to the PFCs caused by runaway electrons has been reported in current devices, e.g. toroidal field coil quench in WEST [92], beryllium melting in the upper part of the JET inner limiter [93], piercing of a cooling tube of the pump limiter in Tore Supra [94], bulk damage to graphite limiter tiles in DIII-D [95]. Another concern is the fact that runaway electrons affect the plasma formation by influencing the ohmic current and heating. The runaway electron current inductively stores a part of the induced loop voltage, thus limiting the electron temperature rise and hindering the I_p ramp-up. This effect could be significant even with a small number of runaway electrons ($n_{\text{RE}} \ll n_e$) due to their relativistic speed ($\sim c$). The runaway electron current, $I_{\text{RE}} \sim n_{\text{RE}} e c A$ (where A is the cross-section of the runaway electron beam), easily becomes of the same order as the ohmic current.

Therefore, a cautious approach should be taken in ITER to ensure a reliable and reproducible plasma initiation scenario [91]. It is important to control and limit the relativistic runaway electron current that may inevitably occur during the plasma initiation phase, especially during the ITER First Plasma Operation.

2.4.3.2. Process of runaway electrons during the plasma initiation. The theory of runaway electrons in tokamak plasmas is well covered in the recent review paper [96]. In the following, only key concepts are summarized which are useful to deal specifically with runaway electrons during plasma initiation.

The definition of the *critical electric field* for relativistic electrons in a fully ionized plasma without radiation losses is

given by the Connor–Hastie formula [97]:

$$E_c = \frac{n_e e^3 \ln \Lambda}{4\pi \epsilon_0^2 m_e c^2} \quad (16)$$

Below this threshold, absolutely no runaway electrons are produced, while above it all electrons with momentum $p > p_c$ (p_c : critical momentum for runaway generation) are rapidly accelerated.

There are two mechanisms for the generation of runaway electrons during the plasma initiation phase i.e. the Dreicer (or primary) generation [98] and avalanche (or secondary) generation [99]. According to the Dreicer kinetic theory [100], runaway electrons are generated by the diffusive flux in the velocity space that forms across the critical velocity under an effect of electric field when the condition $E > E_c$ is met [97]. During the plasma initiation phase, Dreicer generation is the dominant source of runaway electrons. The generation rate is a strong function of E/E_D (see equation (82) in [96]), where E_D (the Dreicer field) is related to the critical field E_c by the following formula (see e.g. [101]):

$$E_D \equiv \left(\frac{m_e c_e^2}{kT_e} \right) E_c \quad (17)$$

The threshold value of the electric field for the effective Dreicer generation can be estimated with E_D i.e. $E > 0.02E_D$ [99].

In weakly-ionized plasmas such as the plasma initiation phase, an effect of neutrals on electrons should be considered [90, 102, 103]. When the critical momentum is much higher than the ionization potential in the phase space, the diffusive process plays a key role [100]. However, when the corresponding energy to p_c is comparable to the ionization potential (typically relevant to an early phase of initiation), the binary nature of inelastic collisions that facilitate the runaway electron generation should be accounted by the non-diffusive electron kinetics [104].

In the avalanche mechanism, knock-on collisions between the existing runaway electrons and the thermal electrons produce additional runaway electrons. The number of runaway electrons exponentially increases with time. During the plasma initiation phase, runaway electrons could be generated by the Dreicer mechanism. Once a stable runaway population has been established, the avalanche mechanism may also play a role.

Note that the above picture is very simplified and in practice other factors have to be taken into account to define the conditions under which runaway electrons can occur in the plasma initiation phase. The amended critical momentum effectively incorporates additional kinetic effects into the runaway generation rate, e.g. by evaluating equations (16) and (17) at p_c . For instance, before the impurity burn-through, a key role played by impurities and neutrals in Dreicer generation can be considered by replacing $n_e \ln \Lambda$ as $n_e \ln \Lambda + n_{\text{bound}} \ln \Lambda_b$ in E_c , where n_{bound} is the bound electron density and $\ln \Lambda_b$ is the associated logarithmic factor [105]. This can be further elaborated by extracting it from the complete steady-state solutions aided by numerical simulations [104, 106, 107]. In addition, as the

runaway electrons are continuously accelerated during ramp-up, an effect of radiations modifies the effective critical field for runaway amplification [108, 109]. If the radiation losses of the runaway electron energy are taken into account, the critical electric field could actually be higher than that given by [110] (equation (16)). A formula for the critical electric field, valid in a fully ionized plasma including synchrotron radiation losses, was introduced in [110] and is in agreement with experimental observations in current devices [111, 112]. The critical electric field taking into account the presence of partially screened impurities (for arbitrary ion species) in combination with synchrotron and bremsstrahlung losses is given in (equation (23)) in [113] (note that complete screening occurs when the electron interacts only with the net ion charge, while there is no such screening effect when the electron experiences the full nuclear charge i.e. fully ionized impurity case).

2.4.3.3. Observation in present devices. Although runaway electrons in the I_p ramp-up phase have been extensively studied for decades (see a review in [114]), these observations have generally not had good diagnostic coverage. In more recent experimental observations, the spatial location of the runaway electrons has been obtained by using arrays of x-ray detectors to measure the bremsstrahlung radiation emitted by in-flight runaway electrons. Energy information is provided by gamma ray spectrometers and synchrotron radiation detectors operating in the visible and IR ranges. Recently, a good experimental study of runaway electrons in the I_p ramp-up phase has been carried out at JET, though the diagnostic capability for runaway electrons was only limited to non-thermal ECE [115].

Spatially resolved measurements of x-ray bremsstrahlung emission made at JET [116] and FTU [117] have shown that runaway electron production during the I_p ramp-up phase of low-density ($\sim 10^{19} \text{ m}^{-3}$) ohmic discharges is localized in the center of the plasma (see figure 9 in [118]). This is consistent with Dreicer generation, since E/E_D is the highest in the central region of the plasma.

After generation, runaway electrons diffuse towards the plasma edge and are lost at the vessel structures. In the early I_p ramp-up phase, the rapid loss of runaway electrons is sometimes observed due to the loss of plasma position control or MHD events. In early JET operations, the radial diffusion coefficient, D_r , was determined by using a diffusion model to simulate the time traces of the experimental x-ray signals from different chords ($D_r \sim 0.2 \text{ m}^2 \text{ s}^{-1}$) [116]. Lower diffusion coefficients ($D_r \sim 0.03 \text{ m}^2 \text{ s}^{-1}$) have been found in the more recent set of JET discharges with the ILW [115]. Similar values have been reported in FTU ($D_r \sim 0.03 \text{ m}^2 \text{ s}^{-1}$) [117].

The process of runaway electron generation stops immediately as the thermal electron density increases. The fate of the runaway electrons depends on the thermal electron density. If this is high enough (e.g. $\sim 10^{20} \text{ m}^{-3}$), all runaway electrons can be suppressed, whereas if the thermal electron density remains low, runaway electrons can be present throughout the discharge, reaching a maximum energy that is determined by the balance between the energy gain in the electric field and

the energy losses due to plasma collisions and radiation losses (e.g. synchrotron radiation) [101]. Values up to ~ 30 MeV have been measured in FTU [119] and up to ~ 20 MeV inferred in JET [93].

The early presence of runaway electrons can influence the subsequent evolution of the discharge. The current may be partly or wholly carried by runaway electrons (i.e. beam-like), or runaway electron seeds could be further accelerated when a disruption occurs at a later stage. Indeed, enhanced production of runaway electrons during disruptions [120] has been observed due to the acceleration of pre-existing suprathermal electrons during the current quench of disruptions with the formation of a post-disruption runaway current plateau. In JET, the loss of runaway electron beams due to the runaway electrons generated at plasma initiation has been identified as the cause of melt damage in PFCs [93].

In addition to V_{loop} reduction, gas puffing or plasma heating by auxiliary heating can be used to control the runaway electron population. Gas puffing increases the thermal electron density and thus suppresses runaway electrons [112, 115]. Due to the large pumping rate of hydrogenic species by the beryllium wall, ITER will need sufficient gas fueling during the plasma initiation phase to avoid the formation of a significant electron runaway population [91]. Heating the plasma with ECH (i.e. increasing T_e) [121] reduces the plasma resistance which leads to a decrease in the accelerating electric field, thereby reducing the runaway electron energy and, if sufficiently high, suppressing runaway electrons. Note that the same effect can be achieved using LH [122] although care must be taken in this case as the presence of suprathermal electrons produced by LH can cause the runaway electron population again once LH is switched off.

At toroidal electric fields well below E_D , runaway electrons have been detected in the FTU EC-assisted plasma initiation experiments [123, 124]. The results report the presence of runaway electrons even at low EC power injection (400 kW) (see figure 16(left)). This shows that EC waves can produce suprathermal electrons, which would increase runaway electrons. A possible responsible mechanism is a non-linear interaction of free electrons with the electromagnetic wave, as described in [84], which modifies the electron distribution functions (giving perpendicular momentum) in such a way that the threshold electric field for runaway electron generation is reduced and more runaway electrons are produced.

According to the theory, electrons can gain energy in the range of a few keV, independently of the EC beam energy. The interaction is more effective at the fundamental resonance (O1) than at higher harmonics, and greater for extraordinary polarization than for ordinary polarization. In FTU (and in the future in ITER) the O1 EC beam has a poor absorption during the EC-assisted plasma initiation phase, due to the low temperature and density. Therefore, most of the RF power is reflected at the inner metallic wall with a consequent polarization scrambling that can produce the extraordinary fundamental polarization (X1), which is more effective in generating suprathermal electrons. Thus, even in the EC beam with a small fraction of extraordinary mode, electrons at room temperature can easily gain energies, via non-linear EC

interaction, of up to a few keV at the first harmonic and of the order of 100 eV at the second harmonic, in the case of the X2 scenario [84].

Analysis of the FTU results with EC-assisted plasma initiation suggests that a possible parameter controlling the generation of runaway electrons is the pre-filled gas pressure rather than the thermal electron density. Figure 16(right) shows that the discharges at low pre-filled gas pressures generated higher levels of runaway electrons. EC-assisted plasma initiation should be operated at a high pre-filled gas pressure to avoid runaway electrons generated by the EC beam itself.

2.4.3.4. Prospect of runaway electrons in ITER. As explained in the previous sections, the runaway electrons that can be generated during the plasma initiation phase in ITER deserve a great deal of attention; at low plasma densities, they can be detrimental to plasma formation. Even in the presence of a formed plasma, they can cause disruption and damage to the PFCs. In ITER the expected plasma initiation parameters may lead to runaways. Considering, for example, the scenario shown in figure 11, the E/E_D ratio can reach values in excess of 0.02 (figure 17), therefore exceeding the threshold value of the electric field for the effective Dreicer generation [99]. Note that the resistive V_{res} must be used for the calculation of E in the E/E_D ratio, namely $E = V_{\text{res}}/(2\pi/R_0)$, where $V_{\text{res}} = V_{\text{loop}} - L \cdot dI/dt$. Future research should address the following issues:

Experimental investigation of the effect of ECH on runaway electrons generation. For ohmic plasma initiation in ITER, a low pre-filled gas pressure would be required. ECH assisted plasma initiation may allow operation at a high pre-filled gas pressure. The FTU experiments above show that a high pre-filled gas pressure should help prevent runaway electrons. However, there is some experimental evidence that ECH may affect the electron distribution function and the possibility of runaway electrons in ITER cannot be completely excluded. Dedicated experiments should be carried out in ITER-relevant plasmas (i.e. ordinary and extraordinary EC wave polarization at $B_t = 5.3$ T and 2.65 T) in order to unambiguously monitor any runaway electron enhancement and to determine the best conditions, if any, to avoid it.

Modeling of plasma initiation including the presence of runaway electrons. To ensure a design of a runaway-free plasma initiation scenario, in which the plasma parameters remain far from the significant runaway electron generation, it is necessary not only to reliably predict their generation but also to validate the model against observations in existing tokamaks. Recently the effects of runaway electrons have been self-consistently modeled by the plasma burn-through modeling codes such as SCENPLINT [125], STREAM [126], and INDEX-S [127]. In the simulations, the fluid treatment is adopted by assuming the steady-state solutions of distribution function. This assumption is verified by the kinetic modeling under the typical time scales of plasma parameter evolution during plasma initiation [128]. The qualitative simulations indicate that the Dreicer generation dominates the

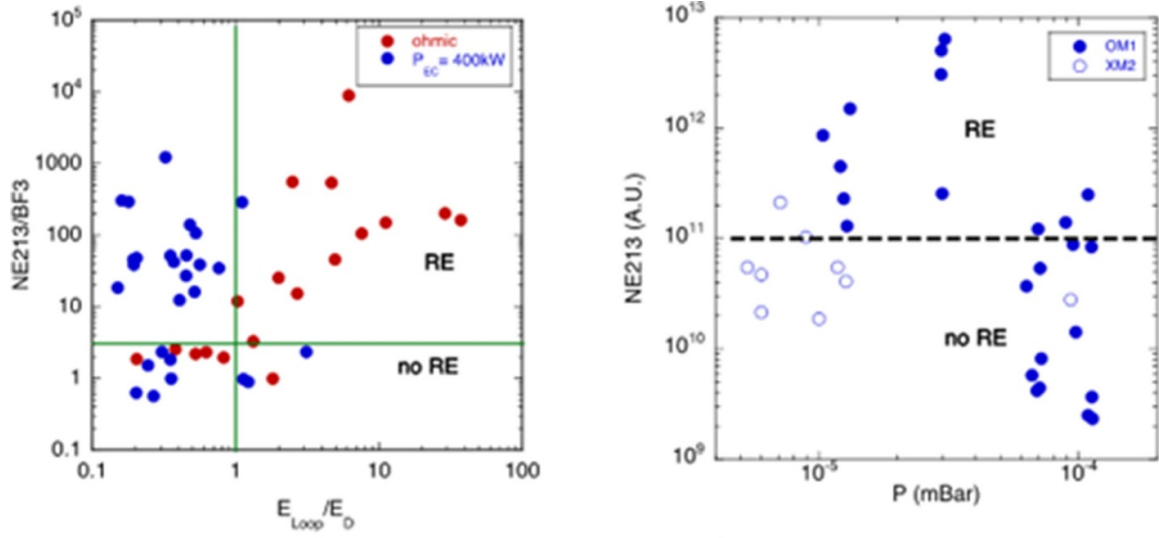


Figure 16. Runaway electron presence in EC-assisted start-up discharges in FTU monitored through hard x-ray (NE213) and neutron signals (BF3); (left) as a function of E_{loop}/E_D with $B_t = 5.3$ T and various filling pressure/injection angles, (right) as a function of pre-fill pressure with different wave polarizations; a reduced effectiveness of XM2 in producing RE is observed, in agreement with the predictions of [65]. Pictures are reproduced from [123]. © 2017 IAEA, Vienna. All rights reserved, with left picture turned to double-logarithmic scale for better clarity.

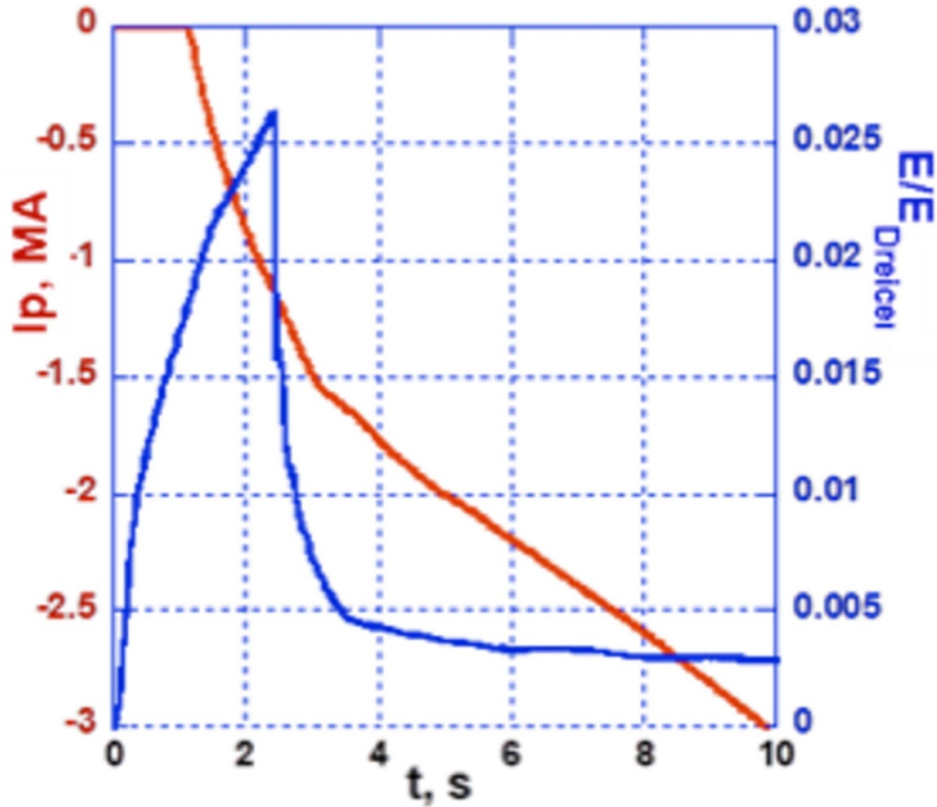


Figure 17. Calculated E/E_D ratio in ITER plasma initiation scenario.

runaway electron generation. Hence, the runaway electron generation condition is significantly dependent on the control knobs on electron density such as pre-fill gas pressure [125], wall condition [127], and gas puffing [126].

This type of analysis should be made more quantitative, including the time evolution of the plasma volume and position, the V_{loop} as well as the effect of impurities such as the binary nature of collisions [104] aided by the closed flux surface

formation physics. In addition, the improved radial transport model is required to extrapolate the confinement time since the existing models [114, 129] are not enough to explain the experimental observations [130]. Furthermore, the response of the PCS, the effect of runaway kinetic energy (less than 0.511 MeV) and the ECH should be also implemented in the model.

Development of an early runaway electron monitoring system integrated with control tools. Appropriate methods for the real-time monitoring and control of runaway electrons for the plasma initiation phase should be established and tested in present devices so that they can become a routine tool in the ITER commissioning phase and beyond. Such methods should be based on real-time diagnostics so that the key parameters controlling the primary and secondary runaway electron generation (E/E_D , E_c) and the runaway electron current I_{RE} can be monitored and immediate control actions to suppress or reduce the runaway electrons (e.g. deuterium gas puffing, reduction of V_{loop} or auxiliary heating) can be taken.

E/E_D is easily obtained from interferometry (electron density), Thomson scattering (electron temperature and density), and flux loop (V_{loop}) measurements. The I_{RE} is proportional to the intensity of the hard x-ray bremsstrahlung radiation measured by scintillators, from which the neutron contribution is subtracted, and to the intensity of the visible/infrared synchrotron radiation measured by visible/infrared spectrometers and cameras [131]. The runaway electron distribution function (space- and energy-resolved) that can be derived from such spectrometers and cameras [132], while potentially interesting and useful, may not be readily implementable for routine use.

2.5. Outstanding issues and conclusions

ITER plasma initiation is expected to be challenging. The limitation of the low toroidal electric field and the large vacuum volume in ITER would allow the ohmic plasma initiation only at a low pre-fill gas pressure below 1 mPa. However, at such a low pre-fill gas pressure, there is a risk of runaway electron generation during the plasma initiation phase, which could lead to plasma initiation failure or disruptions in the later phase. It has been reported that increasing the pre-fill gas pressure could reduce the risk of runaway electron generation during ohmic plasma initiation. For ITER plasma initiation at a high pre-fill gas pressure (a few mPa), EC heating assistance will be required. The EC injection schemes in ITER, the X2 wave (for half toroidal field) or the O1 wave (for full toroidal field), have been demonstrated in the current operating devices. Also, in addition to the conventional poloidal FNC, improved PF configuration TPC has been demonstrated for EC-assisted plasma initiation. In order to extrapolate the present experimental results to ITER and to optimize the EC injection scenario required for ITER plasma breakdown and burn-through, the plasma initiation models should be validated with the calculation of the EC electron energy gain during the pre-ionization and the EC power absorption during the plasma burn-through phase using the present experimental

data. It has also been reported that EC injection in the plasma initiation phase could increase the risk of runaway electron generation even at high pre-fill gas pressure. To optimize the control knobs in ITER and to ensure reliable plasma initiation at high pre-fill gas pressure without runaway electron generation, it is necessary to improve the predictive model for runaway electron generation and validate it in the present devices.

3. ITER low activation phase scenarios

Coordinated by F. Koechl

3.1. Introduction

In accordance with the updated IRP [133–135], a staged approach has been foreseen for the operation of ITER with the main aim to demonstrate stationary DT fusion burn at $Q \sim 10$. With the SRO, an extensive set of scenarios has been planned to be executed as part of low activation phases operating initially in H and later in D with the aim to gradually approach conditions required for a subsequent FPO at reduced neutron yield which involves a low fusion power campaign in D + possible trace amounts of T that would then be followed by a high performance DT burning plasma operation campaign. In this section, the motivation and physics background for the design of the scenarios foreseen for the low activation phase are described. These scenarios are not only required for systems commissioning and physics validation in the ITER environment in preparation of DT operation in FPO, but they are also highly relevant in themselves, as they allow for the first time the experimental demonstration and exploration of plasma conditions and physics mechanisms at play at the fusion reactor scale, involving various aspects that cannot be fully explored in present-day tokamaks as they become relevant only in a reactor environment. In the following section 3.2, the plan and physics motivation for the execution of these scenarios is outlined. In particular, many aspects relevant for making predictions of ITER plasma conditions that cannot be projected with very low uncertainty from the current physics knowledge based on present-day machines can be re-evaluated in ITER experiments in SRO and early FPO, which will allow through local optimization to significantly improve conditions and predictions for the later FPO high performance ITER scenario operation phase that are necessary to operate this phase in an efficient manner with minimum interventions and to ensure that ITER's goals could be achieved or even surpassed. The importance and merit of local optimization in view of fulfilling machine operation objectives has recently been demonstrated at JET where a careful re-optimization of plasma conditions and operation in view of efficient execution and performance maximization of JET DT campaigns was carried out after the change to a Be/W wall [136, 137] ensuring a highly successful plasma operation in DT [138, 139]. In sections 3.3 and 3.4, dedicated experimental and modeling studies in support of these campaigns are reviewed, respectively. An outlook is provided in section 3.5 about questions associated with these scenarios that will need to be addressed in more detail in

the future for improved risk mitigation, for the establishment of the scenarios under consideration, and for the investigation of possibilities for their optimization as well as for the extension of the associated operational space. Finally, a brief summary of the discussion of the current level of understanding for the physics basis of the ITER low activation phase scenarios and recent adaptations of the low activation phase experimental plan in line with the ITER re-baselining programme is provided in section 3.6.

3.2. ITER low activation phase experimental program

The set-up of the low activation phase campaign (SRO) and the initial part of FPO and the projected sequence of scenarios to be established is motivated by various considerations and constraints: The transition from initial plasma configurations at low current, field, and confinement at the start of SRO to the targeted 15 MA/5.3 T DT baseline scenario at $Q \sim 10$ needs to be performed progressively in a staged approach to mitigate risks associated with the exploration of plasma conditions at higher currents and fields at previously unprecedented levels. As part of this transition, it is also important to gradually improve the ITER science basis and reduce uncertainties in predictions in preparation to operation in those conditions foreseen in the later FPO campaigns ensuring that the operation is done in an efficient manner minimizing possible interventions and neutron fluence. The transition needs to be carried out at fields providing optimum resonance conditions for wave heating. Either a sequential increase in current and field or an increase in current at fixed field may be foreseen. The latter may be preferable with W wall coating foreseen with the recent ITER re-baseline [134] to ensure effective central heating at fixed favorable $B_t = 2.56$ T and 5.3 T. With application of NB heating, a lower density limit for the avoidance of noticeable beam shine-through losses needs to be taken into account. In addition to these considerations, constraints due to limited availability of heating, diagnostics, and control systems in the early operational phase need to be considered. At the same time, it is important to ensure that the commissioning of all tokamak systems that will need to operate reliably in high performance plasma scenarios at higher currents and fields, as well as the establishment of targeted plasma conditions that are envisaged for these high performance scenarios can be accomplished in the SRO campaigns. In particular, the achievement of operation in H-mode will need to be demonstrated and associated ELM and W control schemes be validated as early as possible in the ITER low activation phase, while the commissioning of most other systems will mainly be accomplished in H L-mode plasmas. To minimize the neutron production, only plasmas that consist of pure hydrogen or deuterium have been foreseen for SRO, while either pure deuterium plasmas or mixtures of H with only a trace amount of tritium have been planned to be studied in the early FPO phase to maintain for as long as possible a low activation of the ITER device, which facilitates maintenance and ensures a faster execution of the projected experimental program. ^4He has been considered as an alternative to H even though, with H, plasma transport, sputtering and fuelling properties are expected to be

more favorable and better comparable to those for other hydrogenic isotopes. There is experimental evidence for a significantly reduced power threshold for the transition to H-mode for ^4He as compared to H. ^4He scenarios have therefore originally been envisaged as well, as they may help ensure that high quality H-mode conditions can be achieved and fully explored already at an early stage of the experimental campaign when the available auxiliary power is still limited. With the recent ITER re-baseline, plasma operation in ^4He is not foreseen although it could be reconsidered if deemed necessary in certain conditions depending on experimental observations.

The schedule and sequence of ITER low activation phase scenario target configurations and milestones to be realized mainly as part of the SRO and initial FPO campaigns in accordance with [133] and the recent ITER re-baseline can briefly be summarized as follows:

- SRO (H, D):
 - initial limiter plasmas at 3.5 MA in H
 - diverted L-mode plasma operation at $\sim 3\text{--}7.5$ MA in H mainly at 2.65 T
 - diverted H-mode investigation in D at $\sim 5\text{--}7.5$ MA at 2.65 T
 - diverted L-mode plasma operation at high field and increased current from 5 up to 15 MA.
- FPO, early phase (H, H + T, D):
 - L-mode plasma operation in H at fields and currents up to 5.3 T and 15 MA for commissioning of new systems
 - L-mode plasma operation with tritium (H+T) at 15 MA/5.3 T)
 - initial development of hybrid/advanced scenarios in D at 3.75–5.6 MA/2.65 T
 - H-mode plasma operation in D at 5–7.5 MA/5.3 T.

The main goals of the low activation phase campaigns that need to be achieved are the commissioning and demonstration of the ITER PCS [140, 141], the ITER DMS [142–144], the ITER diagnostics system [145, 146], the H&CD systems [147–152], the ITER fuelling and pumping systems [153, 154] and the ITER ELM mitigation systems [155, 156], as well as the establishment and study of the abovementioned (see also sections 7 and 8) low activation scenario configurations and conditions for the improvement of the reactor scale physics basis required to prepare for the execution of the high performance nuclear phase of operation in support of the achievement of the ITER Project's goals.

System commissioning and re-commissioning is foreseen to take place in a continuous manner in and before the start of SRO and FPO campaigns and carried out whenever new systems become available or operation conditions are foreseen to change substantially. Many core tokamak components and systems such as the cryostat, the vacuum vessel, superconducting coils, in-vessel coils, the divertor, as well as fundamental control, safety and interlock systems required for plasma operation are foreseen to be deployed and tested for operation in an Integrated Commissioning phase IC-I preceding the SRO phase. Concurrently with the development of plasma scenarios during SRO, a large number of systems will be commissioned

with plasma involving plasma control and protection systems, ECH including NTM control and ICH, diagnostics, and the DMS. The H-mode scenario development in SRO in D will be important for the testing and demonstration of ELM and W control as well as (D) fuel removal strategies. Before the start of FPO, a larger number of newly installed components including notably the NBI and enhanced EC/IC H&CD systems will need to be commissioned. The impact of operation with a new W water-cooled FW and TBMs will be assessed in early FPO. T containing H plasmas in early FPO will be used to commission the DMS in presence of high energy electrons from beta-decaying tritium during disruptions to demonstrate its effectiveness in plasmas with up to 15 MA/5.3 T and up to 100% T concentration, and they will be applied for tests of tritium removal schemes. Joint experimental and modeling activities to address R&D questions that are particularly relevant to the ITER low activation phase program have been performed within the ITPA IOS group (see also sections 4–6). In particular, the H-mode operation close to the L–H transition power threshold (IOS-1.3), plasma operation with dominant electron heating and low torque (IOS-1.4), H-mode operation in He plasmas (IOS-2.1), and the impact of the main ion isotope effects on plasma operation in H and He plasmas (PEP-IOS-1) have been investigated, and modeling predictions for the main low activation phase scenarios have been carried out (IOS-JA-10). Results obtained in these activities are discussed in the following subsections.

3.3. Key physics aspects for ITER low activation phase scenario development

The development of ITER plasma scenarios in the SRO and early FPO phases is driven by operational constraints due to limitations in the available HSs in the SRO campaigns and the need to establish and explore scenario conditions relevant to high performance DT plasmas at low activation and reduced confinement. These scenarios facilitate the exploration of the operational space and the commissioning of ITER plasma operation and control schemes. Physics related questions that may be specifically relevant to plasma operation in these phases, such as H-mode operation near the L–H transition threshold, plasma operation with different main ion isotope composition, as well as H&CD, fuelling and divertor control properties in the low activation phase scenario conditions are described in this subsection.

3.3.1. H-mode access. The operating range of H-mode in ITER is limited by three factors; the Greenwald density limit, the power loss across the separatrix exceeding the L–H power threshold, and the NB shine-through limit. Hydrogen operation at low density required for H-mode access is close to the shine-through limit, even with impurities present. Type-I ELMy H-mode operation may not be achievable during the hydrogen phase due to the unfavorable mass dependence of the L–H power threshold. However, high quality H-mode operation might be achievable in the range of low magnetic field $B_t < 2.6$ T in hydrogen plasmas.

Helium operation is more attractive for H-mode access because of the reduction of the NB shine-through limit and possibly reduced threshold power at a given density. Recent experiments have shown that the L–H power threshold for helium plasmas can approach or even be equal to the threshold for deuterium, so the H-mode operational space in the ITER helium plasmas may be wider than previously thought.

Accessing and sustaining H-modes in the low activation phase involves specific challenges related to the plasma species used (H, He, and their mixtures) and the characteristics of the available H&CD systems. Key factors include the power that can be coupled to the plasma in SRO, considering plasma absorption for wave heating and shine-through losses for NBI.

Known uncertainties in the prediction of the power required for H-mode access in ITER (e.g. impact of divertor configuration, wall/divertor material, wall conditioning, plasma rotation, edge physics and neutral processes, etc) are known to be significant, and even though they are of obvious interest for all phases of the ITER experimental plan, they are particularly relevant in the low activation phase as the H-mode plasmas will have to be sustained with reduced input power close to the expected threshold power for the L–H transition P_{LH} . If H-mode can be achieved near the lower end of the uncertainty band in terms of P_{LH} , that would be highly beneficial for the routine achievement of high quality type-I ELMy H-mode in SRO, while the investigation of regimes with type-I ELMs required for the commissioning of ELM control and mitigation schemes may be more challenging, if P_{LH} is found to be higher compared to scaling predictions. In addition to these uncertainties, a strong dependence of P_{LH} on the plasma main ion isotope composition that has been observed in experiment and the need to operate near a low optimum density for the minimization of P_{LH} play an important role for the design and operation of H-mode plasma scenarios in the low activation phase.

The impact of plasma species on the H-mode power threshold can be characterized by a simple multiplier to a multi-machine scaling [157] to estimate the L–H transition threshold power in deuterium plasmas, $P_{LH,D}$. This scaling includes a positive dependence of P_{LH} on plasma density for higher densities, on toroidal field, and on the plasma surface area. For hydrogenic plasmas, this multiplier can be expressed as $1/A_{main}$ i.e. for hydrogen plasmas $P_{LH,H} = 2P_{LH,D}$, while for He plasmas $P_{LH,He} = 1.4P_{LH,D}$, in accordance with experimental findings in [158, 159]. It should be noted, however, that significant deviations from these multipliers have been observed in experiments, depending on details in plasma conditions.

Understanding the H-mode power threshold in mixed-species hydrogen plasmas is of particular interest. Experiments from JET have shown that the H-mode threshold behavior in mixed H–D and H–He plasmas is non-linear. AUG identified the detrimental effect of H in He plasmas, but the impact of adding a heavier species (He or D) to H plasmas is more complex. JET results indicate a significant reduction in the H-mode power threshold with 10%–20% concentrations of D or He in H plasmas. In contrast, AUG does not observe this

effect at these concentrations, while DIII-D sees substantial effects at 20%–25% He concentrations. This reduction in the H-mode threshold by adding He could potentially be beneficial for expanding the operational space for the sustainment of ITER H-mode operation in H-dominant plasmas.

In general, P_{LH} deviates from a near-linear increase with density above a certain density ($n_{\text{LH,min}}$). Below this density, P_{LH} strongly increases as density decreases, opposite to the predicted trend. This determines the optimum density for H-mode access, which is highly relevant for the SRO phase due to the reduced margin above the L–H threshold with the available heating power in this phase. The physics behind this minimum value is not fully understood but is believed to involve changes in turbulence with collisionality and stabilization by radial electric field shear, which depends on temperature and density gradients and the sharing of edge power flux between ions and electrons.

According to a model based on AUG, the L–H transition threshold is determined by edge ion power fluxes. In electron-heated plasmas like ITER, inefficient equipartition at low densities increases the required heating power for a given edge ion power flux, thus determining the minimum H-mode access threshold. This model aligns well with findings in some tokamaks, like C-Mod, but is less clear for others like JET, where impurities may play a role. The turnover density and power threshold near this density can depend also on divertor configuration, impurities, ion species, and plasma rotation.

JET conducted a series of experiments using ^4He plasmas with the aim of improving the understanding of H-mode physics and investigating their suitability for low activation phases in future tokamak designs, such as ITER. The experiments used Ar frosting of the cryopumps, which allowed ^4He to be pumped efficiently. The auxiliary heating was provided by NBI, with sources converted to ^4He , providing up to 12 MW of heating. The experiments showed that the L–H power threshold in ^4He has a similar dependence on density and field as for deuterium plasmas, but with an absolute value $42 \pm 10\%$ higher than for a set of D reference discharges, which motivates the multiplier 1.4 applied to $P_{\text{LH,D}}$ for He plasmas in the IRP. The L–H power threshold in He was shown to be lower than that in hydrogen [159].

In experiments carried out in a recent JET-ILW ^4He campaign it was found that access to type-I ELMy H-mode required significantly higher power compared to the L–H transition power. Consistent observations were made also in a recent AUG ^4He campaign in which type-I ELMy H-mode was not accessible with the available power [160, 161]. From these results, one could conclude that there may be no clear advantage of using ^4He rather than H plasmas for testing ELM control, which is one important reason why the operation of ^4He plasmas is not considered with the recent updates to the IRP.

3.3.2. Heating & current drive. H-mode operation in SRO is an important risk mitigation measure in the IRP. It allows early determination of power requirements for H-mode and the commissioning of ELM and W control schemes. Establishing

H-mode scenarios in SRO will be needed to directly assess the impact of TBMs in early FPO on H-mode performance [162] and optimize correction schemes that will require in particular $n = 1$ error field compensation [163]. TBMs in ITER are experimental breeding modules designed to test tritium breeding concepts for future fusion power plants [164, 165]. Installed in dedicated ports of the ITER vacuum vessel before the start of FPO, they enable studies of tritium production, material behavior, and thermal–hydraulic performance under real fusion conditions, while they might influence plasma performance, as they include ferromagnetic structures giving rise to error fields. While the effect of field errors produced by TBMs [166] on ITER FW loads due to fast ions is expected to be small [167], EFCCs and possibly ELMCC will need to be used for TBM-induced error field correction requiring $\leq \sim 10\%$ of the correction coil current design limit [168], and efforts will need to be undertaken to re-calibrate error field correction, which may be an on-going effort, as the optimum error field correction that is linked to the susceptibility to remaining error fields depends on machine and also plasma conditions such as the plasma beta that may change with time as the experimental program advances, but could be facilitated due to innovative, disruption risk minimizing error field identification methods, e.g. involving stabilization of field error induced magnetic islands by ECCD, that may be partly applicable also in early commissioning stages with low power low beta plasmas in SRO [169]. With limited power from the ECH system (30 MW, though 40 MW of EC heating are foreseen to be made available in accordance with the update IRP), and the ICH system (10 MW foreseen with the updated IRP for SRO), H-mode operation in SRO would only be possible at low plasma current and field (e.g. 5 MA/2.65 T with D).

To minimize the required power, low plasma densities (40%–50% of the Greenwald density) will be explored. These plasmas can be further studied in early FPO at the same and higher power levels with different heating mixes that will be possible with additional heating power of 33 MW NBI [170], 20 MW (27 MW optional) of ECH and additional 10 MW of ICH becoming available before the start of early FPO [135]. H&CD sources to be expected in previously considered 7.5 MA/2.65 T helium plasmas with H sources have been studied in [171], showing that a significant NBI-ICRH synergy may be expected that could reduce the neutron rate due to $^9\text{Be-p}$ reactions by one order of magnitude. Additionally, ECH absorption studies have shown that parasitic X3 absorption at the plasma edge may be present but could be mitigated by adjusting the magnetic field or wave polarization. A study has been conducted to assess the feasibility and sustainability of H-mode scenarios at 5 MA and reduced field of 1.8 T that has been considered in previous versions of the IRP for H or He plasmas in ITER, considering all physics and technical factors, and to define the parameters of such plasmas.

The main issue with H&CD application is the ECH of 1.8 T plasmas, as the ITER gyrotron frequency of 170 GHz corresponds to 3rd harmonic absorption. Initially, heating ohmic plasmas at 1.8 T with the 3rd EC harmonic is inefficient, since the plasma temperature of 2–3 keV is too low for effective absorption, as evaluated by the GRAY code.

To determine a viable scheme for 3rd harmonic heating of 1.8 T plasmas, simulations using the JINTRAC code were performed. These simulations accounted for the self-consistent interplay between EC deposition and plasma kinetic profiles. They showed that the central plasma temperature rapidly increases after EC power application, leading to a positive feedback loop of increased absorption and further temperature rise. This results in very high 3rd harmonic absorption within 1–2 s for EC power of 10 MW or higher in L-mode 5 MA/1.8 T plasmas. Predictions for 3rd EC harmonic heating in the conditions expected in 1.8 T ITER plasmas have recently been validated experimentally at AUG [172] and DIII-D.

3.3.3. Fuelling. Due to operation at low field and higher electron temperatures in low activation scenarios, which are optimum for the $E \times B$ drift displacement of pellet-ablated plasmoids, the fuelling efficiency for fuelling pellets that are injected from the HFS is predicted to be high in low activation phase scenarios, despite peripheral pellet ablation [173]. As plasmas may need to be operated at low densities, pellet fuelling is however only considered as a complementary fuelling method in addition to gas fuelling to achieve high density plasmas close to the Greenwald density limit with NB and possibly IC heating. The low activation phase plasmas can only be fuelled by H and D pellets, while the supply of He will need to be done by gas puffing only. For that reason, the achievable density in He or He-dominant plasmas will be limited to the maximum achievable He puff rate that can be applied without divertor transition to detachment. Even though the time scale for the mixture of different isotopes in the core plasma is known to be much smaller than the particle confinement time [174], pellet fuelling might also be beneficial to achieve a fast change in the core isotope composition in mixed H+He plasmas by injection of H pellets.

Even though the ITER cryopump performance is similar for H versus He plasmas, density control may be more delicate for the latter, as divertor detachment could be reached more easily due to enhanced radiation of He versus H in the SOL. On the other hand, due to the larger He radiation fraction, operation at higher power at the same density could be achieved in He plasmas without impurity seeding for divertor control.

3.3.4. Isotope dependence on transport. While a clear negative isotope mass dependence of the L–H threshold has been observed in the experiment for pure hydrogenic plasmas, the impact of the main ion composition on core transport and confinement has been found to be more complex in agreement with experimental observations [175–177]. While transport conditions primarily depend on A_{main} and on plasma properties that are dependent on A_{main} , it has been ascertained that transport diffusivities do not follow the main ion mass dependence that would be expected from a simple gyroBohm type description of the transport. Even though local transport was found to follow the gyroBohm dependence in the electrostatic collisionless no rotation limit [177], a deviation from this dependence towards less degraded or even improved transport conditions

with increased A_{main} can occur in conditions with significant $E \times B$ shearing, collisionality, high beta effects and A_{main} dependent non-linearities introduced by turbulence saturation by zonal flows and fast ions. In addition, non-local effects on confinement are thought to be present originating from the plasma edge and whose strong A_{main} dependence impacts global confinement properties. The impact of the plasma edge was found to be particularly important in H-mode in which a strong positive correlation between the mass number and the confinement has been ascertained. For ITER, the predicted impact of the isotope effect is favorable in view of operation in D and DT at later stages of the experimental campaign for which a confinement improvement can be expected compared to operation in H.

At JET, the energy confinement has been observed to be significantly lower in He giving lower normalized pressures compared to hydrogen [159].

During hydrogen operation, the transition to robust type-I ELMy operation is unlikely, but good H-mode confinement is expected for the He and D cases. Supporting research on operation at densities near the minimum power threshold will need to be carried out during ITER construction to clarify the best scenarios during the low activation phase for plasma system commissioning and ITER licensing [178].

Both type-I ELMy H-mode and type-III ELMy H-mode regimes were explored in ^4He , which showed similar ELM crash characteristics and occupied the same region of loss power space relative to the L–H threshold. However, the ELM frequency increased with increasing power for type-III ELMy H-modes in ^4He , unlike the D trend. The energy confinement in ^4He was found to be lower than in D, with $H_{98y2} = 0.77 \pm 0.04$ for type-I ELMy H-modes and $H_{98y2} = 0.46 \pm 0.05$ for type-III ELMy H-modes in ^4He plasmas. JET's experiments demonstrated that type-I and type-III ELMy H-modes can be sustained for many confinement times in He [159].

3.3.5. Neutron yield in SRO/early FPO phases. For the SRO phase, where plasmas are only heated by ECRH, the only source of neutrons is deuterium in D plasmas and the residual deuterium in the natural hydrogen, and the magnitude of the neutron source is predicted to be low. For the early FPO phase, where ICRH and NBI will also be available, neutron production rates for hydrogen plasma conditions to be explored were estimated to be in the range of 10^{12} – $5 \times 10^{14} \text{ n s}^{-1}$, assuming that the plasma is contaminated by Be due to assumption of a Be coated main chamber wall with previous versions of the IRP, reaching Be concentration values of up to 10% [179].

JET measurements have provided experimental evidence of the significance of nuclear reactions between Be impurities and fast particles produced by auxiliary heating schemes. Neutrons produced by reactions with Be during ^3He minority heating were observed in [180], while evidence of ^9Be nuclear reactions during second-harmonic-majority and hydrogen-minority ICRH in JET-ILW hydrogen and deuterium plasmas is described in [181]. The magnitude of neutron-producing reactions in the low activation phase of the

IRP, where hydrogen and helium plasmas were foreseen to be explored has been evaluated in [179].

For DD discharges in FPO, the neutron and tritium yield could reach 10^{21} particles per shot with a current flat-top duration of 400 s [178].

For the early FPO phase with ICRH, the dominant sources of neutron production are the direct reactions of ICRH-accelerated minorities with beryllium if present. The production of neutrons in early FPO is also strongly affected by local plasma parameters such as temperature. For the commissioning of the NBI and ICRH systems, this production could be decreased by carrying out commissioning in L-mode plasmas at high densities, and thus lower temperatures [179].

Also with the updated IRP assuming operation with a W wall from the start of SRO, small but relevant neutron fluxes may be present and need to be taken into account already in the low activation phase. The neutron yield that may be achieved in experiments in D in SRO needs to be minimized to limit the time required after the D campaign for the in-vessel radiation to decay to a level allowing human access to the vacuum vessel for the installation of system upgrades and the execution of maintenance work by the time when SRO shutdown starts. To ensure this goal, the neutron fluence of the D campaign in the updated IRP is limited to 1.5×10^{20} and the campaign is set to end, at latest, more than ~ 10 months before the end of SRO. SRO experiments after the D campaign will be performed in H plasmas and thus with very low neutron production.

Likewise, to minimize neutron production, first experiments with T-containing plasmas in early FPO are foreseen to be carried out in hydrogen to enable human intervention in case that PFCs would need to be renewed after disruption mitigation tests with mitigation of runaway electrons from beta-decay seeds. The neutron production in H+T is very moderate, as the energy threshold for proton-tritium reactions to occur is > 1 MeV.

Also in later FPO with DT plasmas, the neutron fluence will be limited to enable the performance of maintenance activities in the corresponding long-term maintenance periods, while respecting shutdown dose rate requirements for workers; this fluence is evaluated to be 3.5×10^{25} , corresponding to $\sim 1\%$ of the ultimate ITER Project fluence goal [134].

3.4. Integrated modeling predictions of ITER low activation phase scenarios

Scenarios for H/He operation in the ITER non-activation phase have been developed and studied by means of integrated transport modeling codes in [182] with a focus on He H-mode operation at 7.5 MA/2.65 T. A comparative study using CORSICA and TRANSP has shown that this could be a good candidate for studying H-mode operation in the low activation phase. Various scenarios have been studied for different currents and fields to develop operational paths from low-current/low-field operations to full-current/full-field operations. The EPED1 model and ITER-specific derivatives [183, 184], considering also EPED1 predictions specific for SRO operation, have been used for the modeling of the maximum achievable pedestal

height considering ideal MHD constraints. Sensitivity studies have shown that the development of operation scenarios might depend significantly on the assumptions made about various inputs which will need to be continuously reassessed. Further studies are also needed to improve the understanding of plasma pedestal conditions, core confinement, and impurity concentration [182].

The feasibility of running ITER at 5 MA and 1.8 T, in particular with respect to the actually achievable pedestal height that may be limited by transport processes and remain below the limit imposed by MHD constraints, has been analyzed through integrated modeling to see if it would allow for early H-mode operation in the IRP. To address concerns related to the absorption efficiency of each H&CD system, specific modeling of the H&CD sources was conducted. The results showed that operating the EC system with the 170 GHz baseline gyrotrons is sufficient for successful operation of these scenarios and for providing early H-mode access in SRO. The modeled H-modes did not seem to have specific problems with core/edge integration, despite the low plasma densities foreseen. However, the need for ECH-assisted start-up is an open issue, and operating with an increased gap would not be compatible with the application of high IC power. ICRH modeling shows that the most efficient heating scheme involves second harmonic H heating in both hydrogen and helium plasmas, leading to efficient IC absorption even at relatively low H concentrations for helium plasmas. NBI modeling results show that NBI FW power fluxes, including ripple losses, remain within acceptable limits for the stainless steel shield blocks [185] that support the FW panels and that may be reached by the neutral beams through a gap between adjacent FW panels [186], determining the limit for NBI shine-through losses. Overall, operating at 5 MA/1.8 T in ITER is predicted to provide the required H-mode plasmas to successfully fulfill the ITER R&D goals for the early phases of the IRP [187].

An assessment of plasma conditions for optimum H-mode access and sustainment with ECH in the SRO phase of the IRP with the use of hydrogen and helium has also been carried out in [188]. The study predicts that a 20 MW ECH heating level would allow H-mode sustainment of 5 MA/1.8 T H-mode plasmas with the existing scalings and their physics-based extrapolation. However, it is only viable for helium plasmas and with a reduced operational space. The study further reveals that increasing the heating power to 30 MW would significantly open up the operational space of H-mode plasmas in helium and also allow the exploration of H-mode operation in hydrogen or hydrogen dominant plasmas.

H-mode plasma properties have been evaluated for hydrogen operation by simulating central electron temperatures in excess of 10 keV with ion temperatures ~ 2 – 3 times lower for H-mode plasmas with $n_e = 0.5 n_{GW}$, for which H-mode sustainment is predicted to be possible with the Martin scaling for P_{LH} [157]. The simulations indicate that a difference between electron and ion temperatures is due to the central ECH of these plasmas depositing all additional heating power on the electrons and the inefficient equipartition in the central region of the plasma due to the low absolute value of the plasma density.

It has been concluded that the H-mode plasma features observed in later phases of the IRP are present in hydrogen H-mode stationary plasmas with ECH. Thus, they provide a good basis for the development of H-mode operation in ITER, even though further R&D is required to refine the quantitative evaluations of H-mode access and sustainment, the transport processes expected in these initial H-mode plasmas in ITER, as well as core-edge integration related open questions. The models in use for the prediction of the H-mode plasma behavior in SRO will need to be refined to take into account different isotopes and ion species, which are known to have non-negligible effects on H-mode access, pedestal stability, and H-mode confinement according to experimental findings [188].

Results from integrated core-SOL/divertor transport modeling with JINTRAC for ITER SRO and early FPO are presented in [189]. The goal of these studies is to avoid deep detachment, which was often the limiting factor in reaching the desired densities. The simulations show that pellet fuelling is the most efficient way to increase plasma density in L-mode plasmas and that it will be required to provide unrestricted NB operation in 15 MA/5.3 T hydrogen L-mode plasmas to avoid full divertor detachment. For He plasmas, gas fuelling alone is found to be sufficient to provide access to H-mode conditions at 7.5 MA/2.65 T by reducing the shine-through losses to operationally acceptable values through the associated density increase in the L–H transitions [189].

3.5. Outstanding issues

Despite significant progress in establishing the physics basis for the ITER low activation phase scenarios through IRP preparation and associated R&D efforts, further investigations may still be necessary to refine scenario projections and modeling. Some uncertainties persist, as empirical extrapolation from current experiments to ITER may sometimes be unreliable, and predictive plasma models have not yet achieved the level of validation required for quantitative ITER predictions with sufficient accuracy. Some key areas relevant for SRO that may require particular attention are summarized in the following.

One crucial aspect is the characterization of impurity transport in H-mode plasmas with ITER-like low collisionality low gradient density pedestals. W wall fluxes may be significant also in SRO plasmas with the introduction of tungsten walls with the recent updates to the IRP, making W screening an important factor in SRO plasmas. Additionally, means for the optimization of the plasma shape increasing the distance to the wall to minimize W wall erosion and plasma contamination may be important while ELM control systems are being commissioned in ITER SRO H-modes.

Another area requiring further study is the access to and confinement of H-mode in deuterium plasmas under conditions of electron heating and low plasma density which may be impacted by relatively low edge ion heat fluxes. Since these conditions will characterize the first SRO H-mode plasmas, a

deeper understanding is necessary. Similarly, the effectiveness of ELM control using three-dimensional fields at low input torque remains a question to be addressed. Given that SRO plasmas are expected to operate with low torque, this may limit the capability to suppress ELMs effectively.

Moreover, validating suitable pedestal models for the prediction of the pedestal characteristics in ITER low activation scenarios may be important, particularly concerning isotope effects on pedestal transport and stability and predictions of the change in electron versus ion temperature pedestal gradients in electron heat dominated plasmas (see chapter 3). Finally, integrated core-edge-SOL modeling must be validated against experimental data from ITER low activation scenario relevant plasmas to ensure accurate scenario predictions.

3.6. Conclusions

During the ITER low activation phase, the ITER tokamak will demonstrate the operation of required systems to achieve the ITER project's goals including the commissioning of the PCS, diagnostics, H&CD, and fuelling systems for routine operation up to 15 MA/5.3 T in L-mode and approximately 7.5 MA/2.65 T in H-mode. Plasma scenarios have been foreseen to be developed to meet operational requirements, including core-edge plasma integration involving ELM and divertor power load control. The experimental program has been foreseen to be primarily carried out with H and D plasmas. He or H+He plasmas are not foreseen to be carried out with recent ITER re-baseline but could be re-introduced if required. In preparation to DT operation, the scenarios would then be re-established in pure D and in H with T mixture in the early FPO phase, benefitting from new components, in particular additional auxiliary heating capabilities, that will have been installed in the post-SRO assembly phase.

The SRO and early FPO low activation phases enable integrated operation of ITER with its final baseline configuration, including an initially cooled tungsten wall, to mitigate risks associated with physics uncertainties and plasma operations before the later FPO phases with DT plasmas. Critical control and protection systems, as well as the DMS, will be commissioned with plasma, exposing in-vessel components to peak electromagnetic loads to identify and address potential design weaknesses through corrective maintenance. Additionally, the SRO and early FPO phases focus on developing key plasma scenario components and optimizing vacuum conditioning and tungsten impurity control strategies to ensure smooth and reliable operation in later FPO, where in-vessel modifications will become significantly more challenging due to operation under the full nuclear licensing rules and activation.

The successful preparation involving dedicated R&D as carried out within ITPA IOS TG and the operation of SRO and early FPO are critical for establishing and confirming predictions made for the IPB. This will provide a solid foundation for the execution of later FPO in D-T, ensuring the achievement of ITER's objectives.

Table 3. Variation of ITER baseline performance parameters at $I_p = 15$ MA, 13.5 MA, and 16.5 MA at fixed Greenwald fraction = 0.85. IPB98(y,2) scaling with $H_{98y2} = 1$, $Q \sim 10$ has been assumed. Reproduced from [192]. © EURATOM 2018. All rights reserved.

I_p (MA)	15	13.5	16.5
q_{95}	3 ^a	3.33	2.73
f_{GW}	0.85	0.85	0.85
n_e (10^{19} m^{-3})	10	9	11
$W_{th}/(W_{th} \text{ at } 15 \text{ MA})$	1.0	0.80	1.26
β_N	1.8	1.6	2.06
P_{fus} (MW)	500	317	791
P_{heat} (MW) (including α -heating)	150	113	208
Q	10	6.33	15.8
H_{98y2} required to obtain	1.0	1.115	0.88

^a The q_{95} values in the two simulations provided for ITER range from 2.85–3.27.

4. Baseline scenarios

Coordinated by Sang-hee Hahn

Since proposed in the IPB [190] and the PIPB [191], the IBL operation scenario is described as a plasma target obtainable at ITER with fusion gain $Q \sim 10$ using a discharge at plasma current $I_p = 15$ MA, toroidal field of 5.3 T with a safety factor at 95% flux, $q_{95} \sim 3$, $\beta_N \sim 1.8$, and density $n_e \leq 0.85n_{GW}$, where $n_{GW} \equiv I_p/(\pi a^2)$ the Greenwald density limit. The nominal baseline parameters were defined as table 3 under the hardware cooling system duration limit ~ 3000 s.

The operational status of ITER has remained as under construction during the period (2008–2025) addressed in this paper. Consequently, any discussion or analysis of IBL discharges within this time frame necessarily falls within the domain of predictive modeling rather than experimental observation. The approach of prediction can be described into two categories: One is extrapolations through the JEXs on the scaled baseline discharges on the existing devices to archive measurements and scaling. The other is theoretical modeling, based on known physics models, to design and simulate what we expect how the $Q \sim 10$ plasma would look. It should also be noted that the work mentioned here has been done focusing on the Be/W configuration, before the decision was done to transition to full tungsten walls in 2023. Such changes on the primary PFC will require further reevaluation of the scenarios using these predictive modeling approaches.

Hence this section is organized as following: section 4.1 mainly explains the approaches on the modeling prediction and their limitations. The next section 4.2 summarizes notable results on the related JEXs and the corresponding JAs are presented in section 4.3 collected under the ITPA IOS group. The section 4.4 tries to put key characteristics for anticipated predictions and discusses on the predictive limitations. Finally, section 4.5 mentions about the remaining issues for predictive modeling, leading to the section conclusion 4.6.

4.1. Issues/topics

4.1.1. Predictions for $Q = 10$ baseline. In order to imagine how the $Q = 10$ plasmas would look like, the modeling

prediction approach for IBL access falls into two groups: the first group uses relatively simple models to describe a whole span of the discharge. The span often includes start-up, current ramp-up, current flattop to burn, and scheduled termination of a plasma discharge. The simulation efforts in early 2010s were focused to this area, using mostly 0D and semi-empirical models in order to identify anticipated machine engineering limits. The other group tries to assess specific physics issues using high fidelity physics assumptions. While these models can address specific physics questions, their applicability is usually limited to short time windows and sometimes to single time slices.

The first group, for the predictions to identify machine limits, typically uses semi-empirical models for transport such as Coppi–Tang [193, 194] or BgB transport model [195]. They also assume analytic profiles for the H&CD sources and prescribed CD efficiency. A simulation work using TSC [196] examined the available operating space under the earlier design spec of the ITER PF coil systems and suggested an expansion of PF coil operating space in addition to relocation of PF magnets, in order to enlarge the operating margin of the inductive baseline scenario at $I_p = 15$ MA. Later in 2013, work using CORSICA [197] and DINA [52]) indicated the controllability of ITER plasma at 15 MA operating space under the redesigned coils and FW geometry [19]. Examples of achieved baseline scenario simulations including ramp-up and ramp-down is shown in figure 18. In the same work the achievable ramp-down rate was also investigated, showing 50 s of ramp-up and 66 s of ramp-down phases constrained only by the voltage capabilities of the power supply systems, free of vertical instability. The predictions of kinetic properties by those simulations are often limited by the adapted transport models, hence needs of additional validation were arisen.

The other group tries to use high-fidelity physics model: three categories of notable works are shown as examples below.

The first category is a series of work about the H-mode access and its evolution in H-mode [198–200]. A study for the entry to burn and flattop burn control in ITER [198] combines known L–H threshold power scaling [157] and its hysteresis rules with time-dependent integrated simulations by TSC, JETTO/SANCO [201] and CORSICA. The study concluded that a robust entry to burn/relaxation can be obtained by 73 MW injected power under a wide range, and also calculated a marginal condition as 43 MW of injected power limited by impurity radiation.

The second category of work is simulations for particle control. Coupled core-SOL/divertor transport simulations using JINTRAC [200, 202] for fuelling requirements on the ITER scenarios suggest that the pellet injections are most effective for density increase in L-mode than the H-mode, under 20 MW of heating power to safely operate the NB sources in H and DT, implying that the ITER NB commissioning should be possible in H- and He operations. They also mentioned that pellet fuelling could provide up to 85% of Greenwald fraction at the target ELMy H-mode, $Q \sim 10$, $I_p = 15$ MA, $B_t = 5.3$ T, provided the pellet size to be adjusted. Investigation on compatibility of pellet fuelling and gas injection against the JET-ILW gives an

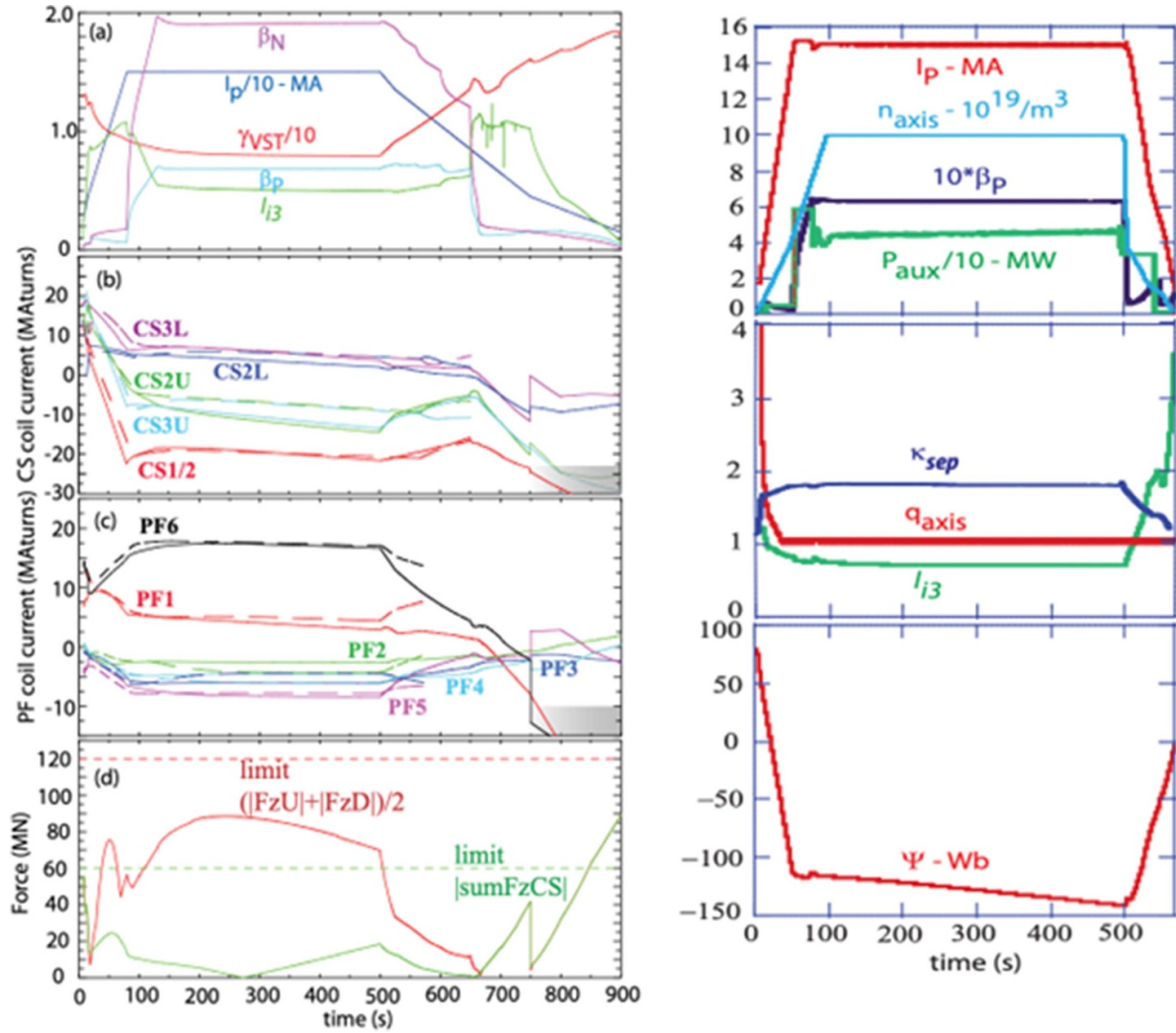


Figure 18. (left) $I_p = 15$ MA baseline simulation by CORSICA, specifying plasma and stability parameters, PF and CS current trajectories and the resulting forces on the CS for the 4.5 keV pedestal case. (right) Prediction of $I_p = 15$ MA baseline scenarios simulation by DINA. Reproduced from [19]. © 2014 ITER. All rights reserved.

insight for the anticipated changes in comparison to the full carbon walls (JET-C) [203].

Another JINTRAC suites simulations on the transition to stationary high Q H-mode regime in ITER are reported in 2017 [199]. The approach and the validation is based on JET experiments. The suite adopts GLF23 [204–206] as the core transport model for H-mode conditions and the BgB is chosen for the L-mode core. The accessibility of $Q \sim 10$ in $I_p/B_t = 15 \text{ MA}/5.3 \text{ T}$ ITER scenario is discussed under various density conditions, including a strategy suggested for low W contamination.

The last one to mention for the second group is an evaluations of tungsten material productions and accumulations through ASTRA [207] and JINTRAC simulations [208]. The work showed that the core W density peaking is expected to be low and manageable under anticipated density gradients for pre-DT and DT, whereas the W accumulation is to readily occur during the termination phase of ITER H-modes unless some optimized heating ramp-down and pellet-fuelling switch-off scenarios are provided.

Cautions are needed to validate those physics models against the experimental data from existing devices. There are experimental data points lacking for tangible extrapolation on IBL from the present experimental results, even from obtained ones at the largest machine. Aside from the issues due to difference of main species/isotopes, i.e. DD versus DT, the expected low torque, dominant electron heating, low collisionality, and the amount of plasma-wall interactions cannot be simultaneously reproduced in any existent machine. Hence the interpretations of the experiments should be cautious, considering which points they can extrapolate to ITER.

There also exist the points that the models cannot completely cover due to lack of certain physics understanding. Transient states with rapid changes of plasma current have no universal physics model that could apply to every device. Lack of MHD cannot explain cause of disruptions. Lack of energetic particle physics cannot correctly estimate the stored energy of the discharge. Lack of heating interaction physics also can mislead temperature or relationship of T_e and T_i . Most

transport solvers do not contain any 2D edge transport models, where the neutral dynamics do matter.

4.1.2. Challenges on predictions. The major challenges in predicting reliable scenarios for the IBL is the lack of a complete set of validated physics models for the operational space of ITER. In specific, the missing models can be categorized in the following list:

- Model for pedestal during transient ramp phases not available.
- Model for pedestal density not available.
- Compatibility of pellet fuelling with SOL conditions and core transport is not certain.
- Thermal and particle transport models are not fully validated in ITER-relevant regimes.
- Modeling of the ramp-up and ramp-down phase has large uncertainties due to a lack of self-consistent evolution of density, H&CD, MHD, and actuator control.

Each of these is not only very challenging but also connected to each other. For example, pedestal density profile is connected to SOL conditions and edge transport. The presence of the control actuators such as RMP, fuelling, and H&CD makes the models more complex.

Fortunately, in the last decade there have been advances on simulation/prediction ability of the burning plasma baseline parameters. Here a few notable works are introduced. As we have shown in 4.1.1, the simulations belonging to the first group are able to simulate the whole span of the discharge, including I_p ramp-up, burn, flattop, and ramp-down. Such predictive modeling suites can simulate not only the inductive baseline but also other ITER scenarios such as the hybrid and steady-state scenarios, by suitable choices of the simulation codes and transport models [209] (see sections 5 and 6).

Modeling work for preparing JET DT campaigns under JET-ILW, especially focused on the optimization of the neutron yield and α -particle physics [210], used recent experimental results from H and D campaigns plus the first DT operation phase (DTE1) results to extrapolate the anticipated DT fusion power per model under maximum input power of JET. According to this work, the fusion power (P_{fus}) was expected to reach 11–17 MW, depending on the transport models used in the simulations. The predicted fusion yield by the work has been validated by the second experimental DT campaign of JET (DTE2) in 2019–2020 [136], by achieving $P_{\text{fus}} \sim 13$ MW in the peak and 9–10 MW on average sustained up to 5 s. The lesson learned in this case is the importance of validating physics models to determine which model can accurately capture the characteristics of the plasma of interest.

Benchmarking of core particle transport simulations was performed in the IBL scenario by a collection of codes widely used in transport modeling of tokamak plasmas [23]. This included formulation of transport equations, difference between electron and ion solvers, comparison of modules of the pellet and edge gas fuelling. At the first phase of benchmarking, the particle transport effects were addressed in the

stationary phase. Firstly, simulations were benchmarked with identical sources, sinks, transport coefficients, and boundary conditions prescribed in the flattop H-mode phase. The transformation of ion particle transport equations was introduced to compare to electron transport solvers. Secondly, the pellet fuelling models were benchmarked in various conditions to evaluate the dependency of the pellet deposition on the pellet volume, injection side, pedestal, and separatrix parameters. Thirdly, edge gas fuelling was benchmarked to assess sensitivities of source profile predictions to uncertainties in plasma conditions and detailed model assumptions. At the second phase, the particle transport effects were addressed in the time-evolving plasma from the current ramp-up to the ramp-down phase. The calculated density profile agrees within 3% in both steps, as shown in figure 19. Differences between the simulation results of the solvers were investigated in terms of equilibrium, grid resolution, radial coordinate, radial grid distribution, and plasma volume evolution term. It is found that the appropriate choice of radial coordinate and corresponding grid node design is crucial for pedestal density prediction, as it enhances the approximation accuracy of the edge source modeling in the plasma edge region. The simulations revealed that electron and ion solvers predict noticeably different density peaking for the same diffusion and pinch velocity while with the peaked profile of helium, expected in fusion reactors. The fuelling benchmarking showed that gas puffing is not efficient for core fuelling in H-modes and density control should be done by the HFS pellet injection in contrast to present machines.

Before going for the next subsection, it should be remarked that most of experimental/simulation researches mentioned in this section have used performance data obtained from discharges with ELM crash events. However, it is widely known that the ITER FW cannot tolerate the heat load caused by anticipated ELM crashes [211]. Moreover, recent research showed that the active control of ELM crashes, through either using the RMP or other ways, results in the different plasma profiles due to increased particle transport through the pedestal [212, 213]. It would be premature to discuss on such integrated controls in this section, since the research has just begun. In order to avoid confusions, we will confine the contents regarding the IBL on the research carried out in a certain period of year 2008–2022.

4.2. Results of JEXs

In this subsection a series of brief descriptions of dedicated JEXs regarding the experimental evaluations of scaled IBL access, physics data obtained through the FW variations, and other notable experiments done under the IOS JEX activities.

The scaled demonstration of the IBL at each existing tokamak-type device has mainly focused on achieving matching β_N , q_{95} , and $f_{\text{GW}} = n_e/n_{\text{GW}}$ to the 15 MA inductive scenario parameters, under available plasma current I_p and toroidal field B_t [192]. The commonly accepted target shape specification per each device has been redesigned to satisfy that (i) the plasma is diverted and bounded by lower separatrix

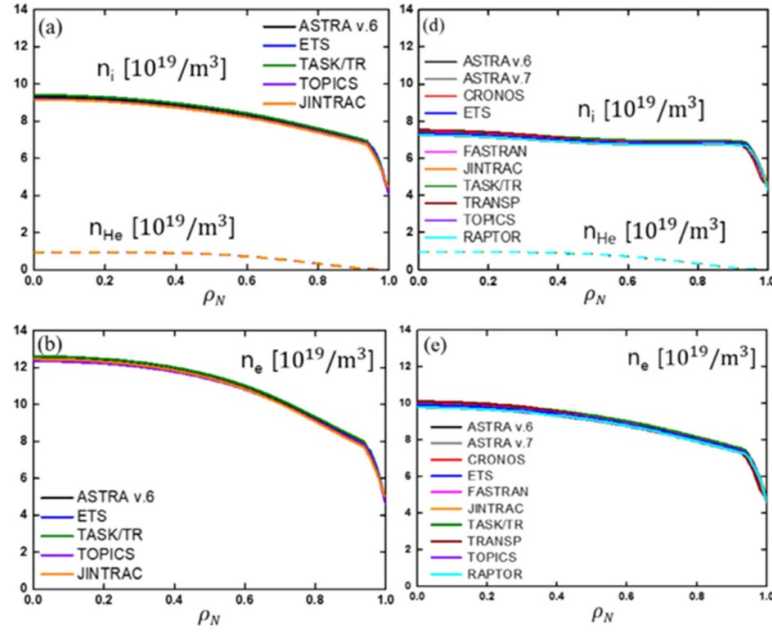


Figure 19. Benchmarking of particle transport simulations on the IBL scenario. Ion (left) and electron (right) transport benchmark, respectively. Reproduced from [23]. © EURATOM 2019. All rights reserved.

Table 4. Experimentally achieved parameters for scaled IBL demonstrations. The ITER $Q = 10$ scenario plan is shown for comparison. Reproduced from [192]. © EURATOM 2018. All rights reserved.

Device	wall	period (year)	I_p [MA]	B_t [T]	n_e [10^{19} m^{-3}]	f_{GW}	H_{98y2}	β_N	P_{NBI} [MW]	P_{ICRH} [MW]	P_{EC} [MW]
AUG CFC	CFC	1998–2006	0.8–1.2	1.4–2.3	2.1–12.4	0.22–0.91	0.6–1.5	2.0	0–12.5	0–6.4	0–1.1
AUG W	W	2008–2014	1.0–1.2	1.7–2.0	7.9–14.3	0.59–1.00	0.7–1.1	1.6	2.2–7.3	0–2.6	0–2.6
C-Mod (H-mode)	Mo	2009–2012	0.6–1.3	2.6–5.4	10.4–59.1	0.19–0.84	0.5–1.0	0.5–1.0	—	0.8–3.6	—
DIID-D	C	1996–2014	0.6–2.0	0.6–2.2	1.3–11.7	0.11–1.13	0.5–1.5	2.0	0–15.5	0–1.3	0–3.5
JET-C	CFC	2000–2009	1.0–4.0	1.0–3.6	2.0–33.9	0.24–1.24	0.5–1.5	1.0–3.7	0–22.9	0–10.4	—
JET-ILW	Be/W	2012–2022	1.4–4.0	1.3–3.7	2.9–9.5	0.4–1.1	0.6–1.2	1.0–2.8	1.1–25.9	0–5.4	—
JT-60U	CFC	1999	1.8	3.1	2.4–4.5	0.3–0.53	0.7–1.1	1.0–1.2	5.9–10.6	—	—
KSTAR	C	2017–2021	0.85–1.1	1.8–2.5	6.0–10.0	0.45–0.61	0.9–1.0	1.5–2.0	3.5–5.5	—	0–1.3
ITER, $Q = 10$	Be/W	2015	15.0	5.3	9.6	0.83	1.03	2.0	33	10	6.7

(aka X-point) as ITER is, and (ii) has smaller lower triangularity (δ_L) than upper one (δ_U). A full-bore shape scenario diverting early in the current ramp-up was favored, in order to keep $l_i(3)$ under control [58].

4.2.1. Scaled IBL experiments. Table 4 shows a summary of achieved parameters for each device in comparison with the $I_p/B_t = 15 \text{ MA}/5.3 \text{ T}$ IBL inductive scenario.

JET. Before installations of the ILW in 2011, the set of relevant data has been collected from physics experiments at q_{95} near 3, rather than dedicated demonstrations with integrated parameter matching. In these experiments discharges with $H_{98} = 0.6$ –1.2, $\beta_N = 0.5$ –2.4, and $f_{GW} = 0.25$ –1.1 are obtained [214].

After switching to the ILW, there has been an extensive scans for achieving ITER performances for the IBL and the hybrid scenarios [215, 216]. These first-wall variation results are separately described in section 4.2.2.

The second experimental DT phase, DTE2, was established in 2019–2020 with the JET-ILW configuration. Through the major upgrade of NBI, the best sustained performances for the baseline and the hybrid scenarios established a new fusion power record: P_{fus} reached up to 13 MW under the JET-ILW, and $P_{fus} \sim 9$ –10 MW has been steadily maintained up to 5 s for both the baseline and the hybrid [136]. It should be noted that the successful recipe was obtained via several control techniques such as cautious ELM pacing, led to the sufficiently high frequency ELMs best for tungsten flushing, and low gas dosing for improved pedestal/core confinement. ICRF

was used not only to boost the ion temperature through the known NBI-ICRF synergy, but also to generate the MeV-range ions for a strong source of electron heating, often led to a variety of AEs observation.

DIII-D. The scaled experimental evaluations of all four ITER operational scenarios were reported in 2010 [217]. Using the 1:3.7 scaled lower single null shape, a conventional ELMy-H plasma was achieved at a normalized current $I_N = I/aB_t = 1.415 \text{ MA mT}^{-1}$ matching to that of IBL (15 MA/5.3 T), with $q_{95} = 3.1$, $H_{98y2} = 1.1$, $f_{GW} = 0.5\text{--}0.65$, and the performance figure of merit [11] $G \equiv \beta_N H_{89}/q_{95}^2 \sim 0.37$ [218, 219].

The effect of external torque for the $Q \sim 10$ IBL has been investigated in the separate work later [220]. Dedicated study published in 2018 [221] analyzed the current density, pedestal and rotation profiles from ~ 200 of a collection of scaled IBL discharges obtained at DIII-D, under both high and low torque. The work concludes that the current profile is identified as a key determinant of the disruptive instability to limit the $Q \sim 10$ operation of IBL plasmas. The datapoints are all obtained under full carbon walls.

DIII-D also has developed a setup of the balanced NB torque conditions, using its unique counter- I_p NB feature, in order to provide the access to the low net torque regime expected in ITER. Using this feature, the impact of low toroidal rotation to the MHD stability has been studied [220, 221].

Experiments to measure the low-torque IBL plasma response to $n = 1$ magnetic perturbations [222] show that there is a partial correlation between the $n = 1$ rotating tearing mode locking events and the low-frequency $n = 1$ plasma responses, stronger under lower plasma rotation at the $q = 2$ surface. Simulations by MARS-F [223, 224] reveal the contributions can come from both the ideal and the resistive MHD.

ASDEX Upgrade. AUG is a device that has data for both full-carbon and full-tungsten walls [214]. The full tungsten wall results are promising: Discharges at q_{95} near 3 at 1.1 MA/1.8 T have been achieved using ECRH 140 GHz in X3 mode. Also a discharge using 30 MHz of ICRH at 1.2 MA/2.0 T has been achieved under full W-wall with major reference parameters match ($q_{95} = 3.14$, $\beta_N = 1.96$, $f_{GW} = 0.79$, and $H_{98y2} \sim 1$). More IBL demonstrations have been performed [225] dominantly using NB heating and ICRH, achieving 1.2 MA/2.0 T, $q_{95} \sim 3$, $H_{98y2} \sim 0.85\text{--}1.0$, $\beta_N > 1.8$ under prescribed fuelling. Like JET, the W accumulation can be avoided by controlling the edge particle balance using the boronization/gas puff level. Effects of metal walls on the performance will be revisited in 4.2.2.

It is also suggested that an alternative IBL scenario at $q_{95} = 3.6$ can be practically useful, mainly for the advantages of stationary keeping ELM bursts under very low q_{95} and $\beta_N < 2$ with better confinement. The alternative scenario, with lower $I_p = 1 \text{ MA}$ and higher $q_{95} = 3.6$, results in higher $\beta_N > 2.0$, better confinement factor $H_{98y2} \sim 1.05$, and higher density $f_{GW} = 0.9$ [225].

Alcator C-Mod. Attempts to access the ITER performances are done until the end of the C-mod operation using pure RF heating under ITER-operating toroidal field value $B_t = 5.3 \text{ T}$:

Equipped with molybdenum (Mo) FW, the C-Mod achieved discharges matching the proposed ITER shape under ITER-compatible toroidal field, at $I_p/B_t = 1.3 \text{ MA}/5.3 \text{ T}$ and $0.65 \text{ MA}/2.7 \text{ T}$, respectively. Up to 5 MW of second harmonic ICRF heating and Ne seeding were used to access $\beta_N \leq 2.1$, $H_{98y2} = 0.8\text{--}0.95$, and $f_{GW} \leq 0.79$ [196, 226]. A reduction of volt-second consumptions in the current ramp-up has been observed up to 15% by the core ICRF heating in proportion to the core density. The attempt to maintain $f_{GW} = 0.85$ reduced H_{98y2} to 0.6, which was recovered by reducing the density to $f_{GW} = 0.65$.

KSTAR. Under the full carbon walls, the IBL shape has been experimentally established in 2017 [227, 228] with parameter matching $q_{95} = 3.2\text{--}3.6$, $\beta_N \sim 1.8\text{--}2.0$, $\delta_L \sim 0.45\text{--}0.5$ at $I_p/B_t = 0.8\text{--}0.85 \text{ MA}/1.8 \text{ T}$. It is observed that a series of fierce, large and even disruptive ELM crashes ($f_{ELM} \sim$ a few—several Hz) is generated, especially depending on the creation of long period ELM-free states after the plasma enters the H-mode at the plasma current flattop. The long-period ELM-free state is accompanied by core impurity accumulation, observed in a form of strong visible light at the core, and is prone to occur when the plasma is built to be attached to the outboard carbon wall. Optimization of ramp-up shape control to avoid the outboard touching enabled to mitigate those long ELM-free periods.

It should be noted that the superconducting magnet tokamak devices are likely to provide weaker inductive power than the copper devices, due to their coupling AC loss constraints on the superconducting magnets, the discharge buildup needs to avoid every possible impurity accumulation in order to evade such a long ELM-free period and consequently huge ELM bursts. Recent buildup on the scaled ITER discharges used the start-up scheme of the ‘early diverting’ [229] for achieving a monotonic decrease of q_{95} with lower l_i , and decreasing possibilities of unnecessary low- n tearing mode onset. In this way the established scaled ITER discharges also match to the parameters, $q_{95} \sim 3.2\text{--}3.5$, $\beta_N \sim 1.6\text{--}2.0$ at the higher $B_t \sim 2.4\text{--}2.5 \text{ T}$ (90%–95% of first ITER toroidal field target).

TCV. Investigations of the IBL scenario on TCV was reported in 2021 [230, 231] as a JEX with AUG. The IBL scenario at TCV is mainly characterized by low q_{95} (3.0–3.6), high positive triangularity $\delta > 0.35$, relatively high elongation $\kappa > 1.65$ and $\beta_N > 1.5$, $H_{98y2} \sim 1.0$. The TCV IBL high performance and low q_{95} cases are limited by the occurrence of $m/n = 2/1$ modes. The broad current density profile, induced by density peaking, as well as high elongation, high β_N and low q_{95} combine to lead to more unstable plasma to both classical and NTMs. The TCV IBL can avoid these modes at medium β_N and/or at high q_{95} with X3 EC heating. It was also shown that lower elongation facilitates achieving IBL discharges at high Greenwald fraction.

4.2.2. FW variation. Related JEXs performed on devices with metal wall components are summarized here. It should be noted that not all the experiments are done under the IBL

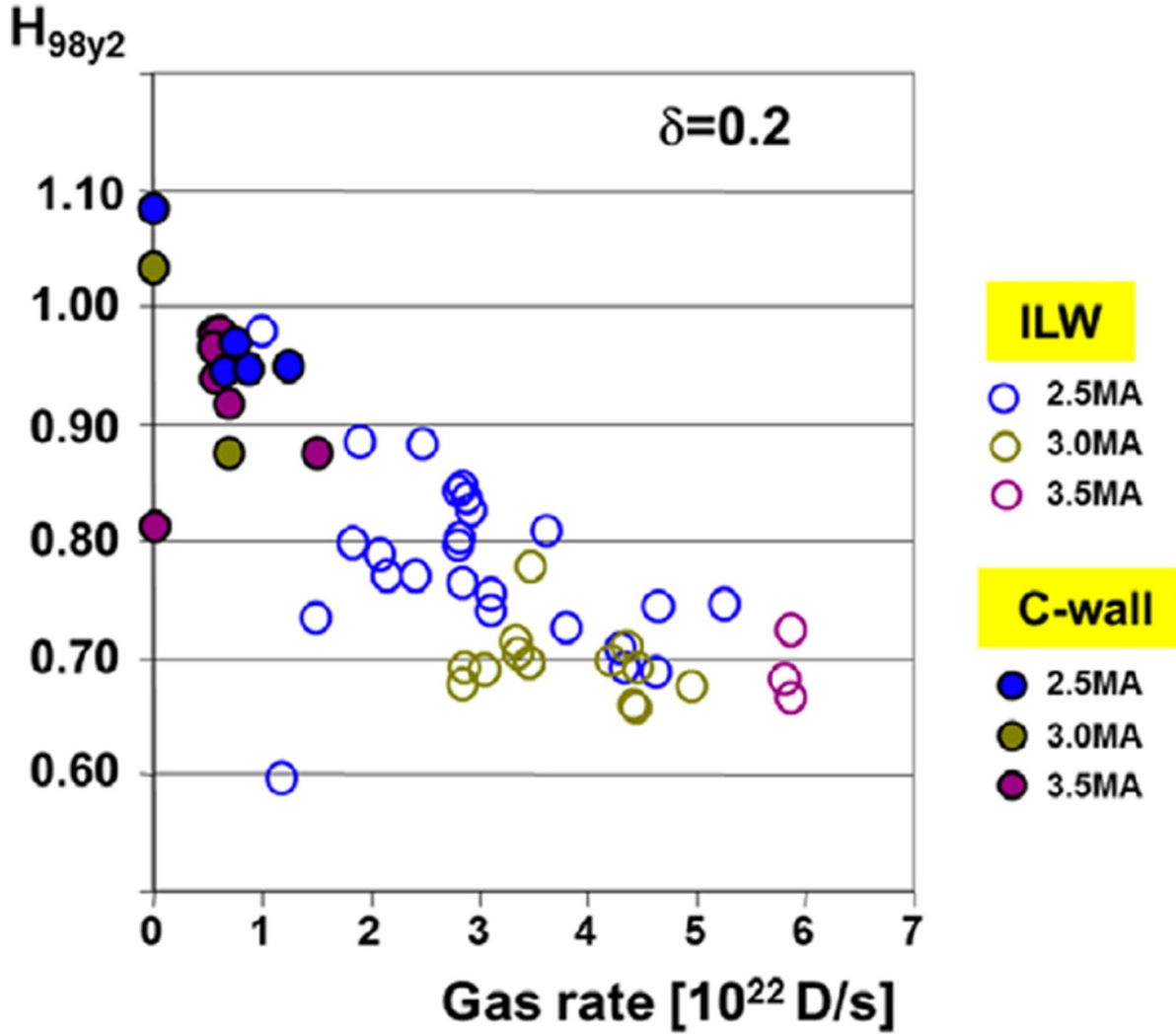


Figure 20. Confinement enhancement factor H_{98y2} versus the gas injection rate for the JET high-current discharges from 2.5 to 3.5MA, showing loss of confinement with JET-ILW. Reproduced from [215]. © 2014 EURATOM. All rights reserved.

conditions, but there are results worth mentioning, particularly in comparison with the traditional full carbon walls.

JET-C versus JET-ILW. The JET experimental campaigns in 2011–2012 after the transition to JET-ILW, consisting of PFCs made of Be in the main chamber and W in the divertor. In the first operation on JET-ILW, a completely different operating domain has been achieved from the one under the carbon walls (JET-C) [215]. Stable type-I ELMy H-modes with $\beta_N \sim 1.4$ was achieved in low and high triangularity ITER-like shape plasmas, however the accessibility to high confinement factor ($H_{98y2} \geq 1.0$) was significantly reduced under JET-ILW due to the large increase of gas injection. Figure 20 shows the observed confinement degradation in terms of H_{98y2} versus gas injection rate for the 3.5 MA JET-ILW plasmas. Hybrid H-mode showed better H-factor than the H-mode, close to those of the JET-C cases. High-Z impurity transport from edge to core seems to be the dominant cause for the observed core radiation peaking.

Detailed comparison regarding the energy confinement in the dedicated H and HD campaign results under JET-ILW

has been summarized in [216, 232]. It is indicated that the decrease of plasma energy confinement is evident, most likely from the strong W accumulations in the core. The reduced pedestal pressure is also observed which generally leads to the change of edge stability. As shown in figure 21, the JET-ILW baseline discharges are more deviated from the line of $H_{98y2} = 1$. The avoidance of the W accumulation in the core requires a special remedy to use ICRH/ECH and to adapt a control of the ELM frequency by known gas injection recipes.

The fusion power (P_{fus}) comparison between DTE1 (JET-C) and DTE2 (JET-ILW) [136], using 50/50 percent of Deuterium/Tritium as main fuel, implies that the performance under the JET-ILW can be steady up to many thermal energy confinement times with higher overall electron density, but lower core T_i and T_e than the carbon wall.

Seeding effects. The heat load controls using impurity seeding have been also investigated for the JET-ILW. Although the operating space of density is changed, the performance change due to impurity seeding was not significant.

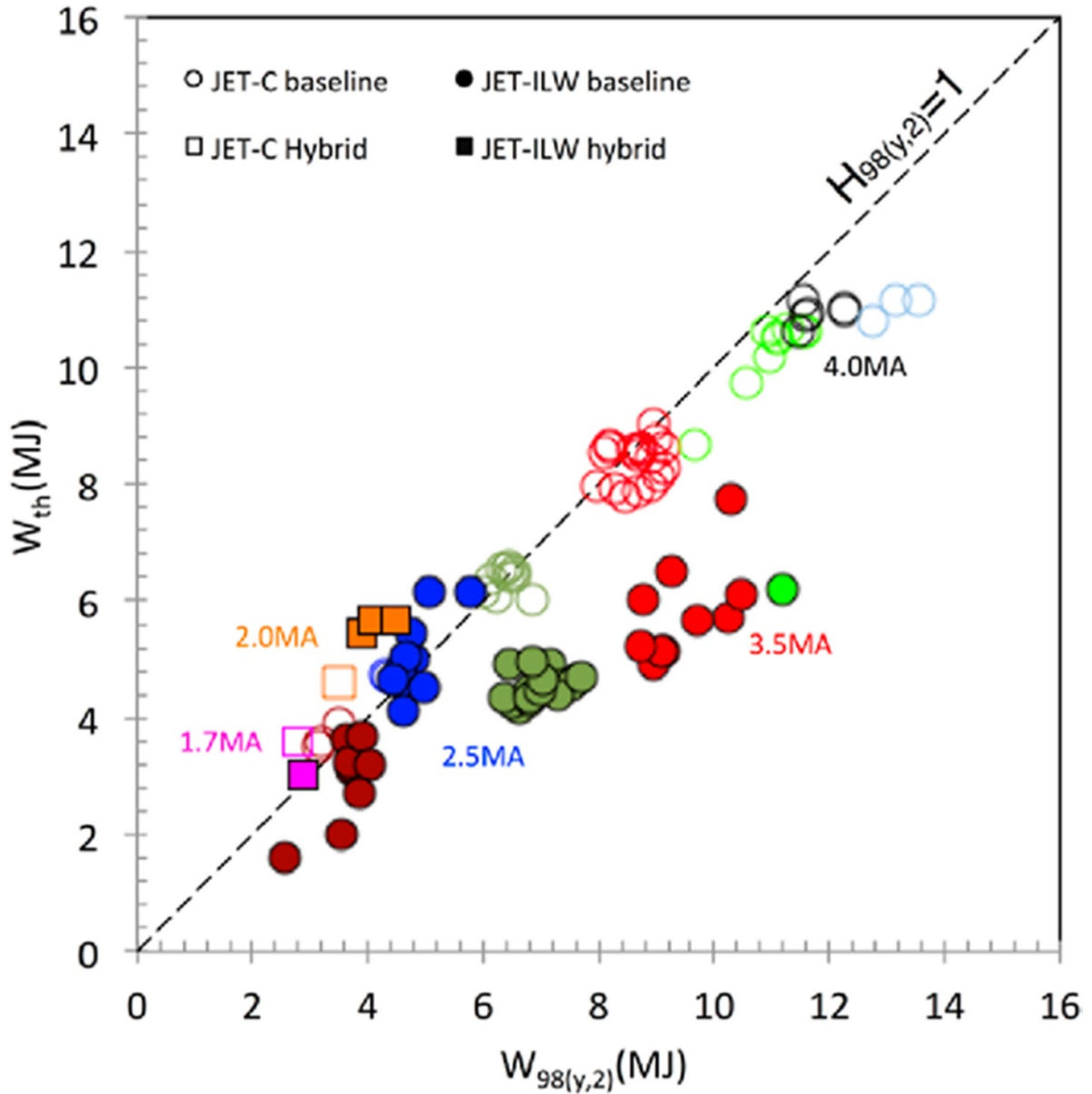


Figure 21. Comparison of energy confinement between JET-C baseline (empty circles), JET-ILW baseline (full circles), and JET hybrid (empty/full squares). Reproduced from [232]. © 2016 EURATOM. All rights reserved.

According to the work in [233] on experimental and simulation approaches to the divertor heat load control by Ne seeding, it was found that the Ne seeding can provide heat mitigation without significant impact on energy confinement, but it is inevitable to lose fusion performances. From simulations of the neutron rate (JETFUSE), impact of Ne seeding increase to the neutron rate is maximized by T_i reduction, whereas the effect of fuel dilution becomes stronger when the thermal reactions dominate.

Recent work done in JET [234, 235] extended the related research to the use of Ne and N_2 seeding under JET-ILW. Demonstration of Ne-seeding compatible with high performance and higher H_{98y2} than equivalent N_2 -seeding. The key of performance improvement is an increase of ion pedestal

temperature, whereas the improved core confinement with higher ratio of T_i/T_e also plays an important role.

Full tungsten wall. The comparison between the AUG carbon (AUG-C) and AUG tungsten (AUG-W) shows similar results as that of JET, indicating that the confinement factor in the discharges under full-W drops to 15%–20% of that of carbon wall, under the same heating and the same β_N , as the Greenwald fraction increases. As shown in figure 22 the change of the FW material—here from carbon to tungsten—modified the confinement scaling regardless of the shape specifications. The AUG team suggests to increase β_N by introducing an alternative scenario operating at larger q_{95} [225].

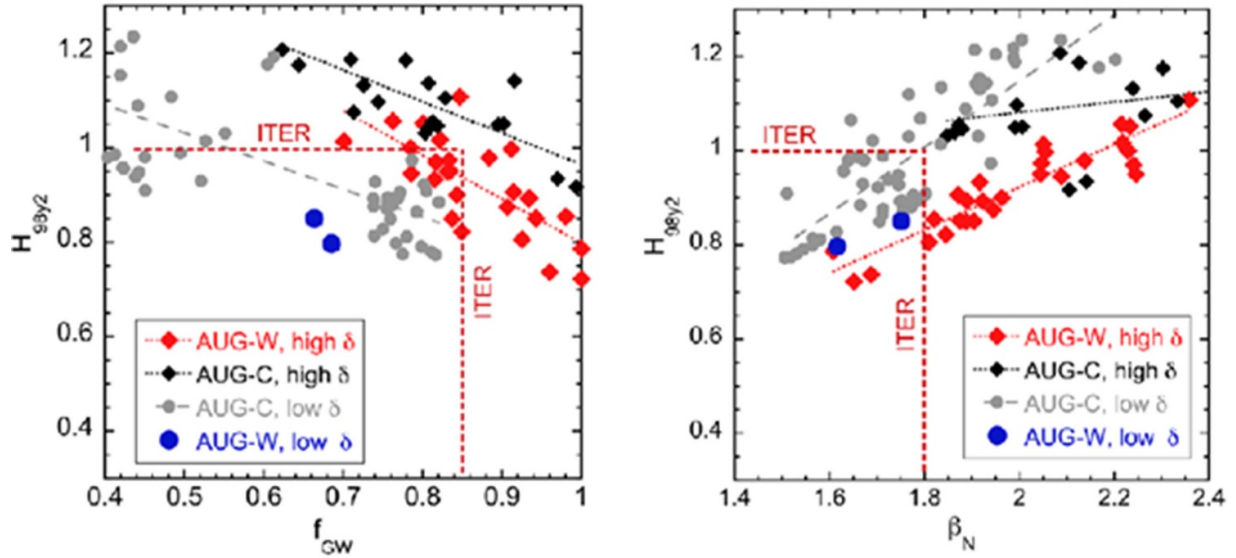


Figure 22. H_{98y2} comparison for AUG-C and AUG-W versus the Greenwald fraction (f_{GW}) (left) and β_N (right), in terms of triangularity. Reproduced from [225]. © 2016 EURATOM. All rights reserved.

Alcator C-Mod. Discharges with elongation $\kappa > 1.5$ and ICRH power of > 1 MW were operated in the period 2009–2012 under molybdenum walls [196]. It is found that a classic ICRF antenna can have correlations with the core molybdenum contamination, as well as the modification of the SOL transport and corresponding density pedestal profiles. A field-aligned antenna was designed to reduce the RF-plasma interactions hence reduce the radiation loss [236, 237].

EAST. Equipped with Molybdenum FW and ITER-like tungsten divertor, improved confinement in the low $q_{95} \sim 3.4$ operation regime, approaching the IBL, is achieved under a quasi-single-null configuration. Key parameters include $\beta_N \sim 1.8$, $H_{98y2} \sim 0.9$, and a central ion temperature of $T_i(0) \sim 3.3$ keV at a moderate line averaged density range of $n_e \sim 3.0\text{--}3.4 \times 10^{19} \text{ m}^{-3}$, with a total injected power of $P_{\text{Aux}} \sim 6.2$ MW, combining $P_{\text{LH}} \sim 2.6$ MW and $P_{\text{NB}} \sim 3.6$ MW. Statistical analysis indicates that the formation of an ITB in the ion thermal channel is strongly correlated with the excitation of fishbone instability in EAST [238].

A summarizing remark for the FW material changes, from carbon to high-Z materials, can be provided here: 1) The normalized confinement factor (H_{98y2}) is reduced under same heating, same β_N , and same density, 2) under same H_{98y2} the Greenwald fraction increases, implying it requires more density and more auxiliary heating power than the equivalent carbon-wall discharges, and 3) prescribed or active controls of impurity accumulation become necessary to avoid excessive radiation loss by high-Z materials. The suggested methods include boronization, impurity seeding, core RF injection, particle controls using gas puff or pellets, and even the creation and regulation through ELM frequency control.

4.3. Results of JAs

This subsection provides a concise overview of the collaborative efforts undertaken to model the IBL scenarios,

encompassing three distinct activities; modeling for the 15 MA scenarios, modeling transport for the metal walls & impurities, and an inter-machine study on collected IBL databases.

4.3.1. Modeling for 15 MA scenarios. As mentioned in section 4.1.1, modeling attempts for the inductive 15 MA scenario was based on the simpler transport models to assess and identify possible machine limits for the tangible construction design. Work using TSC [239, 240] and TRANSP [196] identified ways of modifying magnet configurations at $I_p = 15$ MA for enough operational margin up to $I_p = 17$ MA under available hardware configurations.

Simulation analysis was carried out on the sensitivity of fusion power production by pedestal properties, impurity concentrations, and plasma rotation, using the TRANSP code with boundary conditions from TSC [241]. Later the same simulation suite was used to assess the time-dependent behavior for the entry to burn under varying conditions on injected power level, density & impurity content, and actuator controls [198].

Another combined predictive modeling, consisting of the 1.5D core/2D SOL code JINTRAC and the free-boundary equilibrium evolution code CREATE-NL [242], was used to simulate the IBL 15 MA inductive burn scenario and its variants to assess possible improvement on expected fusion energy output [209]. The JINTRAC suite was also used to develop fuelling scenario requirements for different main species (H, He, and DT) [200], which could be adopted to the planned ITER plasma operation phases using different main species.

The CORSICA code calculates the evolution of 2D free-boundary equilibria including the plasma current profile as well as the surrounding conducting structure currents. Time-dependent simulations using CORSICA have been conducted for investigating the entry-flattop-exit to burn in $Q \sim 10$ plasmas, adding models for the known P_{LH} scaling for H-mode transition and parametrized EPED1 for the pedestal height and

top location [21]. According to the simulations, the $Q \sim 10$ access for 400 s flattop could be feasible at 15 MA in an H-mode ($H_{98y2} \sim 1.0$) with a ~ 50 MW of auxiliary heating power.

The modeling of entry from and exit to burn using JINTRAC integrated suites has been applied for the full predictive simulations relevant to IBL, in particular to analyze the conditions for the fuelling scenarios to ensure a robust achievement of high Q_{DT} regimes in ITER. The validation of the code was done against JET carbon wall data for the transition to stationary H-mode in ITER-relevant conditions ($H_{98y2} \sim 1.0$, low collisionality, comparable evolution in P_{net}/P_{LH}) but different ρ^* and T_i/T_e . Good agreement between the code prediction and the measurement is obtained when assuming that the reduction of the anomalous particle transport through the ETB/near SOL is stronger than that of energy [199]. The work continues the simulations of the L–H transition of the high Q scenarios, in which it is found that the robust burn access is possible under $P_{Aux} \geq 53$ MW by optimizing core and edge plasma fuelling.

A sophisticated modeling study [210] in support of the second JET DT campaign (DTE2) attempted to validate its models, which incorporated several transport models along with CRONOS and JINTRAC, based on H and D experiments at JET. It also sought to evaluate the expected neutron yield for the JET DT campaign. The fusion power estimation largely varied, however, even for JET baseline plasmas, depending on which transport model was adopted. Especially lower values for the BgB model were reported than those for either TGLF [243] or QuaLiKiz [244] model. It is the reason why the validation of transport models is important for prediction; extrapolations to the ITER fusion power would lead to have more uncertainties.

The JA for the ITER demonstration discharge modeling (IOS-JA1) is now an ongoing project aiming the validation and test of theory-based integrated modeling against IBL using existing tokamak database including AUG, C-Mod, DIII-D, JET, and KSTAR. The modeling uses theory-based core transport model (TGLF for all transport channels—particle, energy, and momentum), EPED1 for edge pedestal, and tools for coupled core-edge-pedestal-SOL modeling. Validations over multi-machine database would increase confidence of ITER prediction and identify the modeling gaps/uncertainties in ITER extrapolation. According to the corresponding JA report in 2020, a 0D and profile database for DIII-D IBL discharges has been built with variations of heating power and NB torque. Development of database sharing in IMAS database for DIII-D and JET is ongoing.

4.3.2. Modeling for metal walls and impurity effects. A cross-machine modeling study [245] on N_2 -seeded H-mode discharges and its power exhaust in existing tokamaks such as C-Mod, AUG and JET gives a parametric scaling for the expected divertor radiation power. According to this scaling, the divertor radiation for ITER to be down to 25 MW, depending on the midplane power decay length λ_q .

Divertor heat load control is essential to consider when fuelling scenarios are designed: Another JINTRAC simulations on the fuelling optimizations, without exceeding divertor power density limits, have been performed for the ITER plasma evolution from the diverting to the termination, through coupled core-edge-SOL transport calculations [246]. The simulations span the entire ITER plasma evolution from completion of X-point geometry formation to the termination of discharge, and they show the baseline $Q = 10$ scenario is viable with suggested prescription of the fuelling. It is recommended to take an elaborate ramp-up fuelling scenario to keep the Greenwald fraction restricted, and avoid both the pure ohmic and earlier L–H transition under low I_p .

4.3.3. Study on collected database. Study on operation range of the IBL using multi-machine experimental data collections of global parameters was also performed. During the last decade the IOS TG has compiled a database of stationary H-mode discharges at $q_{95} = 2.7\text{--}3.3$ from AUG, C-Mod, DIII-D, JET, and JT-60U, for both carbon wall and high-Z metal wall experiments with approximately 3300 entries collected [192].

Plasma confinement trends shown in the dataset, dimensionless variables, such as normalized gyroradius ($\tilde{\rho}^*$), collision frequency ($\tilde{\nu}_C$), and beta ($\tilde{\beta}_{th}$), are rather far from them of ITER: The ITER value of $\tilde{\rho}^*$, $\tilde{\nu}_C$, and $\tilde{\beta}_{th}$ are either outside or near the extreme edge of the existing device dataset. The normalized/dimensionless confinement $B\tau_{th}$ defined in [192] shows more Bohm-like scaling $B\tau_{th} \propto \rho^{*-2}$ for many datapoints, but there are points close to gyroBohm ($B\tau_{th} \propto \rho^{*-2.5 \sim -3}$), when the collision frequency approaches the lowest. Thus the pure gyroBohm and electrostatic scaling, $B\tau_{th} \propto \rho^{-3} \beta_{th}^0 \nu_C^{-0.14}$ is better fit with the data than the IPB98(y,2). It should be noted that the $\tilde{\nu}_C$ value closest to ITER is difficult to reach under metal walls, except a few AUG cases following a fresh boronization mimicking the ITER Be/W walls.

The database studies over the H-mode confinement scaling IPB98(y,2) and its residual dependence were analyzed. As shown in figure 23, the inclusion of the metal-wall device data increases the RMS deviation from the original IPB98(y,2) scaling, due to the density increase and consequently the decrease of H_{98y2} down to 0.8. It turned out that the data with $q_{95} \sim 3$ from 5 devices shows a large residual trends in plasma confinement, mainly depending on I_p , R_{geo} , and P_{loss} . The density increase by the metal walls did not positively contribute the confinement improvement.

In fact, the H-factor drop is closely related to the density increase, regardless of the FW material. Operation regime analysis of JT-60U [247] showed that H-factor drops down as the Greenwald fraction increases under the CFC-based FWs plus the W-shaped lower divertor. The interesting point of the data is the degradation of the H-factor is much larger than we observe from the ILWs.

Since the IPB98(y,2) was based on the carbon-wall data only, it might be unwise to heavily rely on the global scaling relations in order to predict the whole span of ITER

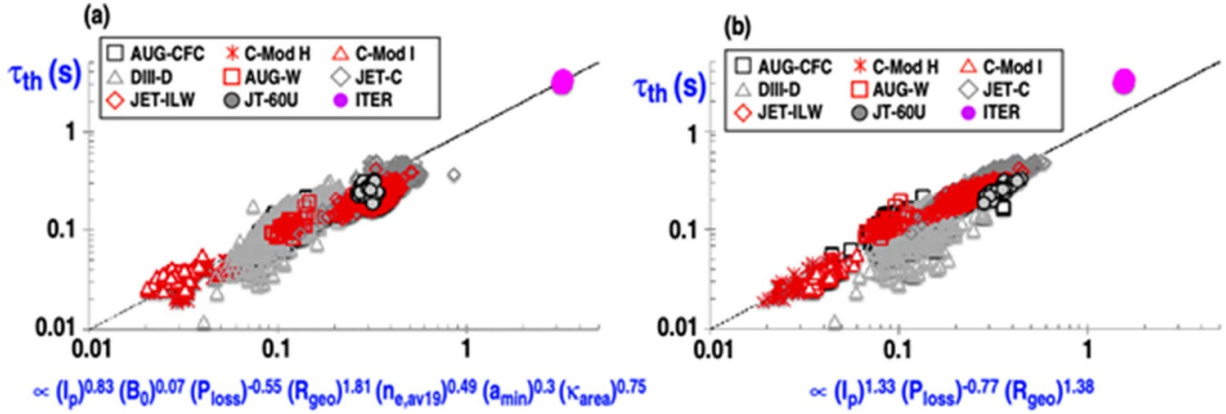


Figure 23. Thermal energy confinement scaling τ_{th} for carbon wall data (Grey points), metal wall data (red points), and ITER exploration (large purple point) using IPB98(y,2) (left) versus DS03 (right). Reproduced from [192]. © EURATOM 2018. All rights reserved.

performance. Even we could create a better scaling for the metal devices, it should be noted that the dataset from present devices shows a plasma with more dense pedestal, lower H_{98y2} , and perhaps lower radiation power than the ones expected from the IBL. The approach through experimentally validated simulations could be more useful to extrapolate the known physics to that of ITER, since the ability of plasma simulations is significantly improved than the time of 1998.

4.4. Prediction for ITER

Here we discuss possible remaining issues, which could improve the accuracy of prediction for IBL. Uncertainties from the extrapolations of the experimental results, reviewed in section 4.3.3, are discussed in section 4.4.1. Key issues desired to clarify for accurate predictions are shown in section 4.4.2. Speculations on major features of the ITER plasmas under the present knowledge are added in section 4.4.3.

4.4.1. Uncertainties for experimental projection. Can the projections, obtained from the partial information from the experiments in the existing smaller devices, reflect the predictions for ITER $Q = 10$ baseline?

According to arguments in section 4.3.3, the answer would be pessimistic. Aside from the smaller collisionality and the larger ion gyroradius, the metal wall experimental results add higher density datapoints that does not exactly follow the known performance scaling IPB98(y,2). It is also doubtful that the creation of a new scaling will help improve prediction. It is found that the pure gyroBohm and electrostatic scaling (denoted as DS03) [248] gives a better fit for the dataset of interest [192], but such approach would have fundamental limitations when any new first-wall material is added.

Would the future fusion reactor use the same material as ITER? If not, how many scalings we should have for ITER and any possible reactors?

Fortunately, there have been advances on the modeling and predictions via simulations in the last decade, as reviewed in the previous sections. Instead of relying too heavily on global scaling relations, predictions based on physics-based models

are recommended to provide better flexibility and adaptability to the variations the device will encounter.

4.4.2. Key issues for ITER prediction. The anticipated uncertainties for simulation approaches would come from the absence or lack of validations, mainly using existing experimental data. However, not all of experimental data would completely cover the operation space of ITER; The missing part of experimental data is not trivial to obtain. Here, we present key issues that should be verified or explained for more accurate ITER predictions, either by another experimental activities or by physics models:

Low torque. The toroidal rotation expected in ITER due to the low input torque extends to the very low level of existing tokamaks, reaching to the level of locking. Since many devices are designed to gain the co-current torque by the NB, the physics of such low rotation is not very popular. Some control techniques that worked with high torque need to be verified and adjusted for lower torque conditions. The DIII-D experiments for extending suppression of ELMs by RMP to ITER-relevant level of beam torque [249] found that reduction of torque from $T_{inj} = 5$ Nm to 3.5 Nm led to the loss of ELM suppression at the ITER level of collisionality ($\nu^* \sim 0.1$). Under their hypothesis explained in the paper, it was claimed that simultaneous regulations of the input torque and the electron pedestal could achieve the RMP-ELM suppression again under a very low torque $T_{inj} \sim 0$ [221]. These DIII-D results imply that specific control techniques available in the high torque provision might not be useful any more in the lower torque regime.

Importance of current profile. Although the low input torque is the cause of the low toroidal rotation, it is not the direct cause of the onset of instabilities. It is found that the $m/n = 2/1$ tearing modes is mainly linked to the edge current density profile, i.e. a current density ‘well’ near the pedestal [221]. Since the IBL requires a certain portion of bootstrap current for sustaining pedestal structure, tailoring the current density profile may become an issue for the stability of the discharge. The same work indirectly mentions that some active controls might be necessary for a plasma with low rotation with high bootstrap

current, rather than relying on the passive stabilization. The plasma responses to $n = 1$ is better explained when both the ideal and the resistive MHD models are involved [222].

Radiation, seeding, and W concentrations. Influences of the W radiation is still one of ongoing research topics. Recent studies indicate that the gas fuelling control is closely related to the resulting ELM frequency and the W transport behavior [250]. Especially, the consequences of uncontrolled W radiation from the mantle is a concern, which will have a dominant effect on the electron power balance. This could cause increase of disruptivity of high-power, H-mode pulses in JET-ILW [251], in particular during the ramp-down phase through either increased radiation in the core or cooling in both the core and edge. Understanding of underlying physics would be important for the prediction of burning plasmas with metal walls, such as the amount of radiated power due to metal content, impurity concentration by low-Z seeding, and anticipated side effects by increased radiation. The recent report from EAST [252, 253], stating that serious W concentration was not observed, is quite promising. However it should be noted that the discharge was operated with dominant electron heating.

Dimensionless parameter space and scaling of confinement. As noted in section 4.3.3, extrapolations from existing devices have inevitable limits. The confinement scalings, originally based on the carbon-wall machine data in [190], does not completely fit to the metal-wall machine results. Moreover, the plasmas obtained from the metal walls show different pedestal profiles and still far from the extrapolation to ITER, especially for the edge collisionality.

ELM control and its own consequences. It is now widely anticipated that the ITER FW would not survive the heat exhaust by the large ELM crashes, the ELM control is ought to be integrated into the discharge controls; the issue is that most of known ELM crash suppression recipes may alter the plasma properties, not only the details of the edge profiles but also the core/global parameters. The external radial magnetic field by RMP alters the pedestal structure to increase the particle transport. The density pedestal changes by the particle transport increase, and in consequence the edge temperature gradient changes as well. Recent experimental control automation using preemptive suppression of the first ELM crashes showed that the preemptive suppression can result in the smaller pedestal ITG [213]. The ELM control effect on the core kinetic profiles is still under investigation.

Pacing methods, such as pellets or impurity seeding, also alter the particle dynamics of the pedestal; pellets temporarily do impact on the pedestal gradients, decreasing the impurity convection across the pedestal and increasing their localization to the mantle. However, the effect of the subsequent small, rapid ELMs on the mantle gradients is to reduce the efficacy of neoclassical impurity screening from the core. There is a risk of disruptions when a large pause occurs due to an unplanned malfunction of injectors [250].

Entry/exit from H-mode: entry to burn, and exit to ramp-down. The biggest issue in this particular topic is that it may

not be valid to use stationary state parameters to those transient situations; the I_p ramp-up prevents the plasma from some premature L–H transition, but it does not guarantee the timing of the entry to burn when it reaches the flattop. The H–L modeling with P_{LH} hysteresis and the termination scenarios with reasonable density model are tricky to achieve, and the off-normal ramp-down scenarios requested for the event handling becomes more tricky due to the possibility of partially losing basic control abilities. It is pointed out that the dynamics of the changes of magnetic and thermal energy, are coupled during the ramp-down, and do not scale the same from present-day devices to ITER [254].

4.4.3. Prediction on key characteristics. At this stage, considering the aforementioned issues, a complete description of the actual target for the IBL scenario may not be fully attainable. Nevertheless, based on the discussions in this section, it is possible to speculate on the key characteristics of ITER plasmas:

- The discharge matches the declared baseline parameters (see table 3), with a performance decrease of within 10% expected due to the metal wall effect. Accordingly, adjustments to the allowable plasma parameter margins and operational details of the scenario will be required.
- The provision of the poloidal flux consumption and the resulting inductive heating lies within the allowed margin. The expected density increase by the metal wall would also increase the required inductive heating power. This might not be a good news for a superconducting magnet device with stronger operable temperature constraints.
- The ELM crashes at the edge are controlled either by completely suppressing them or by mitigating them to keep the heat load on the main divertor within tolerance. The edge pedestal, density and temperature, is regulated under a threshold to keep the particle transport to have no ELM or mitigated ELM.
- The impurity accumulation is also controlled in order to avoid the core radiation that could easily disrupt the plasmas by radiative core collapses. The methodology is not completely identified yet.
- SOB with $Q = 10$ is expected to occur at the 15 MA I_p flattop and should be maintained. Exit from the SOB may trigger event handling procedure in order to halt the discharge and get the safe ramp-down.
- The particle balance will be extremely important for sustaining this discharge, in particular, in the long-pulse realization. Unfortunately, this strongly depends on the hardware specifications, such as the vacuum vessel pumping capability or the fuelling rate of the NB sources. Hence, the balance should be empirically found by the device's own experiments. Data collection from long-pulse machines might be necessary to make any prediction.
- Control of the fusion yield needs to be adapted to some extent to make each discharge efficient; The most efficient control knob is not determined yet.

4.5. Outstanding issues

Here, we add some comments and speculations on the remaining tasks that were not mentioned above but may need to continue research before the completion of ITER construction and operation.

It is apparent that a few properties of the scaled IBL discharges, mentioned and demonstrated in the existing machines in this section, cannot completely cover all the properties expected for real ITER. For example, ITER cannot tolerate lots of ELM crashes under the present design of the FW, but the JEX results shown in this section mainly focused on ELMy discharges which could easily obtain the desired H_{98y2} and β_N . Hence, it is necessary to combine possible controls into the IBL discharges and observe how it changes the plasma profiles.

The consideration of the effect by extending pulse length should not be ignored: experimental results with partial/full W machines showed the reduction of H isotope retention by a factor of 5–10 as compared to the carbon-dominated one. Note that further details provided in section 7 of chapter 5 in this issue, ‘H isotope inventory and recovery’. However, the assessment of the possible changes of the metal wall under high thermal stress, after the exposure to the burning plasma, may require new data obtained from the long-pulse machines. In other words, it is the question how much of confinement and fusion yield degradation would happen due to the changes of the plasma-wall interaction under the time scale. The modeling may be very difficult due to the required time span.

It might be necessary to address which control can be really integrated into the IBL. It depends not only on the requirement of ITER, but also on the availability of diagnostics and actuators. The integration of a few active controls for the particle/heat load on the divertor, as reported at EAST [253], has proven to be beneficial. Integrated simulations would be useful to identify. One of the tasks that should be addressed is identifying necessary diagnostics for the desired controls of IBL, and assessing the required diagnostics for the reactor-relevant regime. More detailed discussions on ITER scenario control needs, aspects, and issues will be presented in section 8.1.

4.6. Conclusions

The definition, predictions, and prospects of achieving the IBL scenario has been explored. The IBL scenario aims for a fusion gain that has never been attained before, at a physical scale surpassing that of any existing devices. Thus any prediction on the IBL requires predictive modeling that has been validated against experimental benchmarks, rather than simple benchmarks or extrapolation solely relying on some linear scaling obtained from carbon-wall based experiments. Recent advances in computational technologies and integrated modeling have enabled not only comprehensive simulations of the entire scenario, but also contributed to more accurate predictions of critical aspects such as the L–H transition conditions and impurity effects. Although the benchmarking results for these models are promising, several key physical phenomena

remain challenging to model yet are critical for the device’s viability.

The experimental realization of scaled IBL plasmas in existing devices during the period of concern (2008–2022) has been highly successful, providing valuable insights into the empirical expectations for the IBL scenario. However, due to inevitable changes of the ITER FW announced in 2023, the introduction of new full Tungsten wall materials necessitates more detailed consideration of material effects in future predictive modeling. Comparative studies between carbon and Tungsten wall experiments reported from various existing devices indicate that achieving $\beta_N = 2.0$ under similar heating resources requires additional optimizations, primarily due to increased average electron density. Consequently, it is increasingly recognized that further adjustments and optimization of other plasma parameters will be essential to meet performance targets under these new conditions. To this end, the development of robust plasma scenarios ensuring both stability and performance appears to be a critical and urgent task through various predictive modeling adapted to the changed environment, alongside experimentally validated optimization strategies.

Prior to the announcement of a new IBL scenario under full tungsten wall, it would be useful to conclude with a reflection on the prediction challenges encountered in the studies of the previous scenario reviewed here. While ITER $Q \sim 10$ plasmas cannot be straightforwardly extrapolated from existing experimental devices, predictive modeling and simulation-based approaches are also subject to challenges originated by physics uncertainties. For instance, the combination of low external torque and tailored current profiles at the ITER scale may lead to unanticipated instabilities, and power balance issues associated with tungsten accumulation could be more severe than expected. In addition, the limited effectiveness of the existing ELM control techniques and known models on entry to burn and exit from burn will be of issue for realistic scenario developments. As a final remark, many of the issues are closely linked to hardware constraints specific to ITER. In this context, experimental data from superconducting devices capable of long-pulse operation under ITER-relevant conditions are particularly important for benchmarking complementary predictive modeling studies.

5. Hybrid scenarios

Coordinated by Yong-Su Na

5.1. Definition and types of hybrid scenario in ITER

In this subsection, first we address the origin of the term ‘hybrid scenario’, which is used to distinguish between conservative hybrid operation and advanced hybrid operation. The hybrid scenario was initially proposed for ITER [12] with a current flattop duration, Δt_{FT} of 1000 s or more. This is achieved by increasing the non-inductively driven current and decreasing the plasma current [35]. The hybrid scenario falls between the inductive baseline scenario (with $I_p = 15$ MA,

$\Delta t_{\text{FT}} \sim 400$ s) and the steady-state scenario (with $I_p = 9$ MA and $\Delta t_{\text{FT}} > 3000$ s). Note that $\Delta t_{\text{FT}} \sim \Delta \Psi_{\text{FT}} \sigma / I_{\text{OH}}$ increases as I_p decreases due to the increase in the flux available for the flat-top phase, $\Delta \Psi_{\text{FT,max}} \sim C_1 - C_2 I_p$ (where $C_1, C_2 > 0$ [255]). The inductive current, $I_{\text{OH}} = I_p - I_{\text{NI}}$ also decreases with the reduction in I_p . The dependence of electrical conductivity, $\sigma(T)$, on I_p is more complicated and is influenced by the auxiliary heating and core transport [184, 256]. Hybrid scenario with $\Delta t_{\text{FT}} = 1000$ s was originally predicted for ITER under conservative conventional transport assumptions with the same energy confinement as the IBL scenario, $H_{98y2} = 1$ [257]. However, it achieves a higher beta, $\beta_N \leq 2.5$ without sawtooth [12, 35].

This conservative hybrid operation was predicted for ITER, with the plasma current reduced to ~ 13 MA and safety factor increased from $q_{95} = 3.0$ to 3.5 while maintaining plasma parameters sufficient to meet the technological goals [12].

Since 1999, various tokamak experiments have suggested scenarios as potential candidates for ITER hybrid operation [11]. These scenarios featured magnetic configurations with a large volume of low magnetic shear and $q(0) \sim 1$, leading to quasi-stationary discharges that performed better than the baseline scenario. They typically exhibited no or very mild sawteeth [258], helping to avoid NTMs triggered by sawtooth oscillations. Moreover, these scenarios could overcome the challenges of stringent control that is often required for scenarios with strong ITBs (see section 6).

As these candidate scenarios rapidly progressed, they demonstrated higher performance with $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$, surpassing the baseline scenario and exceeding the performance of the originally proposed ITER hybrid scenario ($H_{98y2} = 1$, $\beta_N \leq 2.5$). This led to the proposal of the so-called ‘advanced hybrid scenario.’ It is a variation of the hybrid scenario, aimed at achieving even higher fusion yields, allowing for operation with a high fusion gain ($Q \sim 10$) for long-pulse durations. This relies on improved performance through an optimized current profile at a lower current and a lower loop voltage [11].

The improved performance of the advanced hybrid scenario could serve as an alternative operation mode if the confinement quality in the baseline scenario falls short of expectations in ITER. Additionally, by operating at lower plasma current compared to the inductive IBL scenario, it could reduce the impact of disruptions. This advanced hybrid scenario is often simply referred to as the ‘hybrid scenario’ [11]. In this paper, the term ‘hybrid scenario’ refers to the advanced hybrid scenario.

Based on the advanced hybrid scenario, operating at 15 MA, as in the baseline scenario, was also considered a potential route to achieve a very high fusion gain of $Q > 20$, assuming that the confinement enhancement is sustained. To account for this broader possibility of the hybrid scenario in ITER, the term ‘advanced inductive’ scenario was introduced [217, 219]. Plasmas that meet the criteria of $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$ in stationary operation ($t_{\text{dur}} \geq 5\tau_E$) are classified as advanced inductive scenario in present-day experiments [219]. Notably, the criterion of $\beta_N = 2.4$ is sufficient to maintain 500 MW of fusion power at 75% of the baseline scenario’s

Table 5. Types of hybrid scenarios in ITER.

	I_p (MA)	H_{98y2}	Q
Hybrid scenario	< 15	1	> 5
Advanced hybrid scenario (often simply called hybrid scenario)	< 15	≥ 1	≤ 10
Advanced inductive scenario	~ 15	≥ 1	> 20

current or at $q_{95} = 4$. A summary of these types of hybrid scenarios for ITER is provided in table 5.

5.2. Experiments in support of ITER and beyond

Following the definition of the hybrid scenario in ITER provided in the previous subsection, we define a hybrid scenario for discharges that meet the following criteria, making them potential candidates for the advanced hybrid or advanced inductive scenarios in ITER;

- $q(0) \sim 1$ and low magnetic shear in the central region with $0.6 \leq I_i(3) \leq 1.2$
- $3.0 \leq q_{95} \leq 6.5$ where sawtooth appears inherently if operated in conventional H-mode
- without or with mild sawtooth activities
- $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$ above the IBL scenario
- stationarity with durations longer than $5\tau_E$.

It is noteworthy that the term ‘stationarity’ implies that the plasma profiles reach a state where they no longer change over time. A more relevant evaluation of stationarity would consider the global current relaxation time or the wall saturation time. However, since these parameters are generally unavailable in the database collected from various machines, $t_{\text{dur}} \geq 5\tau_E$ is used as a practical criterion for defining a stationary plasma, even though this is typically shorter than the current relaxation time and the wall saturation time [219].

A variety of experiments have been independently conducted across different devices in search of such hybrid scenario conditions. The most crucial condition to meet is the formation of a stationary q -profile with a large region of low magnetic shear in the core and $q_{\text{min}} \sim 1$, where improved performance compared to conventional H-mode discharges could be originated. The experimental approaches to tailor the q -profile can be categorized into several methods, as shown in figure 24;

- Early heating during the current ramp-up phase: Moderate heating power is applied during the current ramp-up phase to avoid the formation of a strong ITB and sometimes to prevent an L–H transition. Early heating delays current penetration into the core, enabling low magnetic shear in the plasma’s central region and avoiding the formation of a $q = 1$ surface, which postpones the onset of sawteeth until the main heating phase. Once the current flat-top is reached and before sawtooth instabilities occur, the main heating is applied. In some cases, feedback control of beta is used to prevent MHD activities, such as 2/1 NTM, caused by beta

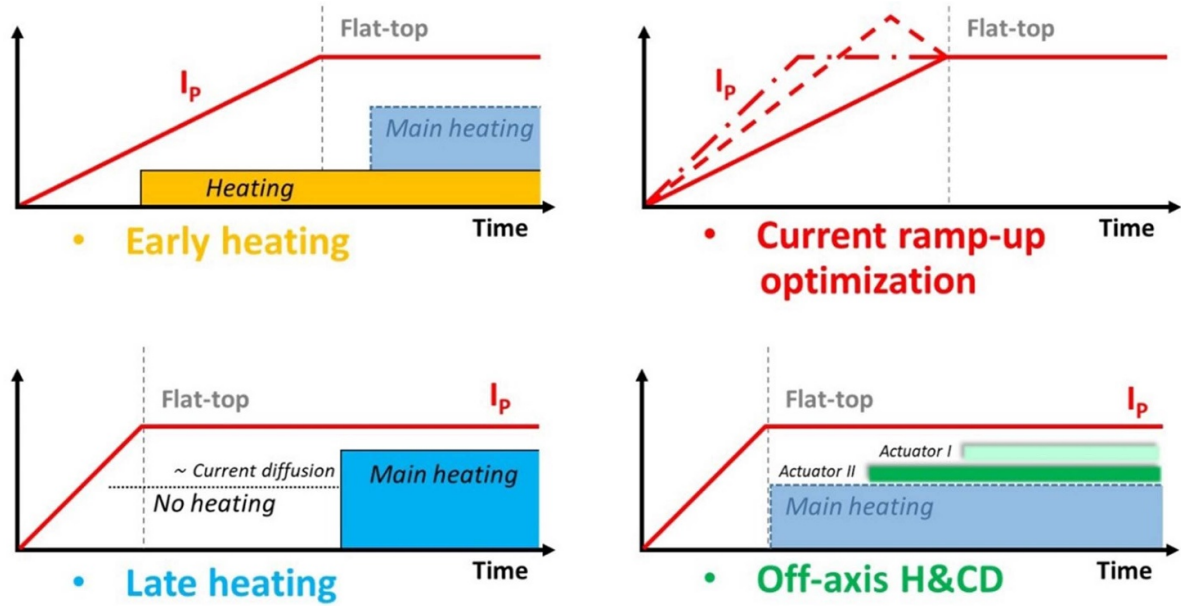


Figure 24. Experimental approaches to establish hybrid scenario.

excursions as the main heating is applied. Alternatively, the main heating may be delayed until after full current diffusion to establish a more stable q -profile and to avoid the beta excursion.

- Plasma current ramp-up optimization including current overshoot: A higher ramp-up rate increases the surface current, allowing low magnetic shear to form in the central region while delaying current diffusion. However, in superconducting tokamaks, the ramp-up rate may be limited. In such cases, the current overshoot technique can be applied; the plasma current is ramped up to a higher level than required and then ramped down after a short delay. This technique creates a flat shear region, with an extended radius compared to fast ramp-up alone. Furthermore, it produces high magnetic shear in the outer plasma region, which is used to optimize the s/q profile based on ITG theory [259], where s represents magnetic shear. Main heating is typically applied at the beginning of the current flat-top phase, right after the ramp-down of the plasma current.
- Late heating in the current flat-top phase: No heating is applied during the current ramp-up phase. Instead, full heating is introduced once the current is fully diffused, using a power level below the threshold for ITB formation. This approach could be unsuitable for low q_{95} operation; when q_{95} is low, a $q = 1$ surface may form before the main heating that could result in lower performance as applying the main heating in a sawtooth plasma.
- Off-axis H&CD in the flat-top phase: Off-axis H&CD sources can actively modify the q -profile for the hybrid scenario. An optimized actuator mix sustains the current profile while maintaining a flat q in the core region. This method can be applied to the current ramp-up phase as well.

It is noteworthy that different q - and magnetic shear profiles can be achieved in the stationary phase depending on the

approach used. Initial variations in the q - and magnetic shear profiles can give rise to different MHD instabilities, which influence the evolution of plasma parameters and impact transport. Consequently, the fully diffused q -profile may vary depending on the applied scheme.

5.2.1. Hybrid scenario development in various devices.

Here, we address the progress of hybrid scenario development in present devices. Main features of hybrid scenario experiments in various devices are summarized in table 6 updated from [219]. Here, proxies are used for $\rho^* = (W_{th}/nV)^{1/2}/Ba$, $\nu^\dagger = (n^3 V^2 / W_{th}^2)(R^5/a^3)^{1/2}$, where W_{th} in MJ and n in 10^{19} m^{-3} . The details of development of hybrid scenario in each device are described in the following.

5.2.1.1. AUG. AUG has developed ‘improved H-mode scenarios’ as a candidate of ITER hybrid scenario [260–262]. They usually do not exhibit ITBs, except for some cases with transient ITBs in the very early phase [263]. High density scenario of $n_e/n_{GW} > 0.8$ has been established at high triangularity ($\delta = 0.43$) with $\beta_N = 3.5$ in stationary conditions with type-II ELMs [260, 261, 264, 265].

To optimize the improved H-mode scenarios, the impact of the target q -profile shape was investigated in relation to MHD activities in both the early heating scheme and the late heating scheme [266]. It has been speculated that three mechanisms play a role in linking the observed changes in MHD behavior and current profile to changes in the kinetic profiles; increased transport due to the MHD modes themselves, variation in the ratio of magnetic shear (s) to safety factor (q), which modifies the critical temperature gradient length for the onset of ITG, and effects on the H-mode pedestal pressure.

After full coverage of the wall with W, AUG evaluated the compatibility of these high performance scenarios with an

Table 6. Main features of hybrid scenario experiments in various devices.

Device	Experimental approach	Operation window	Wall material	Integrated scenario
AUG	Early heating	$I_p = 0.6\text{--}1.21$ MA $B_t = 1.39\text{--}2.79$ T	• C	Radiative divertor
	Current overshoot	$q_{95} = 2.8\text{--}5.0$ $\rho^* \sim 0.06\text{--}0.11$	• W	
	Late heating	$\nu^\dagger = 3 \times 10^5\text{--}6 \times 10^6$ $f_{\text{GW}} \sim 0.35\text{--}0.95$		
DIII-D	Early heating	$I_p = 0.68\text{--}1.5$ MA $B_t = 1.01\text{--}2.15$ T	• C	Low torque
		$q_{95} = 3.0\text{--}5.2$ $\rho^* \sim 0.07\text{--}0.14$		Steady-state
		$\nu^\dagger = 1 \times 10^5\text{--}2 \times 10^6$ $f_{\text{GW}} \sim 0.30\text{--}0.75$		ELM suppression
				Radiative divertor
EAST	Early heating	$I_p = 0.4\text{--}0.5$ MA $B_t = 1.5\text{--}1.6$ T $q_{95} = 3.4\text{--}4.4$ $\rho^* \sim 0.08\text{--}0.11$ $\nu^\dagger = 2 \times 10^6\text{--}9 \times 10^6$ $f_{\text{GW}} \sim 0.40\text{--}0.65$	• C with W divertor	
JET	Current overshoot	$I_p = 0.9\text{--}2.0$ MA $B_t = 1.12\text{--}2.27$ T $q_{95} = 2.8\text{--}5.3$ $\rho^* \sim 0.04\text{--}0.09$ $\nu^\dagger = 1 \times 10^5\text{--}3 \times 10^6$ $f_{\text{GW}} \sim 0.40\text{--}0.85$	• C	Radiative divertor
			• Be with W divertor	20 s long-pulse DT
JT-60U	Early heating	$I_p = 0.85\text{--}1.80$ MA $B_t = 1.54\text{--}4.06$ T $q_{95} = 3.1\text{--}5.6$ $\rho^* \sim 0.04\text{--}0.07$ $\nu^\dagger = 1 \times 10^5\text{--}2 \times 10^6$ $f_{\text{GW}} \sim 0.35\text{--}0.60$	• C	28 s long-pulse
KSTAR	Early heating	$I_p = 0.6\text{--}0.8$ MA $B_t = 1.8\text{--}2.0$ T	• C	~ 20 s long-pulse
	Current overshoot	$q_{95} = 3.5\text{--}6.5$ $\rho^* \sim 0.07\text{--}0.10$	• C with W divertor	ELM suppression
	Late heating	$\nu^\dagger = 8 \times 10^5\text{--}7 \times 10^6$ $f_{\text{GW}} \sim 0.30\text{--}0.60$		

all-W FW and divertor. Stable operation with a tolerable core W concentration (below 5×10^{-5}) requires a careful combination of gas puffing and localized central heating. In the boronized AUG, discharges with heating power above 10 MW were safely managed with active cooling of the divertor plasma, achieved by enhancing radiation through nitrogen seeding [267]. The plasma current overshoot method was also applied for high-beta improved H-mode scenarios, achieving transient high performance with $H_{98y2} > 2$ and $\beta_N > 4$ [268, 269]. Although high beta and confinement were achieved, the non-inductive CD fraction remained relatively low. With the availability of more ECRH power [270], improved H-modes featuring a weak 3/2 mode, similar to those observed in [266], were re-established [271].

Inter-discharge optimization of the H&CD actuators of an improved confinement scenario with $q_{min} > 1$ has been

attempted with the RAPTOR fast transport code [272]. The model predictions were successfully used to make the ramp-up with early heating more robust to the onset of 3/2 modes, while reaching a stationary state in the early flat-top phase. This model-based scenario development approach reduces the number of shots required for developing the scenario and offers a path to real-time model-based control.

5.2.1.2. DIII-D. DIII-D has pursued high performance long-pulse discharges near the no-wall β limit. One such approach was the hybrid scenario with $q_{min} \sim 1$, which demonstrated a long-pulse of 6.3 s with $\beta_N H_{89} \sim 7$ [278]. Feedback control of the energy content and particle inventory enabled reproducible and stationary operation. The particle balance was maintained through a combination of gas fueling and active

pumping [279]. DIII-D has also worked to establish integrated hybrid scenarios under ITER-relevant experimental conditions, including low external torque, ELM mitigation and suppression, and core-edge integration with a radiating divertor. Efforts to extend these scenarios to steady-state conditions have been supported by the application of ECCD. These results of integrated scenario development are discussed later in this section.

5.2.1.3. JET. JET began experiments with the carbon wall by conducting identity experiments with AUG, aiming to match some dimensionless parameters and plasma shape as closely as possible [280, 281]. These experiments have been extended to lower ρ^* regime. The plasma current overshoot technique was applied to produce a target q -profile characterized by high magnetic shear in the outer half of the plasma and low central shear in the inner half, with $q(0) \sim 1$, which is advantageous according to the ITG theory [282], as mentioned previously. This approach achieved high performance, with $H_{98y2} = 1.3$ – 1.4 and $\beta_N \sim 3$, at both high and low triangularity [275]. However, it is noteworthy that the target q -profile could not be properly sustained using the current overshoot technique. After one current diffusion time, the low magnetic shear region in the plasma core was gradually lost, leading to the onset of sawteeth and NTMs. Nevertheless, as shown in figure 25, it was possible to maintain high confinement in this scenario once the $n = 3$ NTM activity disappeared, even in the later stages of the discharge. This indicates that, in addition to the q -profile shape, other mechanisms, likely involving both the core and pedestal regions [283], played a role in maintaining good confinement.

The hybrid scenario was further investigated in JET-ILW with the goal of developing a path to high fusion power for the second DT campaign at JET (DTE2). Development of the hybrid scenario at $I_p = 2.5$ MA, $B_t = 2.9$ T, $q_{95} \sim 4$, was conducted with input power of 23–27 MW, using both NBI and ICRH. Good confinement, $H_{98y2} = 1.1$, was achieved at $\beta_N = 2.3$ with low gas injection. However, very low gas injection resulted in severe high-Z central impurity accumulation, as the ELM frequency became too low [284]. Further development showed that increasing the input power to 32 MW at $I_p = 2.2$ MA, $B_t = 2.8$ T improves the confinement to $H_{98y2} = 1.2$ with $\beta_N = 2.9$. Notably, increasing the ICRH power at low H minority concentration helped reduce impurity accumulation, though it did not fully prevent it [285, 286].

The final D campaign before DTE2 helped identify the physics mechanisms needed to extend the duration of the hybrid scenario while preserving high confinement [136, 287]. This was achieved by using an optimized entry into H-mode with very low neutral gas injection during NBI power addition. In this phase, an ELM-free plasma was obtained with high T_i at the pedestal, which developed strong resilience to impurity accumulation due to the outward pinch effect of neoclassical transport [288]. Shortly after reaching maximum NBI power, the neutral gas injection rate was increased, and the plasma began to develop type-I ELMs. Low impurity accumulation

was maintained throughout the discharge. Optimum conditions were found at $I_p = 2.3$ MA, $B_t = 3.4$ T, $q_{95} \sim 4.9$, at a slightly lower $\beta_N = 2.0$ than in previous developments at JET. High B_t was useful to minimize harmful MHD effects, though it had little impact on confinement, which mostly depends on reaching $\beta_p \geq 1$.

5.2.1.4. JT-60U. JT-60U has established scenarios for advanced tokamak research based on high β_p ELMy H-mode [289]. These scenarios are characterized by improved core confinement, featuring ITBs in the particle, ion energy, and often the electron energy channels, along with an H-mode pedestal. The aim was to achieve steady-state operation with a high fraction of bootstrap current exceeding 50%, enabled by both ITB and ETBs, which facilitated long sustainment times due to modifications made to the W-shaped divertor. In these discharges, the ITBs were sustained with a stationary current density profile, contributing to high performance with a wide low shear volume, and $q(0) \sim 1$, $H_{98y2} \gtrsim 1.0$, $\beta_N > 2.4$ [276, 290].

5.2.1.5. EAST. One significant advancement in the development of hybrid scenarios is that these scenarios have been achieved under stationary conditions in superconducting tokamaks such as EAST [274] and KSTAR [277]. EAST has developed this scenario with NBI and LH, where ITBs were observed in all transport channels with a centrally flat q -profile while the 1/1 fishbone instabilities were present. $\beta_N \sim 2.1$ and $H_{98y2} \sim 1.1$ were achieved with the ITER-like tungsten divertor. It was found that the fishbone mode can be beneficial to sustain $q(0) \sim 1$, thereby facilitating stable ITBs and higher plasma performance. Additionally, the formation of the ITB of the electron density was linked to the fishbone activities [238, 274].

5.2.1.6. KSTAR. In KSTAR, hybrid scenario has been developed in the operation window of $3.5 < q_{95} < 6.5$, where the sawtooth inherently appears, with various experimental approaches; early heating, current overshoot, and late heating. For the early heating approach, a moderate NB heating power of ~ 2 MW is applied during the ramp-up phase, followed by an additional NB power of around 2 MW or slightly higher when the current flattop begins. However, this can lead to MHD instabilities due to the transient runaway of beta when the main heating is applied. To avoid this, full heating is rather delayed until after the current diffusion while keeping the moderate heating during the ramp-up. ECH is employed to reduce T_i/T_e and the toroidal rotation to establish ITER-relevant conditions. Achievements included $\beta_N \lesssim 3.0$ and $H_{89} \lesssim 2.4$ at $f_{GW} \sim 0.7$ with low T_i/T_e and low toroidal rotation through ECH [277].

A transition from the single null to the double null configuration and triggering CEM are also explored to access the hybrid regime, where the pedestal becomes broader and higher, and the fast ion content increases due to a decrease in plasma density, thereby improving the overall plasma performance [291–293].

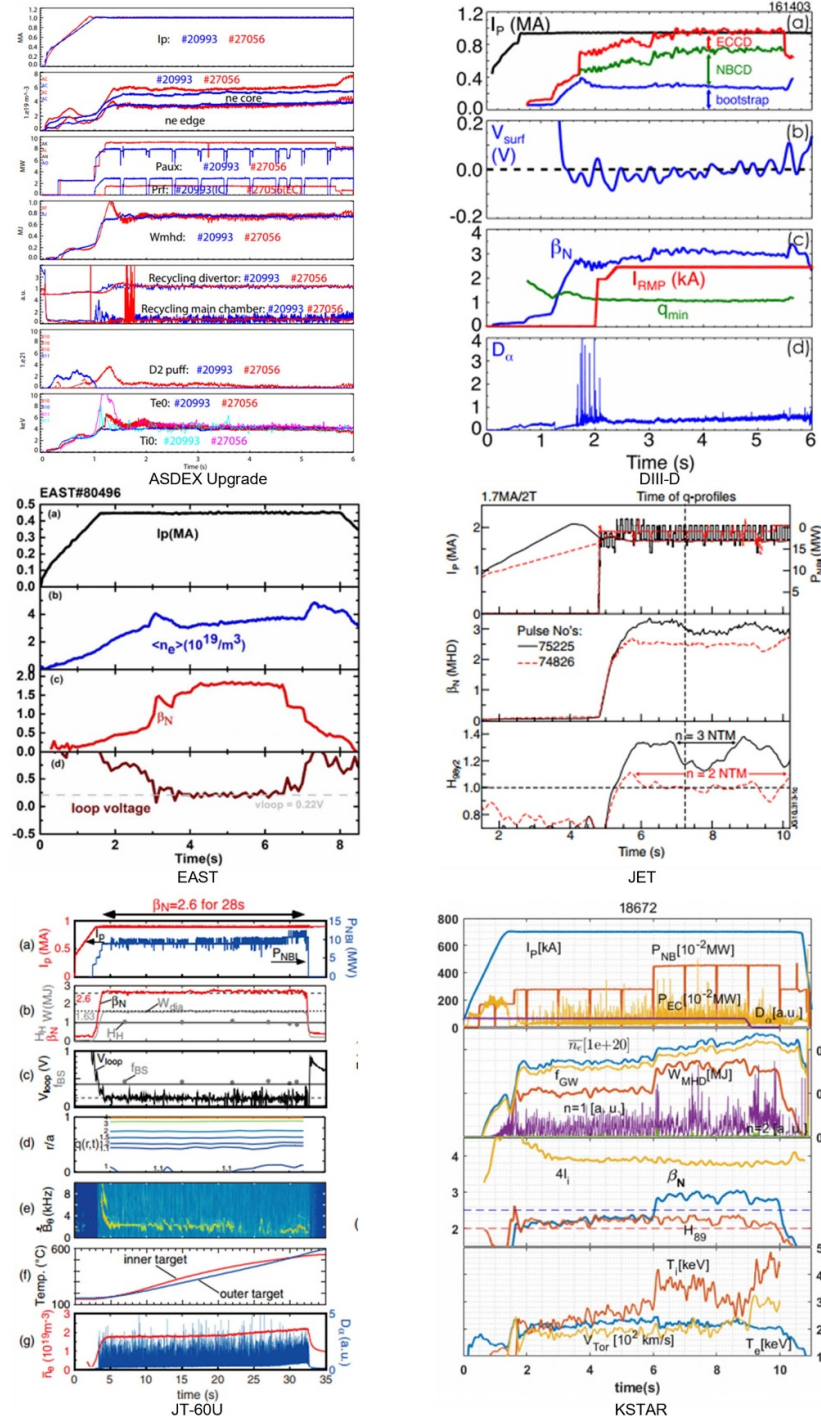


Figure 25. Examples of hybrid scenario developed in AUG (shot 20993 and 27056) (reproduced with permission from [271]), DIII-D (shot 161403) (reproduced from [273]), © 2017 IAEA, Vienna. All rights reserved), EAST (shot 80496) (reproduced from [274]), © 2020 IAEA, Vienna. All rights reserved), JET (shot 74826 and 75225) (reproduced from [275]), © IOP Publishing Ltd All rights reserved), JT-60U (shot 48158) (reproduced from [276]), © 2009 IAEA, Vienna. All rights reserved), and KSTAR (shot 18672) (reproduced from [277]), © 2020 IAEA, Vienna. All rights reserved).

Additionally, ML techniques have been applied to achieve high performance in hybrid scenarios in KSTAR [294, 295]. A data-driven simulator was developed for KSTAR operations using LSTM-based NNs, then deep RL was utilized to empirically learn optimal decision-making at given states,

rewarding good actions to achieve target β_p , q_{95} , and I_i with the action variables of I_p , κ , δ_i , δ_u , Δ_{in} , and R_p [294, 295]. It demonstrated that the target plasma states could be successfully achieved with the AI-designed operation trajectory in a real KSTAR experiment.

5.2.2. Integrated hybrid scenario development in ITER-relevant conditions. Once robust hybrid scenarios were developed, efforts focused on establishing integrated scenarios under ITER-relevant conditions by (i) reducing external torque, (ii) extending to steady-state operation, (iii) applying ELM mitigation or suppression, (iv) utilizing radiative divertors, and (v) enabling long-pulse operation. Notably, JET has attempted to develop hybrid scenarios in both T and DT experiments.

5.2.2.1. Reducing external torque. Low external torque experiments have been primarily conducted on DIII-D and JT-60U. DIII-D has sustained hybrid scenario plasmas with a high performance with very low external torque of ~ 1 Nm, a level expected to generate a similar amount of rotation as NBI in ITER [296]. While a high $\beta_N \sim 3.1$ was achieved, the confinement was reduced to $H_{98y2} \sim 1$, associated with the lower rotation and torque. Discharges with zero net torque were created, but these were found to be increasingly susceptible to 2/1 NTMs. Additionally, in higher-power hybrid plasmas (~ 19 MW) with counter- I_p NBI sources, the torque was reduced to ~ 5.6 Nm. This led to a 20% reduction in while maintaining the same β_N as discharges with ~ 15 MW of co- I_p NBI power [297]. JT-60U has also explored low toroidal rotation conditions, as expected for ITER, by using balanced injection of tangential NBIs along with some additional perpendicular NBIs. The toroidal rotation speed of 23.1 km s^{-1} , or a toroidal rotation frequency of 1.08 kHz , was obtained at the plasma center, which is similar to the value expected for ITER ($\sim 1.3 \text{ kHz}$ at the plasma center [16]). High β_N discharges with high H_{98y2} have been established in JT-60U, where no confinement degradation was reported, mainly due to the benefits of ferritic steel tile installation [276]. After the installation of ferritic steel tiles, fast ion losses due to toroidal field ripple were reduced by a factor of 1/2 to 1/3. The increase in absorbed power at the same injection power reduces the required number of NBI units to sustain a given β_N , improving the flexibility of torque input by increasing the available combination of tangential NBI units [290]. KSTAR has attempted to conduct experiments with reduced torque by adding ECH on top of NBI or by applying RMP. As a result, the toroidal rotation has been reduced; however, a high level of rotation still remains [277].

5.2.2.2. Extending to steady-state operation. Steady-state hybrid scenarios with high performance has been developed in DIII-D by applying strong central CD with 3.0 MW of ECCD. Fully non-inductive CD fraction with the contribution of $\sim 50\%$ from bootstrap current was achieved with $H_{98y2} \sim 1.4$ and $\beta_N \sim 3.4$ with the ITER similar shape. Despite the additional on-axis ECCD, these hybrid plasmas do not show significant sawtooth activity due to the anomalously broad current density profile from poloidal flux pumping [298]. Additional experiments demonstrated stationary, fully non-inductive operation at $\beta_N \sim 3.7$ for ~ 3 s, which is longer than the current diffusion time on DIII-D [297]. This will be further discussed in section 6.

5.2.2.3. Applying ELM mitigation or suppression. ELM suppression with RMPs has been attempted in DIII-D and KSTAR hybrid scenarios. Type-I ELM suppression on DIII-D hybrid plasmas was first achieved with $n = 3$ RMPs in plasmas up to $\beta_N \sim 2.5$ [299]. Later experiments achieved a steady-state regime with full ELM suppression by $n = 3$ RMP [273]. The impact of $n = 3$ RMP to the confinement was surprisingly little during the ELM suppression. Although higher ECCD power reduced the confinement, the AE activity was replaced by a benign fishbone-like mode during EC which results in reducing anomalous fast ion transport so that the improved confinement could be maintained, $H_{98y2} \lesssim 1.4$. Subsequent experiments found suppression of type-I ELMs led to a ‘grassy’-ELM regime for $q_{95} \sim 5.3\text{--}7.1$ in hybrid plasmas on DIII-D [300]. KSTAR conducted experiments to stabilize ELMs with $n = 1, 2$ RMP and the ELM suppression regime was sustained for 34 s [301] and extended up to 45 s [302]. However, RMP reduced the confinement about 10%–30%, $H_{98y2} < 1.0$, so that high confinement could not be sustained, [212] and the hybrid regime is thought to be lost.

5.2.2.4. Utilizing radiative divertors. Radiative divertor operation has been attempted for heat flux reduction in AUG [267] and JET [303, 304] using nitrogen and in DIII-D [305] using neon or argon [306]. It is noteworthy that all W wall and ILW have been explored in AUG and JET, respectively which can make a significant difference because the carbon replacement by the seeding impurity can no longer take place. The heat flux to the outer divertor strike point was found to be reduced by a factor of 4 in JET and AUG and 2.5 in DIII-D. Although JET reached $f_{GW} = 0.95$, the type-III ELMs were transitioned with reduction of confinement to $H_{98y2} = 0.83$. On the other hand, AUG reached $f_{GW} = 0.8$ only, but the confinement was improved, H_{98y2} up to 1.3. This improvement mainly results from the increase in pedestal temperature without any deleterious effects of impurity accumulation or radiation from the core. DIII-D kept f_{GW} around 0.6 by the ‘puff and pump’ scheme and maintained the type-I ELMs with no reduction of confinement while the argon input was increased.

5.2.2.5. Enabling long-pulse operation. Long-pulse operation has been demonstrated to evaluate the suitability for high-fluence application. JT-60U modified power supply systems for three perpendicular NBIs to keep peaked pressure profile for a long period of central NB heating up to 30 s. The installation of ferritic steel tiles reduced fast ion losses so to achieve good thermal plasma confinement $H_{98y2} \sim 1$ [290] as mentioned in section 5.2.2.1. With these NBI systems and ferritic steel tiles, β_N of 2.6 and $H_{98y2} > 1$ were sustained for 28 s corresponding to $\sim 14\tau_R$, where τ_R is the current diffusion time, under the ITER-relevant small toroidal rotation condition after the adjustment of the toroidal magnetic field so as to reduce the $n = 1$ mode amplitude [276]. In the latter phase of the discharge, the divertor recycling was increased related to the reduction of the net wall pumping speed due to the increase in the wall temperature which resulted in the increase of the pedestal density and decrease of the pedestal

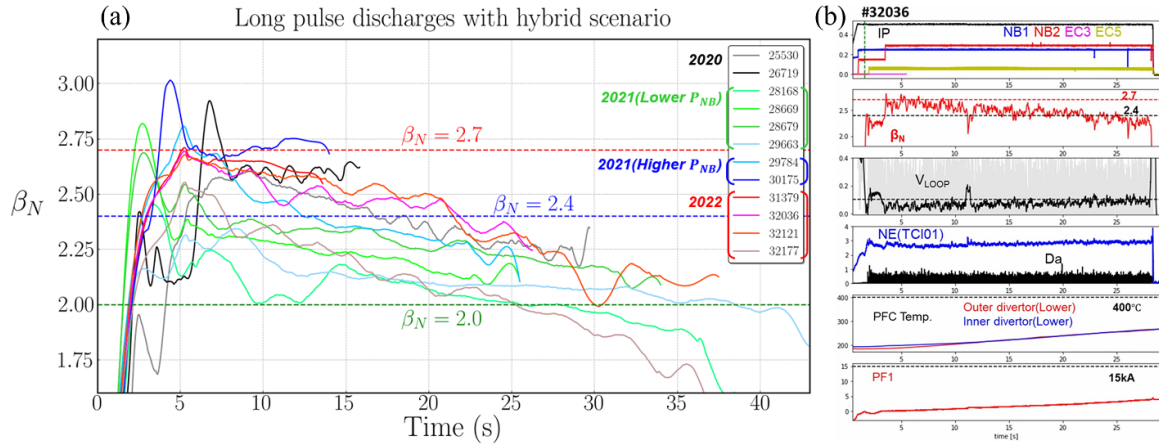


Figure 26. (a) Long-pulse hybrid scenario development in KSTAR, (b) a hybrid scenario with $\beta_N \geq 2.4$ for > 20 s targeting quasi-steady-state long-pulse operation. Reproduced from [309]. © 2024 The Author(s). Published by IOP Publishing Ltd on behalf of the IAEA. CC BY 4.0.

and the core temperature as shown in figure 25. JET developed 20 s hybrid scenarios with $\beta_N \sim 2.5$ and $H_{89} \sim 2.0$ by implementing a number of control features such as boundary flux control, strike points sweeping, and total plasma current control through external non-inductive CD systems. The stationary hybrid scenario was achieved with the high-triangularity configuration with the plasma current of 1.3 MA, toroidal field 1.5 T, and $q_{95} = 3.5$ where up to 186 MJ have been injected in the plasma via NB with up to 50% of current density driven non-inductively [307]. KSTAR is developing long-pulse hybrid scenarios since 2020 and $\beta_N > 2.4$ was maintained for 20 s. Additionally, an even higher $\beta_N \sim 2.7$ was sustained for over 10 s, but a degradation of β_N performance was observed over time, despite the discharge being able to be sustained for more than 40 s as shown in figure 26. This degradation in time has been observed in other long-pulse discharges in KSTAR [308]. Although, various reasons have been raised such as drift signals in the magnetic measurements, recycling and impurity influx with the wall temperature rise, and presence of AEs, the issue has not been fully resolved yet [309] and it will be further addressed with the newly installed W divertor.

5.2.2.6. DT operation. Lastly, hybrid scenario has been established in DT experiments. JET developed hybrid scenario in the DTE2 campaign following the recipes found in the previous D campaigns. Impurity accumulation was found to be more severe in the presence of T and it was necessary to re-optimize the gas wave forms in order to compensate for higher densities and lower ELM frequencies in DT. Nevertheless, favorable confinement properties, $H_{98y2} = 1.2$, at high β_p above 1.0 and $\beta_N = 2.5$ were also found in DT with $I_p = 2.3$ MA and $B_t = 3.4$ T, $q_{95} \sim 4.9$. In agreement with theoretical studies performed previously at JET [177], deviations from gyroBohm mass scaling for heat and particle transport and confinement were found in DT compared to D as the thermal energy content was higher in DT. Also, the q -profile was tailored by considering the isotope effect observed in the mixed HD experiments where a systematic increase of $q(0)$

was shown at the start of heating due to core impurity radiation [310]. It ended up with new records of the fusion power above 12 MW for 3 s and 10 MW for 5 s satisfying the DTE2 target [311]. Fusion energy of 59 MJ was also achieved in the T-rich beam-target dominated plasmas [312].

5.3. Main issues of hybrid scenario

It is essential to develop scenarios and conduct physics research on individual devices while simultaneously pursuing a comprehensive study that integrates findings across multiple devices. Therefore, the IOS TG attempted to define the operational domain of these hybrid scenarios and identify commonalities in performance and phenomenology, focusing primarily on four tokamaks: AUG, DIII-D, JET, and JT-60U. This effort seeks to establish a physics basis for applying the results from current devices to ITER. In the process of identifying commonalities, several critical issues related to hybrid scenario operation and energy confinement have emerged. The three main issues are:

(i) Can the enhancement of energy confinement be reproduced in ITER? (ii) Will moderate MHD stability be achieved under ITER conditions? (iii) Can the unique shape of the q -profile be maintained in ITER?

IOS TG has begun addressing these issues through three JEXs outlined below.

- Access conditions for hybrid with ITER-relevant restrictions (IOS-4.1)
This is to document plasma conditions to obtain a selected number of q -profiles for the hybrid scenario at β_N close to 3 and to document experimentally in each device the effect of q -profile on confinement and stability before current profile relaxes.
- ρ^* dependence on transport and stability in hybrid scenarios (IOS-4.2 (SSO-2.3))

Hybrid discharges in different experiments allow coordinated studies on to the effect of ρ^* on confinement and stability of this regime. The aim of these experiments

is to make identity experiments at the same ρ^* for detailed transport studies (with common and improved diagnostics) for the hybrid scenario at $q_{95} = 4$ and to get the largest possible variation in ρ^* for confinement scaling and stability boundary scaling studies. Data from these experiments will be included in the global confinement database, to take advantage of the operational capabilities of this high beta regime.

- Collisionality scaling of confinement in hybrid scenarios (IOS-4.3)

Analysis of the ITPA hybrid scenario performance database indicated that there is a substantial increase in H_{98y2} with decreasing collisionality. Dedicated experiments are required to assess the impact on projection to ITER, since even a weak variation of confinement with collisionality can have a substantial influence, due to ITER lying at much lower collisionality than present-day experiments.

It is noteworthy that, in parallel to addressing the issues mentioned above, each machine has conducted experiments and analyses to better understand the physics mechanisms behind confinement enhancement. Key factors being investigated include the $E \times B$ shearing rate, pedestal improvement, magnetic shear effects, fast ion dynamics, alpha stabilization, and the influence of fishbone instabilities. These details are primarily discussed in chapters 2 and 3. Some are highlighted in [238, 274, 277, 283, 313–318].

Based on the JEXs of IOS-4.1 to 4.3, an experimental database of more than 500 plasmas was collected from AUG, DIII-D, JET, and JT-60U where the parameter range is broad in q_{95} and density normalized to the empirical density limit [219].

This database allowed to address the first main issue of hybrid scenarios, ‘Can the enhancement of the energy confinement be reproduced?’ with the energy confinement scaling study. It was found that H_{98y2} increases with ρ^* , decreases with ν^* , and is roughly independent of β_{th} . It is noteworthy that the database from which the IPB98(y,2) scaling was derived using the database containing very little data with $\beta_N > 2.4$ corresponding to the operation window of hybrid scenarios. Other physics parameters such as T_i/T_e , rotation were considered which are not included in the standard engineering parameter scaling relation. It was found that T_i/T_e and the rotation cannot explain the variation in normalized confinement observed [219].

To systematically reveal the dependency of energy confinement on dimensionless parameters, dedicated experiments were conducted across devices under IOS-4.2 and 4.3. These experiments show a weaker scaling with normalized gyroradius than the standard H-mode scaling. In contrast, the collision frequency exhibits a favorable scaling that is stronger than the standard H-mode scaling. However, the database was not sufficiently well-conditioned to yield a new scaling [219]. So further studies are needed with more dedicated JEXs and theory/simulation works in collaboration with T&C TG.

To evaluate the role of pedestal to the energy confinement, dedicated experiments have been performed focusing on the pedestal behavior between AUG and DIII-D. The results show

that the pedestal continues to rise as more power flows out through the edge and that shape changes can have a significant impact on the pedestal parameters. Aside from the stable access to higher β_N , no obvious distinction was found between the pedestal behavior in hybrid and standard H-mode plasmas [315]. A reduction in confinement, similar to that observed in the larger conventional H-mode dataset, is seen in hybrid scenarios with ELM mitigation. This was also noted in KSTAR [212] but was not as pronounced in DIII-D steady-state hybrid scenarios [273].

To address the second issue of ‘Will the moderate MHD stability appear?’, the stability boundary or mapping of MHD modes in operation spaces need to be identified. The upper limit in β_N is set by the onset of $n = 1$ tearing modes in many tokamaks [218, 277, 289, 316, 319, 320]. Due to their nature of resistive mode, the impact is often a dramatic loss of energy confinement rather than an immediate disruption. They can slow the plasma rotation and continue to grow as a non-rotating mode, potentially leading to disruption. However, their relatively slow growth allows for detection and the opportunity for active means to recover the plasma performance or avoid a disruption. Below the maximum achieved β_N , there is consistency in the type of MHD observed [276, 279, 319, 321]. Small sawteeth are generally observed for $q_{95} < 4$, but are often absent at values of q_{95} above this. Fishbones are frequently observed with the minimum q -value close to unity. In AUG and DIII-D, the fishbones tend to appear at higher density, while at lower density, $n > 1$ tearing modes prevail. KSTAR showed that the sawtooth was gradually replaced by fishbones as the hybrid regime was accessed from the standard H-mode regime [277].

The third issue of ‘Will the unique shape of the q -profile be maintained in ITER?’ is strongly linked to the issue of clamping the q -profile against the neoclassical theory and the long-pulse operation. This issue is still being addressed in the IOS TG and the physics mechanisms of clamping q -profile will be discussed in section 5.5.

5.4. Prediction of ITER

Significant progress has been made in the area of integrated modeling since the publication of the IPB in 2007. Advanced anomalous transport models, such as the TGLF model [243, 322] and QuaLiKiz [244] have been developed, along with improvements in particle [23] and impurity transport modeling. In parallel, pedestal models, such as EPED1 [183], have been developed and implemented for integrated modeling of ITER scenarios. Modeling efforts for hybrid scenarios have increasingly focused on using these transport and pedestal models in conjunction with the H&CD systems designed for ITER, with the aim of assessing hybrid scenario performance in ITER through integrated transport modeling.

Projections to ITER using integrated modeling have evolved step by step in last decades. Initially, simple extrapolations from experimental profiles in existing machines were attempted. Then, basic 0D assumptions for the confinement time was adopted, along with density and pedestal prescriptions. As a next step, energy transport was solved

using transport models, while the density profile was prescribed to evaluate its impact on the q -profile and Q . Later, particle transport was also solved. The projection effort was then extended from simulations of stationary conditions to full-discharge simulations, including the current ramp-up and ramp-down phases. Finally, discharge simulations incorporating the evolution of the PF and CS coils were conducted with free-boundary codes. In parallel, the anomalous transport models and pedestal models have been validated against experiments which has been mainly conducted by TC (see chapter 2 in this special issue) and PEDELM TG (see chapter 3 in this special issue), respectively and have been applied to predictive simulations for ITER mainly by IOS TG.

For this projection study, two subjects were explored as JAs using various integrated modeling codes. The first focused on benchmarking the integrated transport codes available worldwide. This work laid the foundation for future development of integrated modeling. Given that one of the main goals of the hybrid scenario is long-pulse operation, the second JA was specifically launched to explore this aspect as a potential alternative scenario for hybrid mode.

- Hybrid benchmark (IOS-JA3)

The goals are to establish a physics basis, in conjunction with present tokamak experiments, for these operating modes in ITER. Since it is not possible to reproduce all the physics parameters of ITER plasmas simultaneously in present experiments, simulations are used to project to the ITER regime using theoretically based physics models, that are being tested against present tokamak experiments. These efforts include energy, particle, current, and momentum transport, ideal MHD and H&CD in time-dependent discharge simulations. In this activity, only the energy and current transport are examined, and in some cases momentum transport, and the H&CD sources are also examined.

- Optimization of operational space for long-pulse scenarios (IOS-JA9)

The operational space of long-pulse H-mode plasmas has been characterized by 1.5D transport modeling to assess the viability of achieving ITER's long-pulse scenario goal of 1000 s of burn time with $Q \geq 5$. A range of plasma currents and plasma densities is explored and the burn length on the basis of the calculated flux consumption and the capabilities of the ITER PF system.

5.4.1. Benchmark study. The benchmarking was performed for the ITER hybrid scenario condition to establish a common basis for comparison; $I_p = 12$ MA, $B_t = 5.3$ T, and the reference plasma boundary shape [15] after a preliminary study [323]. Both fixed and free-boundary 1.5D transport codes were involved in this activity including CRONOS [324, 325], ONETWO [326, 327], TSC/TRANSP [328–331], TOPICS [332–334], and ASTRA [335, 336].

Firstly, the energy and current transport were examined without restricting the physics models used in each code. The GLF23 model [204, 205] was employed for the anomalous transport in all the codes except TOPICS where the CDBM

model was used. Two types of scenarios were explored; one with NBI and ICRH, and the other with NBI, ICRH, and ECCD. Here, the negative ion NBI (1 MeV, 33 MW, steered to full off-axis), ICRF heating (20 MW, 53 MHz, 2nd T harmonic), and ECH&CD (170 GHz, 20 MW, midplane launch steered for off-axis CD) were used. The simulations projected to fusion gains of 5.6–8.3, β_N values of 2.1–2.6, and fusion powers ranging from 350 to 500 MW. Agreement about 10% variation was achieved between codes in global quantities such as α particle heating power, stored energy, and internal inductance but at the more detailed level of temperature profiles and power density of the heating showed poorer agreement. The differences were mainly originated from the H&CD source models, specification of input parameters of the GLF23 model, and treatment of the equilibrium. Some arose from treatment of physics models in each code such as fusion power, bootstrap current, radiation, and so on. The central q -value, $q(0)$ was below unity range of 0.43–0.69 which could not be elevated much in spite of applying ECCD.

Secondly, after fine tuning of models among transport codes, the other benchmark study has been conducted using only GLF23 model to evaluate the electron and ion temperatures with the prescribed density profile for NB, EC, and IC H&CD sources. Here, the current diffusion was also calculated self-consistently with the NBI modules and prescribed sources from EC and IC. As shown in figure 27(a), much better agreement was achieved compared with the previous benchmark [15] due to identification and reconciliation of differences in the implementation of GLF23 and in the NBI H&CD modules in the various codes [219].

Lastly, another benchmark test was performed to evaluate the components of the current density for fixed temperature and density profiles. Figures 27(b) and (c) shows the total and bootstrap current densities and the q -profile for this test. The main source of the remaining difference has been identified as the way of describing the magnetic equilibrium in each code.

These significant amount of benchmarks for transport solvers with different models and H&CD showed that $Q > 5$ seems to be possible, though credible pedestal predictions are necessary. Here, 5 keV was assumed for the pedestal temperature. Simulations show β_N in the range 2.1–2.6, while present experiments indicate that values closer to 3.0 can be achieved, just below the $n = 1$ external kink no wall β limit [337, 338]. This needs to be addressed based on further understanding of the origin of confinement enhancement in hybrid scenarios. The simulated q -profile exhibited monotonic shapes with $q(0)$ below unity that is far from the hybrid q -profile. This requires whole discharge modeling including the current ramp-up phase and more attempt with various H&CD mixes as well as understanding of the unknown mechanism which frozen the q -profile.

5.4.2. Simple extrapolation from experiments. For projection to ITER, a simple exercise has been initiated based on experimental results. Extrapolation was pursued with the AUG hybrid experiment [314, 339]. The electron density and the ion temperature profiles were assumed to be the same as those of a

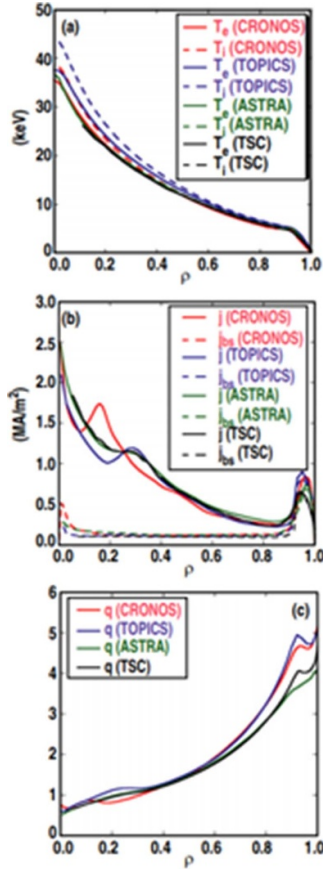


Figure 27. Profiles of predicted electron and ion temperature (a), current density (b) and q (c) for the modeling benchmark tests in ITER hybrid scenario. Reproduced from [219]. © 2014 IAEA, Vienna. All rights reserved.

AUG improved H-mode discharge but multiplied by factors to satisfy 80% of the Greenwald density fraction, $H_{98y2} = 1.5$ and $\beta_N = 2.9$ as in the experiment, while assuming $T_e = T_i$. The plasma current was assumed to be 11.3 MA to match $q_{95} = 3.8$ in the experiment. ASTRA was used to simulate this projection with an NBI power of 30 MW and an ICRH power of 23 MW by solving the current diffusion equation, which ended up with $Q = 4.93$ and $q(0)$ below unity [314]. Using the same thermal beta ($\beta_{N,th}$) and plasma profile shapes as obtained in AUG and setting 85% of the Greenwald density fraction predicted high fusion power of 1070 MW and $Q = \infty$ at $q_{95} \sim 3$ with $I_p = 14$ –15 MA. At lower I_p of 9.5–13 MA, fusion power above 400 MW could be achieved with $Q = 6$ –15. However, using the IPB98(y,2) scaling expression, the auxiliary power requirements at high $\beta_N > 2.5$ and at $I_p < 11$ MA may exceed the maximum P_{Aux} planned for the first stage of ITER (73 MW) [339].

5.4.3. 0D assumptions for the confinement time with prescribed density. A 0D transport simulation with 2D equilibrium solver was performed using METIS [340], where the density was prescribed, heat transport coefficients were renormalized to enforce the prescribed confinement scaling laws, and the full current diffusion equation was solved [341]. It

confirmed that the q -profile with $q(0) > 1$ can be reached at the end of the current ramp-up phase. But peaked density profiles were required to reach simultaneously the fraction of the bootstrap current above 30%, satisfying the poloidal current close to 0 [342], and Q above 5 and to sustain the q -profile above unity.

Full discharge modeling was done with CRONOS using the DS03 scaling without β degradation [343]. The density profile was prescribed assuming a strong density peaking so that $n_{e0} = 1.5n_{e,ped}$ with 92% of the Greenwald fraction and $Z_{eff} = 1.8$ with a flat profile. It showed that the target $Q = 10$ could be reached at a reduced plasma current in ITER, $I_p = 13$ MA. However, even at the reduced plasma current and high electron temperature, the $q = 1$ surface enters the plasma after about 650 s. Therefore, this simulation showed additional off-axis CD and current profile control might be needed for the ITER hybrid scenario.

Another CRONOS modeling with the higher bootstrap current provided by the peaked density profile showed that $q > 1$ in the stationary phase was attained with 12 MA of plasma current [344]. Here, $H_{98y2} = 1.3$ was assumed with off-axis ECCD and prescribed temperature pedestal width ($\rho = 0.05$) and height (4.5 keV, consistent with pedestal scaling laws [345]).

5.4.4. Solving energy transport with transport models with prescribed density. Further integrated modeling was continued with CRONOS by employing GLF23 to evaluate the anomalous energy transport which exhibited possibility of $q > 1$ with $Q > 5$ while with the pedestal temperature is higher than 4 keV [346]. The EPED1 model was additionally incorporated for pedestal width and height [341]. Here the density was prescribed and the peaking factor was scanned. It was found that H_{98y2} is around unity and the resulting bootstrap current fraction is around 30% ($\beta_N \sim 2$) which is the marginal value to maintain the q -profile above unity. The increase of the density peaking at a fixed density weakly affects the fusion performance and the ability to sustain $q(0)$ above unity for more than 1000 s.

5.4.5. Full discharge simulations. The predictive simulations including the benchmark tests have been focused upon the stationary phase but the optimization of the current ramp-up phase in terms of the waveform of the plasma current and external H&CD and the timing for L to H transition was systematically investigated to (i) optimize the q -profile at the start of the current flattop for improved fusion performance and (ii) minimize the resistive flux consumption to allow for long-pulse operation while keeping the current in the CS and PF coils within the ITER operational limits [347, 348]. The optimization of the q -profile relies on reaching a target q -profile with $q(0) \gtrsim 1$ and low magnetic shear in the core that improves stability and energy confinement.

CRONOS was used to explore the current ramp-up simulations and figure of merits, F and C_E were employed to evaluate the ramp-up scenarios [341]. Here, $F = \langle s/q \rangle q_a$, where $\langle s/q \rangle$ is the volume averaged values of s/q and C_E is the

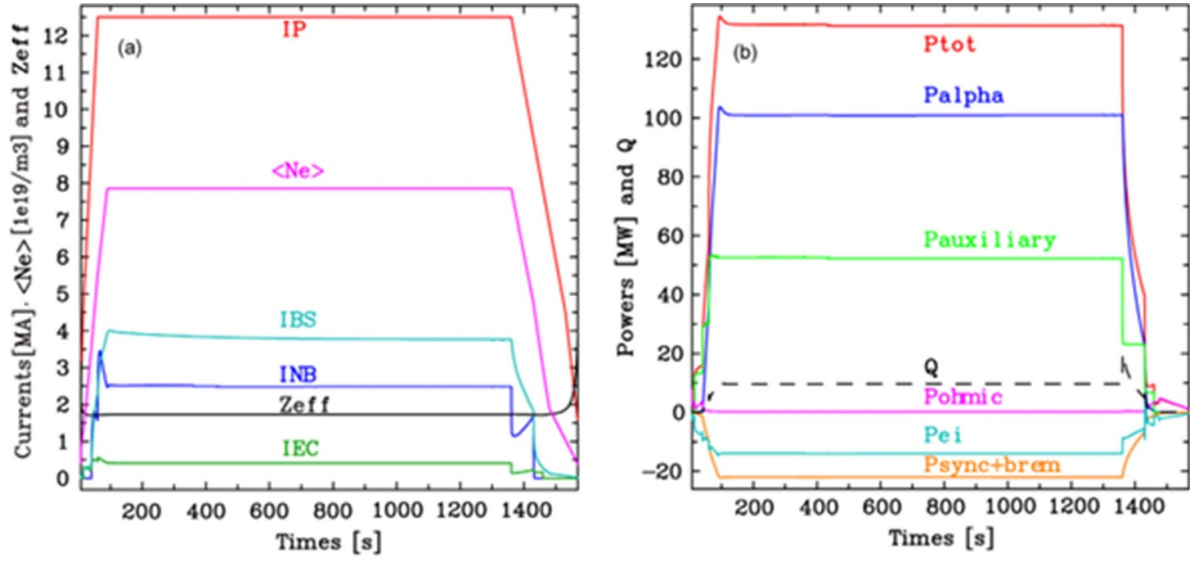


Figure 28. Predictive free-boundary modeling with CORSICA for 12.5 MA ITER hybrid scenario (reference simulation case in [359]). Reproduced from [359]. © 2016 ITER. All rights reserved.

Ejima coefficients at the end of the current ramp-up phase [349] to consider the onset of ITG and resistive flux consumption, respectively. Two transport models were used, the empirical scaling based model with $H_{98y2} = 0.4$ and the L-mode Bohm/gyroBohm model. Various simulations, i.e. (i) with and without plasma current overshoot, (ii) with early heating, (iii) with or without early H-mode transition showed that a hybrid q -profile could be reached at the end of the current ramp-up with the H&CD systems available at ITER within the operational limits, though a trade-off was needed between optimizing the q -profile and minimizing the resistive flux consumption.

Simulations using TRANSP was also performed with time evolved boundary conditions, the plasma shape, and the plasma density profiles provided by the free-boundary TSC [240, 350] for time-dependent predictive modeling of ITER hybrid scenario with $I_p = 12.5$ MA, $n_e(0) = 8.9 \times 10^{19} \text{ m}^{-3}$, $n_{el} = 8.7 \times 10^{19} \text{ m}^{-3}$ corresponding to 85% of n_{GW} , and $Z_{eff} = 2.3$ during the flat-top phase of the discharge [351]. The multi-mode model v7.1 [352–357] or the GLF23 model were employed for anomalous energy transport. The Porcelli module [358] was used to model the sawtooth crashes. Momentum transport was also solved to compute the toroidal angular frequency profile which, in turn, is used to evaluate the self-consistent flow shear suppression of anomalous transport. The simulations indicated that the hybrid q -profile was obtained with NBI and ICRF and the fusion power production was approximately 500 MW corresponding to a fusion $Q = 9.4$ at 1000 s.

5.4.6. Free-boundary simulations. Fully self-consistent free-boundary transport simulations have been performed with the advanced free-boundary transport simulation code, CORSICA, including relevant physics, engineering constraints, and ITER design parameters to provide information on the PF coil voltage demands and to study the controllability

with the ITER controllers, as shown in figure 28 [359]. The Coppi-Tang model [194, 239], was used for anomalous energy transport and the density profile was prescribed. A parameterized edge pedestal model based on EPED1 was applied. The free-boundary simulations showed that operating ITER in the hybrid operating mode is feasible with $Q > 5$ and $q(0) > 1$ within coil current field, and force limits. Same was obtained with CRONOS and DINA-CH [360].

The modeling results described in sections 5.4.2–5.4.6 are summarized in table 7. It is predicted that a $Q > 5$ hybrid scenario under the $q > 1$ constraint is feasible with the current neo-classical resistivity models. The H&CD system would be able to deliver the tailored q -profile without including any clamping mechanism. Current ramp-up and H&CD mix can be optimized in order to get specific flat-top q -profiles. LH could help to reduce flux consumption. However, it is important to note that the achievable Q and $q(0)$ values may vary depending on the pedestal assumptions and models.

The predictive core modeling also shows the benefit of the tailored hybrid q -profile on confinement, through a fundamental impact on ITG stability thresholds. It is noteworthy that the core modeling was conservative in the sense that it assumed mostly zero rotation, and no EM-stabilization of ITG. As seen in JET and AUG studies, the EM-stabilization is expected to be of more relevance for the hybrid scenario due to lower density (more suprathermal pressure fraction) and low magnetic shear in the inner-core region, which increases the EM-stabilization impact as described in chapter 2. Likely the operational margin of the ITER hybrid scenario is larger if relaxing these assumptions. However, this hasn't been studied in detail yet for ITER hybrid scenario simulations.

5.4.7. Alternative long-pulse scenario with lower density. For long-pulse scenarios in ITER ($\Delta t_{burn} \sim 1000$ s, $Q \geq 5$), as

Table 7. Summary of integrated modeling for hybrid scenario projection to ITER.

Code	Transport (and pedestal) model	I_p (MA)	$q(0)$	Q
ASTRA [314, 339]	Scaling measured kinetic profiles in AUG to ITER	9.4–14.2	< 1	$4.93 - \infty$ (depending on heating power and energy confinement scalings)
METIS [341]	Prescribed H_{98y2} and density profile	12	> 1	> 5 (depending on H_{98y2} and density peaking)
CRONOS [343]	DS3 scaling or GLF23 with a prescribed density profile	13	< 1	10–13 (depending on the density peaking)
CRONOS [344, 346]	$H_{98y2} = 1.3$ or GLF23 with a prescribed peaked density profile	12	> 1 with off-axis ECCD	> 5 (with the pedestal temperature > 4 keV)
CRONOS [341]	GLF23+EPED1 with a prescribed density profile	11.5–11.8	< 1 (reducing the volume of $q < 1$ with off-axis ECCD)	4.71–5.06
TRANSP+TSC [351]	Multi Mode model v7.1 or GLF23 with a prescribed density profile	12.5	~ 1 (Porcelli model for sawtooth)	9.3–9.4
CORSICA [359]	Coppi-Tang model with a prescribed density profile and the parameterized edge pedestal model based on EPED1	12.5	> 1	> 5

alternatives of hybrid scenario, the operational space of I_p - n_e has been assessed by 1.5D core transport modeling with pedestal parameters predicted by the EPED1 model [361] by taking into account boundary conditions predicted by SOLPS4 [184, 362]. For ITER, SOLPS predictions for high power and high density scenarios required for fusion energy production, indicate that neutral penetration into the core plasma is negligible, breaking the link between separatrix density, controlled by gas puffing for divertor load control, and pedestal density, controlled by pellet fueling [363]. These effects become more important at lower main plasma density for the separatrix values expected from SOLPS. The higher electron temperature increases the NB and EC driven currents and reduces the resistive loss, thus allowing longer pulse operation in these conditions than for the reference 15 MA at $Q = 10$. As EPED1 with the SOLPS boundary conditions predicted no degradation of the pedestal as the density is reduced in contrast to standard EPED1 assumptions, lowering the plasma density to $\sim (5-6) \times 10^{19} \text{ m}^{-3}$ led to an increased plasma temperature so to reduce the loop voltage and increase the duration of the burn phase to ~ 1000 s with $Q \geq 5$ for $I_p \geq 13$ MA at moderate normalized pressure ($\beta_N \sim 2$) [184]. Interestingly, these H-mode plasmas do not require hybrid-like q -profiles nor improved confinement in the core for the transport models considered. Thus, these medium density H-mode plasma scenarios with $I_p \geq 13$ MA could be an possible attractive alternative to hybrid scenarios to achieve ITER's long-pulse $Q \geq 5$ scenario, particularly when if the hybrid q -profile would not be reproduced in ITER. These types of discharges could increase the neutron fluence for TBM tests.

It is noteworthy that various transport and pedestal models have been employed for the aforementioned predictive integrated modeling. The validity of these models has been tested against present tokamak experiments, as reported in [293, 314, 341, 343, 364, 365], where various anomalous transport models, such as Bohm/gyroBohm, CDBM, GLF23, TGLF, and

Weiland, as well as pedestal models like EPED1 were validated. Further details are provided in the chapters 2 and 3.

5.5. Outstanding issues

There are several outstanding issues remaining which need to be addressed with further JEX/JA and collaboration with other TGs.

Firstly, the physics mechanisms of clamping q -profile against the current diffusion needs to be more extensively investigated. The favorable energy confinement and stability properties of hybrid scenarios are thought to be closely tied to the shape of the q -profile, the central safety factor above one and flat magnetic shear in the plasma core, sustained in stationary conditions. The electric conductivity and bootstrap current from the neoclassical theory could reproduce the q -profile evolution in JET [341, 366], however it could not in AUG [314], DIII-D [298], and JT-60U [367]. An example from AUG is presented in figure 29, where the safety factor values calculated using neoclassical models deviate from the experimental results. Various mechanisms have been proposed to explain this phenomenon; hyper-resistive term [368], enhanced neoclassical resistivity due to the (1, 1) mode [367], flux pumping by tearing mode or interchange mode or ELM-NTM coupling [369–373] and helical core equilibria [374, 375]. Self-generated current sources from turbulence [376–378] or unknown mechanism [379] at the off-axis region could be another possible mechanism. Revision of neoclassical formulations would be a potential path to identify the mechanism using a closure theory [380]. But they need to be evaluated in various devices prior to projection to ITER. Even if we identify the mechanism, it remains uncertain whether the same mechanism will be present and produce the same effects in ITER. This uncertainty arises because mechanisms, including non-ideal MHD effects, cannot be directly extrapolated from present experiments to ITER due to its distinct dimensionless

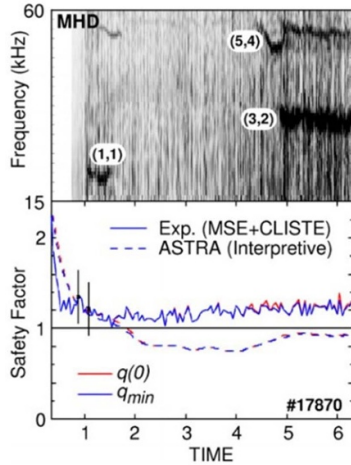


Figure 29. Anomalous current diffusion in hybrid scenario on AUG. Reproduced from [314]. © 2006 IAEA, Vienna. All rights reserved.

plasma parameters. In addition, assessing the optimal range of q – and magnetic shear profiles for given temperature, density, and rotation profiles, with respect to stability and transport, is a valuable direction for future research.

Secondly, the mechanism behind confinement enhancement requires further clarification. Although the energy confinement scaling study has been conducted via JEXs, the database was insufficient to derive a robust scaling law. Additional experimental data are needed, particularly for conditions with high Greenwald density fractions, high electron heating fractions, and low input torque by new experiments. Furthermore, the beta dependency of the energy confinement should be clarified in collaboration with T&C and PEDELM with core-edge coupled integrated modeling as in [381]. Dedicated power scan experiments are required as part of a new JEX, similar to those conducted at JET. In these experiments, power scans at $I_p = 1.4$ MA and $B_t = 1.7$ T revealed that beta effects, with significant contributions from fast ions at high power, were key mechanisms driving transitions to high confinement in both the core and the pedestal regions [283]. Dedicated simulations of these plasmas indicated that such effects could lead to high fusion power in DT [382]. The various effects of $E \times B$ shearing rate and zonal flows, alpha stabilization, fast ions, pedestal dynamics, etc need to be quantitatively evaluated at each radial position [277] for accurate projection to ITER. It is important to note that particle and momentum confinement have not been as thoroughly studied as energy confinement in hybrid scenarios. Some predictions have shown high sensitivity to toroidal rotation when transport models rely on $E \times B$ flow shear stabilization [364].

Thirdly, the dependency on wall material and isotope effects needs to be explored. The choice of wall material can highly influence the experimental approach to establish the hybrid regime, particularly in avoiding severe impurity influx. JET provides a good example of this. Although the confinement analysis with the collected database includes the AUG dataset from operations with tungsten as the dominant plasma facing material [383], it does not include the

JET with ILW. Regarding the isotope effects, the DTE2 campaign at JET has demonstrated that access to the hybrid scenario can be impacted by the presence of T, and that well-established recipes in D may require adjustment in DT operations. Nevertheless, the hybrid scenario was successfully developed in DT during DTE2, even achieving higher confinement compared to pure D plasmas. However, more systematic studies are required to fully assess the evolution of the q -profile, L–H transition, and pedestal dynamics within the scenario, to better understand these effects in DT plasmas.

Lastly, predictive integrated modeling of hybrid scenarios needs to be further enhanced. Although the status of modeling for the hybrid scenario has reached a high level of maturity, demonstrating broad agreement of fusion power in the JET DT plasmas with previous modeling results obtained long before DTE2 campaign [384], more studies are necessary to better capture conditions that align with expected ITER plasma environments, thereby improving the extrapolability of present-day models (see section 4). In addition, further understanding of the aforementioned origin of the confinement enhancement is required. Recent mechanisms contributing to improved confinement in hybrid scenarios are not accounted for in older transport models used in old modeling. As the shape of the q -profile can affect confinement and stability, detailed simulations of current diffusion are essential. Developing models for flux pumping, the characteristics of which remain largely unknown in ITER, is highly on demand. It is worth noting that little benchmarking of models has been conducted in predominantly electron-heated plasmas. Tungsten transport analysis is another critical issue in the modeling of hybrid scenarios [385] that needs to be extended to core-pedestal-edge-SOL-divertor/FW integrated simulations. Alongside transport, modeling is needed for higher stability against non-ideal MHD modes, such as frequently interrupted mode (FIR) [386]. The role of CEM also needs to be clarified to understand and model the transition to the hybrid regime [292]. With respect to control simulations, feedback control is necessary during the entry to burn for density and impurities due to extreme non-linear behavior. Although the ITER coil system can handle hybrid regimes with high confinement (and high β_p) from current ramp-up to ramp-down [359], special plasma control will be required to manage high β_p during current ramp-down.

5.6. Conclusions

The hybrid scenario was originally proposed to extend the current flat-top duration at lower plasma currents, situated between the baseline and steady-state scenarios in ITER. Due to its intermediary position between baseline inductive and fully non-inductive operations, it was named the ‘hybrid’ scenario. Experiments conducted in various tokamaks have demonstrated that the candidate hybrid scenarios exhibit higher performance levels compared to conventional H-mode operations. Consequently, advanced hybrid scenario was proposed, targeting a higher fusion gain of $Q \sim 10$ at a reduced plasma current of less than 15 MA, and advanced inductive scenario aiming for $Q > 20$ at 15 MA in ITER.

In this paper, discharges that satisfy the followings are categorized into hybrid scenario; (i) $q(0) \sim 1$ and low magnetic shear in the central region with $0.6 \leq l_i(3) \leq 1.2$, (ii) $3.0 \leq q_{95} \leq 6.5$ where sawtooth appears inherently if operated in conventional H-mode, (iii) without or with mild sawtooth activities, (iv) $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$ above the IBL scenario, (v) stationarity with durations longer than $5\tau_E$. The experiments conducted for hybrid scenarios in various tokamaks reveal that the hybrid scenario represents a robust mode of operation in divertor tokamaks. Hybrid scenarios have been achieved across a wide range of operational windows, demonstrating high performance in stationary conditions on the time scale of a fully relaxed current profile. The early heating approach is routinely used and works well to establish hybrid scenarios in many devices, except for JET, where the current overshoot approach has primarily been employed to achieve improved confinement. Understanding these approaches is challenging due to the non-linear relationship between the q - and magnetic shear profiles with transport and stability. However, it is essential, as it is linked to the self-organization mechanism of the current density profile in hybrid scenarios.

Since 2007, development of integrated hybrid scenarios have been actively explored in ITER-relevant conditions in terms of the low external torque, RMP ELM suppression, radiative divertor, long-pulse operations, and DT. Even steady-state conditions are being explored. The results show promise, however, since they have been conducted individually, integrating them into a single discharge is one of the future demands. Although engineering-related issues remain for long-pulse operations exceeding 30 s, good progress has been made toward fulfilling the fluence goals of the ITER project. This may also provide a lower-risk alternative approach to achieving the primary physics objective of $Q = 10$ operation at 500 MW fusion power for 400 s.

Apart from experimental achievements in each device, the issues of high confinement, MHD stability, and maintenance of the q -profile have been addressed via international collaboration in the framework of ITPA IOS TG. The collection of database from various devices and the dedicated experiments indicates a weaker scaling with normalized gyroradius and a stronger scaling with collision frequency than the standard H-mode scaling. However, the database was found to be insufficiently conditioned to yield a new scaling.

Projections of hybrid scenario have been attempted with integrated modeling, starting from benchmark studies. Modeling results with various integrated suites such as ASTRA, CORSICA, CRONOS, METIS, ONETWO, TOPICS, and TSC/TRANSP showed that a hybrid scenario with $Q > 5$ and $q > 1$ is feasible with the anticipated H&CD system in ITER, under certain assumptions, including the pedestal temperature. As an alternative scenario, by lowering the plasma density, the duration of the burn phase was extended to approximately 1000 s with $Q \geq 5$ for $I_p \geq 13$ MA at moderate normalized pressure, even without a hybrid-like q -profile. This may be helpful if the assumptions turn out to be optimistic when improved models become available.

The robustness and progress of this scenario is encouraging. Particularly, establishment of hybrid scenarios under ITER-relevant conditions, such as low external torque and DT operation, is promising when extrapolating results from present devices to ITER. However, the high performance observed in current experiments does not guarantee that similar high confinement and β_N can be achieved in ITER. This is due to the non-linear interplay between ITER's specific H&CD systems and plasma stability, as well as transport through magnetic and kinetic profiles, as shown in figure 1. Moreover, none of the present experiments are fully ITER-relevant, as they cannot completely replicate ITER plasma conditions. The development of an integrated hybrid scenario is machine-specific. For example, RMP ELM suppression and radiative divertor behavior sometimes show inconsistencies between machines, and long-pulse operation depends on the wall material. In this context, integrated modeling is useful for addressing these issues, though it could be limited by model constraints and simulation assumptions.

Additionally, several outstanding issues remain. The mechanisms governing q -profile clamping need further clarification through new JEXs or JAs. The physics behind the confinement enhancement should be further elucidated in collaboration with the T&C and PEDELM TGs. Additionally, the dependency on wall materials and isotope effects needs to be explored. Lastly, predictive integrated modeling of hybrid scenarios requires further enhancement with more advanced models toward core-pedestal-edge-SOL-divertor/FW integrated simulations, utilizing the results of the JET DT experiments.

The hybrid scenario has been mainly developed to achieve high fusion gain with $q_{95} \sim 4$. However, steady-state hybrid scenarios with elevated q_{95} have been established mainly in DIII-D. In the next section, scenarios aimed at steady-state operation will be discussed, focusing on reduced plasma current or increased q_{95} .

6. Steady-state scenarios

Coordinated by A.M. Garofalo

6.1. Introduction

It is expected that several years of ITER operations will be available for long-pulse and steady-state research preparing the way for economical fusion reactors, possibly commencing already while pursuing the achievement of ITER's main $Q = 10$ performance objective. Steady-state scenario research will aim at optimizing performance (with target $Q = 5$) and duration (up to 3000 s) in discharges where the current is driven fully non-inductively [10, 11]. Two main options for steady-state operation in ITER were discussed in [11]; scenarios with strong reversed shear and high bootstrap current (high bootstrap fraction, f_{BS} , scenarios), and scenarios with weak reversed or low magnetic shear (high auxiliary power fraction, f_{Aux} , scenarios).

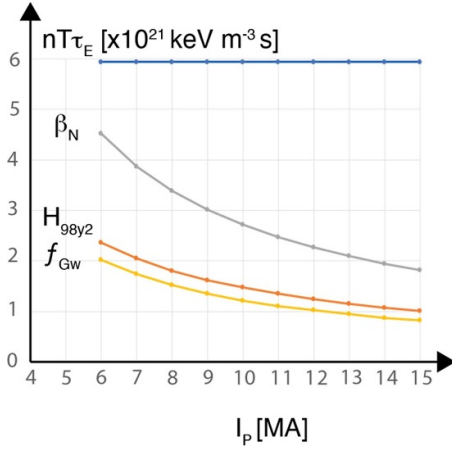


Figure 30. Schematic illustration of the required changes in H_{98y2} , β_N , and f_{GW} relative to the 15 MA ITER Baseline target to maintain each component of the fusion triple product constant with lower plasma current and the same heating power.

In both cases, the plasma current I_p needs to be significantly reduced from the IBL value of 15 MA, in order to facilitate reaching fully non-inductive drive. The reduction in plasma current (at constant toroidal field B_t and plasma geometry) results in the following three main challenges, assuming that one would want to minimize the reduction in fusion triple product, $n\tau_E T$:

- To increase the energy confinement quality ($H_{\text{scaling}} = \tau_E / \tau_{E,\text{scaling}}$) so that the energy confinement time is approximately maintained as I_p is reduced. Note that $\tau_{E,\text{scaling}}$ usually has a strong dependence on I_p , e.g. $\tau_{E,98y2} \sim I_p^{0.93}$.
- To increase β_N , so that β_i is approximately maintained as I_p is reduced,
- To increase the Greenwald fraction (f_{GW}), so that the density is approximately maintained as I_p is reduced.

Note that the Greenwald density limit only applies to the edge density [28, 387], thus a line averaged $f_{GW} > 1$ can be achieved without violating the Greenwald density limit. Figure 30 illustrates the required changes in H_{98y2} , β_N , and τ_E to maintain constant fusion triple product as the plasma current is reduced from the IBL target, assuming simple 0D physics. Plasma current values lower than 6 MA are not considered here, because it is expected that the fast ion losses would become significant [388]. In practice, these requirements can be relaxed somewhat because a lower level of fusion performance ($Q = 5$ instead of $Q = 10$) and thus a lower fusion triple product, is targeted for the long-pulse, fully non-inductive ITER demonstration (note that Q and triple product are correlated but not directly proportional [389]).

The three requirements discussed above work synergistically in the advanced tokamak approach to a fusion reactor [390]. Higher β_N at lower I_p means higher poloidal beta, β_p , which increases the alpha stabilization of turbulence and thus increases H_{98y2} [391–393]. With the edge density limited by the Greenwald value, a high Greenwald fraction implies high

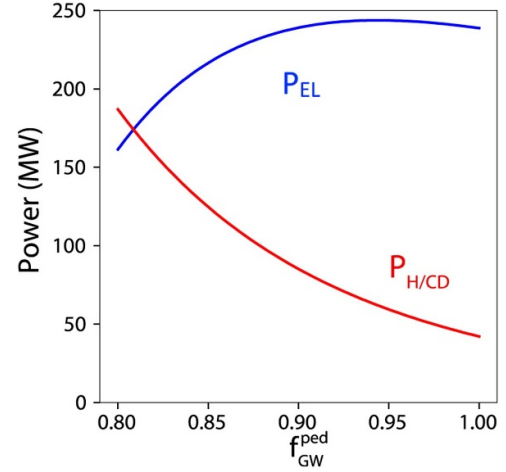


Figure 31. Benefit of a rise in Greenwald density fraction in FASTRAN code simulations of a compact AT reactor producing about 200 MW of electric power (P_{EL}). Note that the higher f_{GW}^{ped} corresponds to slightly lower absolute density in these simulations, since the plasma current is decreasing at the higher f_{GW}^{ped} with constant β_N . Reproduced from [395]. © 2021 IAEA, Vienna CC BY 3.0.

core density gradients, predicted to amplify alpha stabilization at high β_p [394], and thus contributing strongly to increasing the energy confinement quality. In turn, high H_{98y2} enables high β_N at lower I_p with fixed heating power. Note that a high Greenwald fraction is not unfavorable to fully non-inductive operation, even though higher density naturally reduces the efficiency of external CD. Illuminating on this issue is the FASTRAN code modeling discussed in [395], of which figure 3 is reproduced here in figure 31. This figure shows that, as the Greenwald density fraction rises in simulations of a compact AT reactor producing about 200 MW of electric power, the level of auxiliary power needed to maintain fully non-inductive conditions at constant β_N decreases rapidly. This is due to mainly two effects. On one hand, density gradients are more efficient than temperature gradients in driving bootstrap current [396]. On the other hand, higher density increases the fusion power more strongly than the higher temperature (in the right temperature range), leading to new solutions with lower plasma current and similar net electric power output.

The high f_{BS} and the high f_{Aux} scenarios are different approaches to the ultimate advanced tokamak envisioned in [390]. The former favors lower auxiliary power at lower plasma current, at a cost for the fusion power output since at fixed toroidal field and plasma geometry the fusion power density scales as $P_{fus} \sim \beta_i^2 \sim \beta_N^2 I_p^2$, with β_N ultimately limited by MHD stability. The latter favors higher fusion power density at higher plasma current, at the expense of the fusion gain because of the high energetic cost of the external off-axis CD needed to maintain the scenario. This is illustrated schematically in figure 32.

In general, the high f_{BS} approach is characterized by $q_{min} > 2$ and $f_{BS} > 80\%$ in a reactor. The high f_{Aux} approach includes $f_{BS} < 80\%$ scenarios with high q_{min} ($q_{min} > 1.5$) and with $q_{min} \sim 1$, such as the ‘steady-state hybrid’ and the ‘high I_i ’ scenarios. Towards an advanced tokamak fusion reactor, the

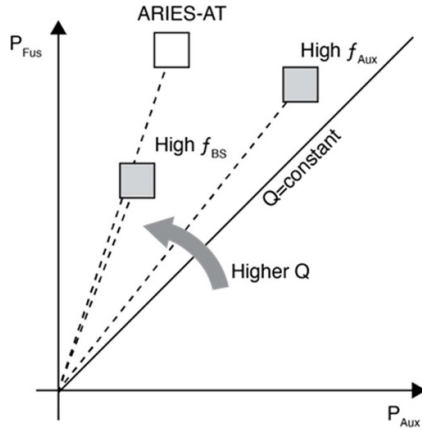


Figure 32. Schematic illustration of the different high f_{BS} and high f_{Aux} approaches to the ultimate AT scenario envisioned in [390].

choice between the different approaches will depend on considerations of full core-edge-boundary plasma integration, and eventually on considerations of capital cost, balance of plant, and the requirements of the electricity grid. Towards the ITER steady-state goal, both approaches could provide a viable path.

6.2. Experiments in support of ITER

Since the 2007 IPB [397], significant progress has been achieved in experiments around the world towards the demonstration of high normalized fusion performance in steady-state. In particular, two superconducting tokamaks, EAST in China and KSTAR in Korea, have begun high power plasma experiments aiming to develop and demonstrate ITER-relevant physics and technologies for long-pulse operation.

Note, however, that no experiment has achieved and sustained in steady-state and in an ITER similar shape, the levels of normalized fusion performance necessary for $Q = 5$ in ITER. Fully non-inductive discharges tend to be somewhat low in performance, while high performance discharges tend to be somewhat low in non-inductive current fraction [398]. Furthermore, discharges in ITER similar shape tend to be somewhat lower in performance than discharges with stronger shaping. Some scenarios, notably the steady-state hybrid and the high- I_i scenarios, can yield very good and robust normalized fusion performance in present-day tokamaks, alas theory-based modeling of these scenarios on ITER is either lacking or does not reach the desired ITER performance goals, as will be discussed in section 6.3.

AUG. Since completion of the all-W wall cover on AUG in about 2007 [399], the activity on advanced scenario development was stopped until ~ 2012 . In 2012, an improved H-mode (also named hybrid scenario) with AT like parameters $H_{98y2} > 2$, $\beta_N > 4$ (transiently) has been found [269]. Even though high beta and confinement were reached, the non-inductive fraction was not very high.

With the availability of more ECRH power [270], improved H-modes with a weak 3/2 mode similar to [266] could be re-established [271]. Based on theoretical work [372] on the

dynamo effect related to (1,1) interchange modes, focus of the flux-pumping experiments returned to (1,1) fishbone driven improved H-modes, for which flux pumping was quantified in dependence of β and central ECCD [373]. Note that the flux pumping has been discussed in sections 5 and 7 as well. In parallel, work on plasmas with q_{min} above unity resulted in a scenario with almost completely non-inductive CD at currents of 600 kA (shown in figure 33, left panel) and 800 kA [400, 401]. Values of $q_{95} \approx 5.3$ and $\beta_N = 2.7$ have been achieved, with $1.1 < H_{98y2} < 1.2$ and reasonable stability for $\beta_N < 2.7$, independent of the initial conditions (shown in figure 33, right panel). Above $\beta_N \approx 2.7$, ideal and resistive MHD instabilities appear. These discharges, free of (1,1) modes and sawteeth, show centrally peaked T_i -profiles. This can be reproduced with GENE, but not reliably with TGLF [402, 403]. Only in cases with strong $E \times B$ shear TGLF gives the right gradients, while GENE indicates that the $E \times B$ stabilization should be weak also in those cases [318]. Comparing NBI and ICRF heated cases to vary the $E \times B$ shear, GENE clearly shows that the dominant stabilizing effect at least for that operational range is due to the fast ion energy distributions of H and D [404]. This suggests that high-fidelity modeling of ITER equilibria should be carried out in order to assess the significance of these fast ion effects, even though ITER is expected to have smaller fast ion beta than these experiments.

Finally, the dependence of the results due to variations of the machine conditions has been studied systematically, looking at the remaining recycling fluxes at zero gas flux. It turned out that the pedestal conditions and with them the confinement are very sensitive to small variations of these fluxes [405].

DIII-D. Progress on DIII-D also relied on improved capabilities. In 2011, a massive hydraulic lift was installed beneath one of the NBI lines to allow redirecting up to 5 MW of NBI power and CD from on- to off-axis, to sustain broader current and pressure profiles.

Nearly stationary plasmas with good confinement ($H_{98y2} \geq 1.3$) and q_{min} near 1.4 have been sustained at $\beta_N \sim 3.5$ and $\beta_t \sim 3.6\%$ for two current profile diffusion time scales (about 3 s) limited by NB duration [406], as shown in figure 34. However, this high normalized fusion performance is achieved in a high triangularity double null divertor shape (not an ITER similar shape). The non-inductive current fraction f_{NI} is 70%–75% in these plasmas. To compensate for the non-inductive fraction deficiency, DIII-D requires additional external off-axis CD. Toward this, a second line of NBI was redirected from on- to off-axis, yielding encouraging initial results. Specifically, off-axis NBI broadened energetic particle profiles leading to better control of AEs in high- q_{min} plasmas, more classical fast ion confinement, and higher ideal-wall kink mode β_N limits, in some cases resulting in higher achieved β_N . Net non-inductive plasmas were achieved with $q_{min} \sim 1.3$, $\beta_N \sim 3.7$, and $q_{95} \sim 6.6$, limited by available ECCD [407, 408].

Compared to the high f_{Aux} scenario above, the high f_{BS} scenario can more easily achieve fully non-inductive conditions by operating at somewhat reduced plasma current, but at a cost for the normalized fusion performance. An example of the DIII-D high β_p scenario is shown in figure 35. Also in this

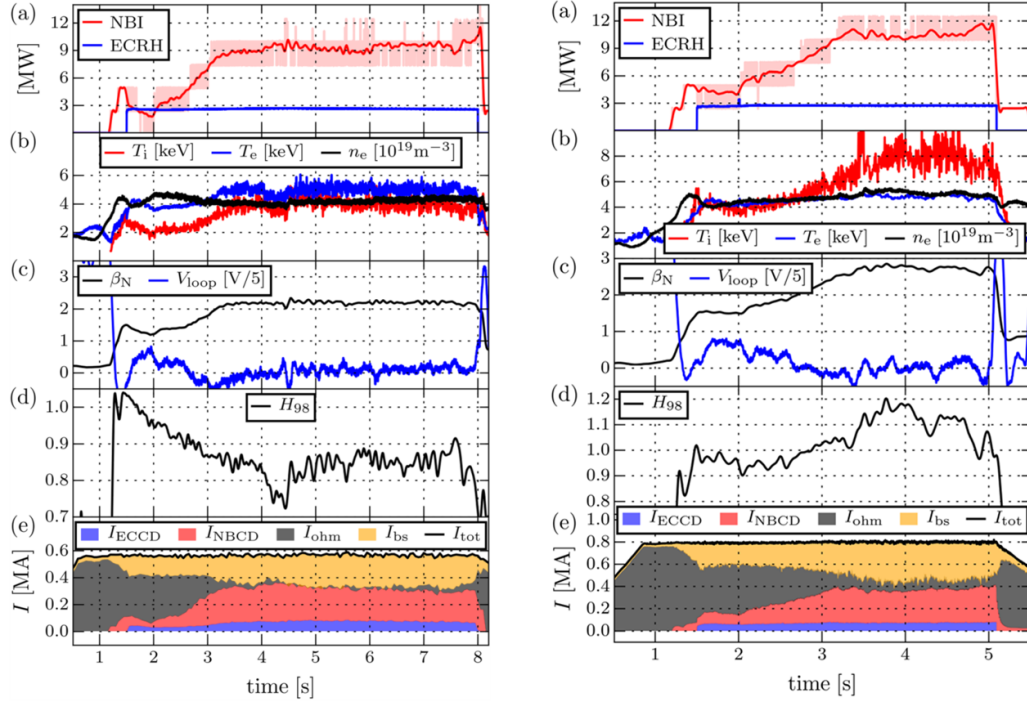


Figure 33. (a) Heating power. (b) Core electron/ion temperatures, electron density. (c) β_N and loop voltage. (d) Confinement quality as given by H_{98y2} . (e) Composition of toroidal plasma current. Left panel is for 600 kA discharge #33 379 (Reproduced from [401], with the permission of AIP Publishing). Right panel is for 800 kA discharge #32 305. Reproduced from [400]. © 2017 IAEA, Vienna. All rights reserved.

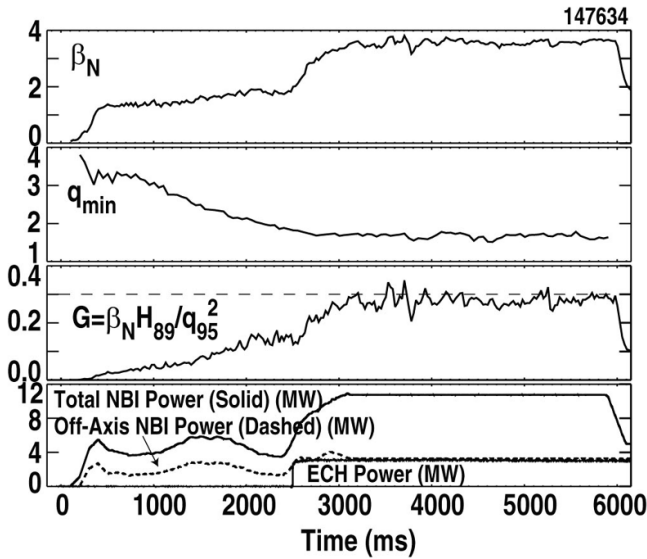


Figure 34. Time traces of key parameters for quasi-stationary DIII-D discharge 147 634. (a) Normalized beta (β_N), (b) minimum safety factor ($q_{\min}$), (c) normalized fusion performance ($\beta_N H_{89}/q_{95}^2$), (d) injected auxiliary power. Reproduced from [406]. © 2014 IAEA, Vienna. All rights reserved.

case, the plasma cross-section is a double null divertor shape. After an initial high- $q_{\min}$ equilibrium is established (~ 1.7 s) the current in the ohmic heating coil is held constant to remove any inductive CD. This discharge achieves and maintains $\beta_p \sim \beta_N \sim 3.3$ with $f_{BS} \geq 70\%$, a good energy confinement quality

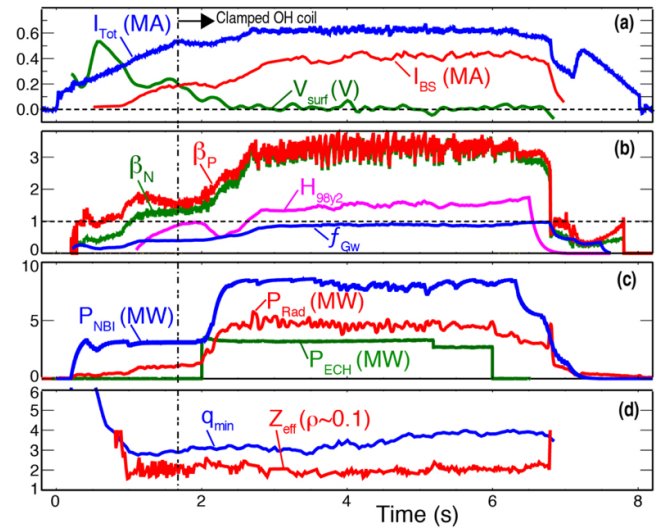


Figure 35. Time traces of key parameters for high β_p DIII-D discharge 154405; (a) total plasma current, bootstrap component, and surface toroidal voltage; (b) β_p , β_N , H_{98y2} , and the Greenwald fraction; (c) total radiated power, NBI power, and EC power; (d) minimum safety factor and core Z_{eff} from visible bremsstrahlung [409]. Reproduced from [409]. © IOP Publishing Ltd All rights reserved.

($H_{98y2} \sim 1.5$) associated with the formation of an ITB at large normalized minor radius ($\rho = r/a \sim 0.7$) in all channels (density, temperature, rotation), Greenwald density fraction $f_{GW} \sim 1$, and constant radiated power and Z_{eff} , while in fully non-inductive conditions for about three current relaxation times

($\tau_{CR} \sim 1$ s) throughout the high β phase [409]. At $\beta_t \sim 1.8$, the normalized fusion performance may not suffice for $Q = 5$ in ITER ($\beta_t \geq 2.0$). However, these experiments have led to increased physics understanding of the high f_{BS} scenario, highlighting features that are essential towards the realization of a steady-state fusion reactor, such as; excellent confinement quality that does not rely on plasma rotation shear [393, 409], high normalized beta limits and no core accumulation of high-Z impurities despite the presence of ITBs [410, 411], compatibility of excellent core confinement with low edge pedestal (for small ELMs) and detached divertor operation [412, 413]. More recent experiments have also extended this scenario towards higher β_t (> 3.0) and higher H_{98y2} (> 1.5), albeit for shorter durations (~ 1 s) until now [414], and to an ITER-similar single null divertor shape, with good performance ($\beta_t \sim 1.8$, $H_{98y2} \sim 1.1$) sustained simultaneously with full divertor detachment and ELM suppression achieved for ~ 1 s by neon impurity injection [414].

DIII-D experiments have also investigated the potential of high internal inductance (l_i) and hybrid scenarios for steady-state operation of ITER. Fully non-inductive hybrids with high beta ($\beta \leq 2.8\%$) and high confinement ($H_{98y2} \leq 1.4$) in the ITER similar shape have achieved zero surface loop voltage for up to two current relaxation times using efficient central CD from ECCD and NBCD [273] as discussed in section 5. This scenario was successfully integrated with ELM suppression by applying an odd parity $n = 3$ RMP with little confinement degradation. The high l_i scenario, also tested in the ITER shape, has reached, on DIII-D, performance appropriate for the ITER $Q = 5$ mission, with $\beta_N \approx 3.5$, $H_{98y2} \sim 1.3$, $q_{95} \sim 5$ sustained for about 1 s, but with non-inductive fraction $f_{NI} \sim 0.7$ [415].

EAST. Figure 36 shows the rapid progress during these last few years towards longer duration and higher performance H-mode discharges on EAST [252, 416–421]. With features such as dominant electron heating, zero/low NBI torque, and an ITER-like tungsten divertor, EAST can offer unique contributions to research in support of long-pulse operation in ITER and future fusion reactors.

A record duration of 310 s fully non-inductive H-mode plasma ($H_{98y2} \sim 1.3$, $n_e/n_{GW} \sim 0.7$, $f_{BS} > 50\%$) was recently achieved on EAST, exploiting several improved long-pulse capabilities [421]. This steady-state discharge achieves a significant normalized fusion performance ($H_{98y2} \sim 1.3$, $n_e/n_{GW} \sim 0.7$, $f_{BS} > 50\%$, $\beta_p \sim 2.5$, $\beta_N \sim 1.6$) close to the target for CFETR steady-state scenario at fusion power near 1 GW [5]. This performance is not sufficient for $Q = 5$ in ITER, but the experiment provides both physics and technology answers that are essential for ITER, including: improved energy confinement quality with respect to standard H-mode, despite using zero torque, wave-only power injection for fully non-inductive CD; good control of high-Z impurities in the core using full metal wall, and good particle and power exhaust using water cooled tungsten divertors in long-pulse operation. Representative time-traces of this discharge are shown in figure 37. The ultimate limitation for this long-pulse H-mode plasma is due to hot spots forming in the lower divertor due

to leading edges, and also to sputtering and erosion of the RF-antenna and guard limiter.

These limitations have been overcome in more recent experiments that have achieved a reproducible fully non-inductive H-mode duration of 403 s [422], after correction of the leading edges, and Lithium coating optimization that enables maintaining lower neutral densities near the antennas to mitigate sputtering and erosion. In an even more recent experimental campaign, Lithium coating was replaced with boronization. These joint ITER-EAST experiments have yielded a wealth of data that will be analyzed to address specific ITER questions on the reliability and operability of tungsten as a plasma-facing material [423].

JET. Experiments on JET using NBI heating (only) at $I_p/B_t = 1.1$ – 1.6 MA/1.6–2.3 T identified two favorable domains for high β_N , H_{98y2} operation; one with safety factor on axis $1 < q(0) < 1.5$, where $H_{98y2} = 1.35$ was achieved, and a second domain with $q(0) \geq 2$, where $H_{98y2} = 1.25$. The non-inductive component is typically 40%–60% in these experiments and the application of further non-inductive drive is required in the region $\rho = 0.4$ – 0.6 to maintain stability [424].

The experiments with $q_{min} < 1.5$ achieved good stability at $\beta_N \sim 4$ (above the expected no-wall limit of $4l_i$) for many τ_E , as shown in figure 38. In these plasmas the current density profile broadened during the heating phase, gradually shrinking the radius of the $q = 2$ surface until the 2/1 mode became unstable.

The experiments with $q_{min} \sim 2$ achieved $\beta_N \sim 3$ for many τ_E [425]. The duration of the high performance phase was typically limited due to the slow q -profile evolution into the $q_{min} < 2$ domain, in accordance with modeling suggesting that the gradual increase in magnetic shear at the $q = 2$ surface eventually leads to instability [426].

This second domain was later selected in order to demonstrate the scenario in plasmas with dimensionless parameters such as the normalized Larmor radius (ρ^*), collisionality (ν^*), ratio of ion to electron temperature (T_i/T_e), and normalized rotation as close as possible to those expected in ITER [427]. These experiments used higher toroidal field 2.7 T and higher plasma current 1.8 MA, and used ICRF and LH wave heating in addition to NBI heating. This produced plasmas with steady (i.e. for times $> 10\tau_E$) $H_{98y2} = 1.2$, total and thermal β_N of 2.7 and 2.0, respectively, $\langle T_i \rangle / \langle T_e \rangle = 1.05$, $f_{BS} = 40\%$ at global $\rho^* / \rho_{ITER}^* \approx 2$, $\nu^* / \nu_{ITER}^* \approx 4$, achieved simultaneously for the first time (shown in figure 39). The improved confinement leading to $H_{98y2} > 1$ is due to the presence of an ITB in the ion temperature profile.

The above JET experiments were carried out with a carbon wall. More recently, an NBI operation scenario was developed in order to achieve the formation of elevated q -profile plasmas also in the JET-ILW. For the first time since the installation of the ILW, ITBs have been clearly observed in a number of discharges performed in the framework of this experimental effort [428].

JT-60U. On JT-60U, experiments during the preparation of the 2007 IPB [397] and up to the facility shutdown in 2008, focused on expansion of the operational space of AT plasmas

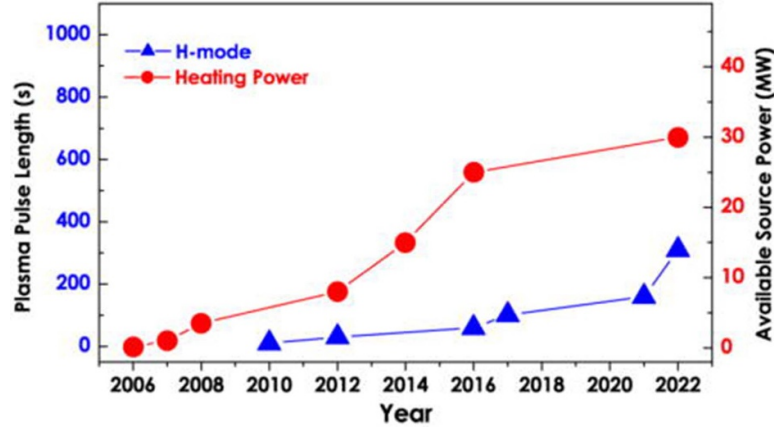


Figure 36. Progress of the pulse length (blue triangle) for H-mode plasmas on EAST and the total available source power (red dot) of the auxiliary H&CD systems since the first operation in 2006. Reproduced from [421]. CC BY 4.0.

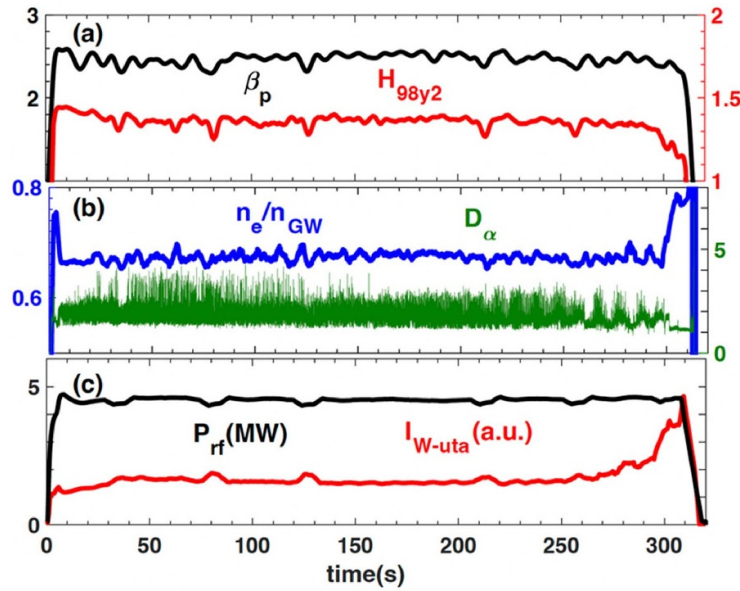


Figure 37. Time traces of key parameters for the EAST record-long H-mode discharge, #110488. (a) Poloidal beta (β_p), confinement enhancement factor over H-mode confinement scaling (H_{98y2}), (b) Greenwald fraction (n_e/n_{GW}), D_α signal, (c) injected power of RF (P_{rf}), intensity of tungsten unresolved transition array (W-UTA) I_{W-uta} . Reproduced from [421]. CC BY 4.0.

towards higher β_N . This was enabled by improvements in the high power HSs in 2007, together with the installation in 2005 of ferritic steel tiles to reduce fast ion losses due to toroidal field ripple, which allowed higher β_N in a large volume configuration with reduced plasma-wall distance [429].

In weak magnetic shear plasmas (high f_{aux} scenario), high $\beta_N \sim 3$ exceeding the $n = 1$ ideal MHD limit without a conducting wall ($\beta_N^{no-wall}$) was sustained for ~ 5 s ($\sim 3\tau_R$) with non-inductive current fraction $f_{NI} \sim 0.8$ – 0.9 , and with RWM stabilization by toroidal rotation [430]. Waveforms of a typical high β_N discharge are shown in figure 40.

In reversed shear plasmas (high f_{BS} scenario), utilizing wall stabilization enabled expanding the operational regime towards lower q_{95} and higher β_N . In the example shown in figure 41, high $\beta_N \sim 2.7$ and $\beta_p \sim 2.3$ are achieved, with $q_{min} \sim 2.4$ ($q_{95} \sim 5.3$), and $f_{BS} \sim 0.9$, although the high performance is transient and terminated by a disruption [431]. The

achieved β_N is much higher than that obtained in previous high f_{BS} scenario experiments that were limited to $\beta_N \sim 1.7$ – 2.2 . The disruption is caused by the RWM destabilized because of very low toroidal rotation resulting from the near negligible injected torque (using balanced tangential NBI). In ITER, the co- I_p NBI torque may provide sufficient rotation to maintain RWM stabilization. These results suggest that in future reactors without external torque injection, an RWM control scheme should be developed for steady-state high f_{BS} plasma operation. A high energy confinement quality, $H_{98y2} \sim 1.7$, is obtained with internal and edge transport barriers, even with the very low toroidal rotation. The integrated plasma performance nearly satisfies the requirements for ITER steady-state operation, except for the fuel purity. In particular, the high integrated performance is achieved under the reactor-relevant conditions of $T_e \sim T_i$, low momentum input, and high Greenwald fraction, $n_e/n_{GW} \sim 0.9$.

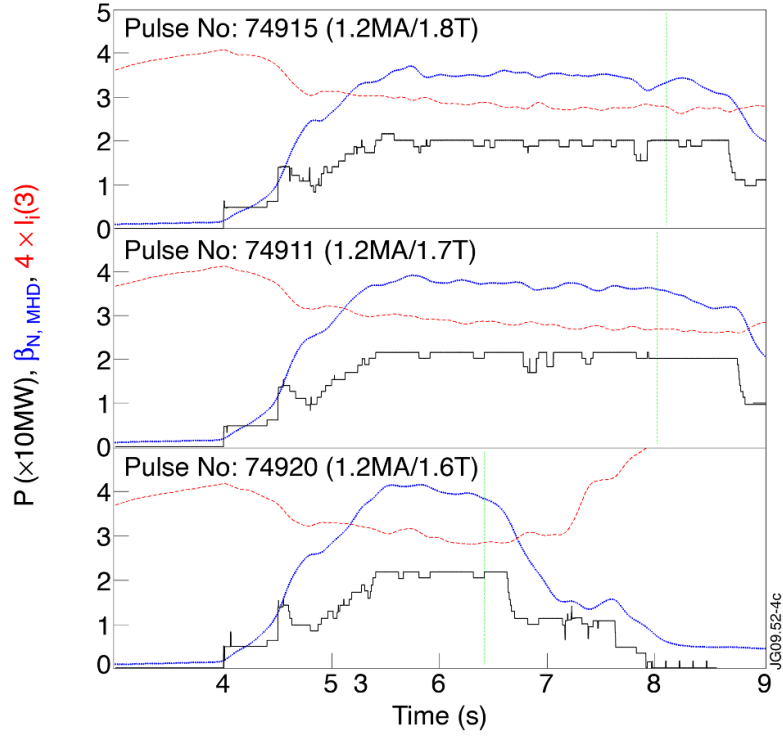


Figure 38. Time traces of normalized beta (β_N), NBI power, and $4I_i$ for JET discharges with $q_{\min} \sim 1.0$ – 1.5 . Vertical dashed lines indicate onset of 2/1 mode. Reproduced with permission from [424].

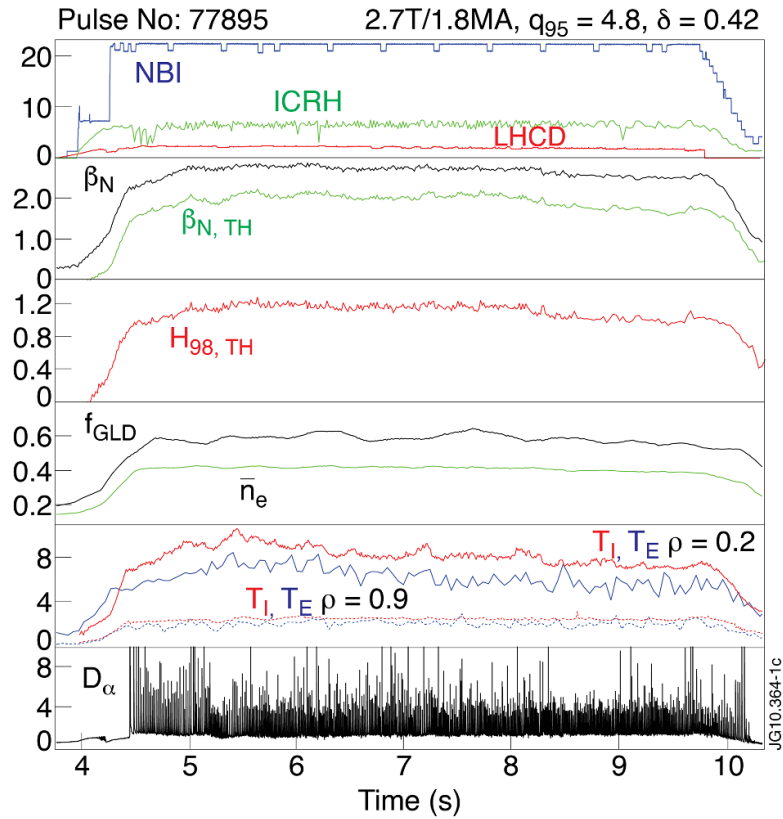


Figure 39. Time traces of key parameters for JET discharge #77895. Injected powers from NBI, ICRF, and LH heating, total and thermal normalized beta, confinement enhancement factor over H-mode confinement scaling, line averaged electron density in 10^{20} m^{-3} and Greenwald fraction, ion and electron temperatures in the core and near the edge, D_α light signal. Reproduced with permission from [427].

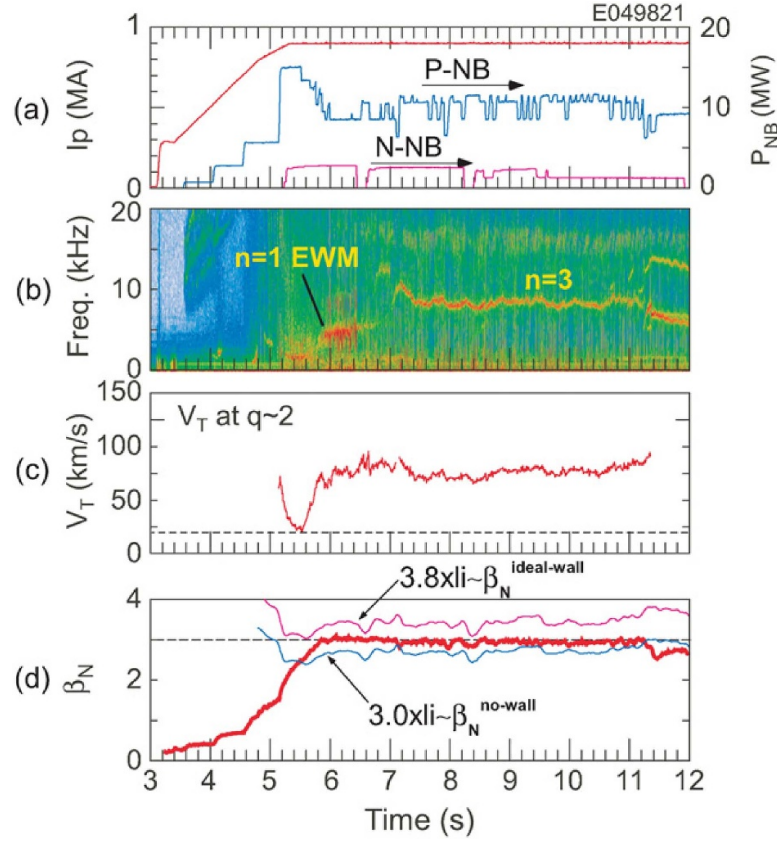


Figure 40. Waveforms of high β_N discharge exceeding $\beta_N^{\text{no-wall}}$. (a) Plasma current and NB heating power for both positive-ion based NBI (P-NB) and negative-ion based NBI (N-NB). (b) Spectrogram of magnetic fluctuations. (c) Toroidal rotation measured near $q = 2$ surface. The dashed line indicates the required toroidal rotation speed for RWM stabilization. (d) Normalized beta, measures of the ideal wall limit ($\sim 3.8l_i$) and the ideal MHD limit without conducting wall ($\sim 3.0l_i$). The dashed line indicates the normalized beta of 3. Reproduced from [429]. © 2009 IAEA, Vienna. All rights reserved.

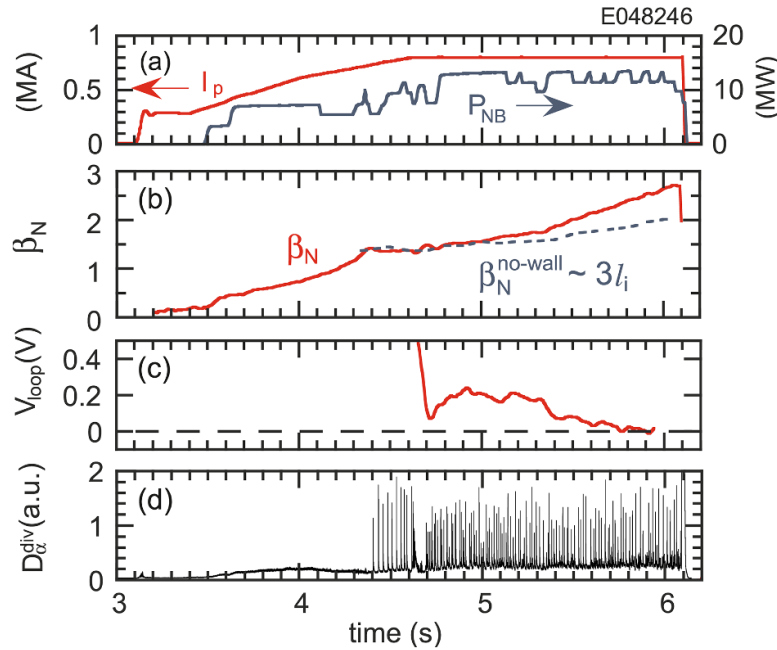


Figure 41. Waveforms of high f_{BS} discharge. (a) Plasma current and NB heating power. (b) Normalized beta and measure of the ideal MHD limit without conducting wall ($\sim 3.0l_i$). (c) Surface loop voltage. (d) D_{α} intensity at the divertor region. Reproduced from [429]. © 2009 IAEA, Vienna. All rights reserved.

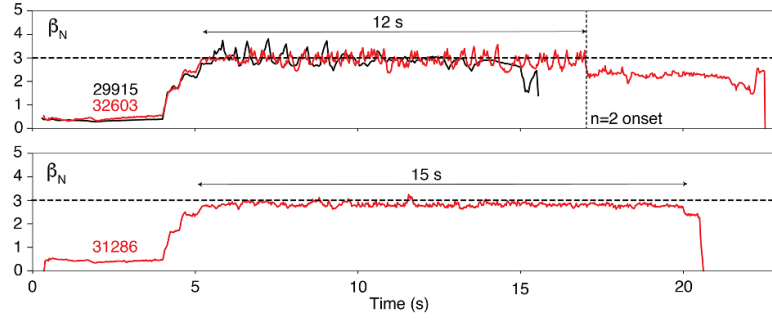


Figure 42. High $\beta_N \sim 3$ operation using the high- l_i scenario [436].

KSTAR. KSTAR aims to contribute to the next step of fusion research by achieving successful operation of the high performance steady-state tokamak plasmas by employing advanced scenarios that utilize fully superconducting magnets [432]. Various candidate scenarios have been developed such as high- β_p , high- l_i , hybrid, ITB, FIRE, and I-mode.

Significant progress has been made in the development of early plasma shape control during start-up that can be applied in various scenarios. This allowed for more practical control of the L–H transition and the effective use of external H&CD, in conjunction with instrumented plasma control and shaping parameters. Extending the operating window, especially pulse length longer than the wall saturation time (a few tens of seconds), is also a major challenge in superconducting machines, and KSTAR demonstrated an H-mode plasma discharge for ~ 102 s.

High- β_p scenarios have been developed for this long-pulse operations focusing upon achieving high f_{NI} supported from a high bootstrap current fraction at low plasma current of 400–500 kA with $q_{95} > 6.5$ [433–435]. The plasma confinement was significantly improved by using ECH as a key control knob. In particular, ECH has been shown to play a critical role in achieving high- β_p scenarios with zero loop voltage operation, as confirmed through experiments where ECH power was alternately turned on and off [434, 435].

KSTAR experiments have demonstrated the potential of steady-state high β_N operation [229, 433, 436]. A discharge with high $l_i \sim 1$ and $\beta_N \sim 3$ was sustained for 12 s at $q_{95} = 4$, using a high- l_i approach as shown in figure 42. This approach involves a low q_{min} scenario with a peaked current profile, which improves confinement at strong magnetic shear and maintains high β_N without wall stabilization. The optimization of the l_i value should consider the balance between high l_i and pedestal, which is advantageous for core-edge optimization. This was achieved by introducing a positive dR_{sep} (radial distance between the upper divertor separatrix and the lower divertor separatrix, as determined at the outboard midplane) to simultaneously reduce the edge pedestal and increase l_i . The high β_N operation has been extended for about 15 s at $\beta_N = 2.8$ without any indication of performance degradation and β_N oscillation resulted from long ELM-free period and large ELM crash also coupled with MHD.

The formation of an ITB has been studied, and an early injection of a sufficiently large NB (> 4 MW) under an

inboard-limited L-mode has been found to be a key factor in accessing it, in KSTAR [437]. The ITB operation window was also extended by upshifting the plasma shaping under a marginal NBI power of up to 3.0 MW [438]. The shaping may reduce the ITB power threshold and enables vertically-off-axis beam deposition. The experiment allows KSTAR to access the ITB in the range 2.9–5.0 MW of NBI heating and demonstrates that this performance can be maintained for about 10 s, which is equivalent to $100\tau_E$.

A new ITB scenario has been established in KSTAR, wherein the central ion temperature exceeds 10 keV and can be sustained for up to ~ 50 s [439, 440]. This confinement regime was achieved with NB heating of low-density plasmas mainly in the diverted configuration [441]. This scenario offers several advantages for steady-state operation, such as having no severe instabilities including ELMs so to reduce the maximum heat load on divertor, no notable core impurity accumulations, and exhibiting high performance comparable to H-mode or even hybrid mode with low loop voltage (< 0.1 V) as shown in figure 43). Based on these characteristics, the scenario has a long-pulse capability without requiring a delicate feedback control. The scenario is named Fast Ion Regulated Enhanced (FIRE) mode since the large population of fast ions were turned out to play a critical role in the core confinement [439, 442, 443]. The high performance of the FIRE mode is typically achieved at $l_i > 1$ and some FIRE mode discharges exhibit I-mode edges which contributes further increase of the performance [440, 444].

The upcoming KSTAR campaign involves implementation of a modified tungsten divertor, making it a priority to evaluate its impact on recent high performance scenarios. Additionally, obtaining long-term solutions for materials and measurement drifts is critical for longer pulse discharges of the high performance scenarios. Once these changes are established, the focus will shift to improving core and pedestal control and accessing higher fusion gain confinement regimes using a β_N feedback control with increasing H&CD systems [309].

6.3. Predictions of ITER steady-state operation

An extensive, albeit not exhaustive, review of steady-state scenario simulations aiming for $Q = 5$ in ITER is provided in a recent paper by Kim *et al* [445], summarized in table 8. There are two types of simulations included in this table; steady-state

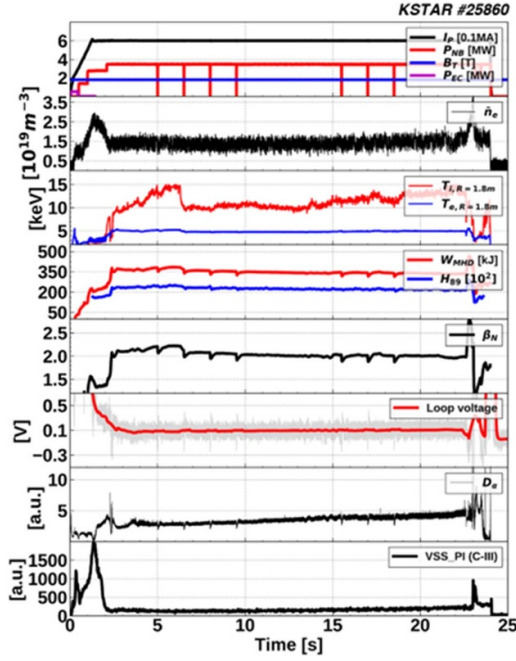


Figure 43. A FIRE mode discharge in KSTAR, with high ion temperature operations $T_i > 10$ keV sustained for ~ 20 s.

calculations that provide an operational point compatible with a steady-state solution, e.g. [15, 446, 447], and time-dependent calculations that also study the access to, and sustainment of, such an operational point considering necessary physics and engineering constraints, e.g. [17, 445, 448–450]. Note that in some simulations the transport properties are predicted using theory-based models, e.g. [15, 17, 449], while in other simulations, the transport properties are instead specified according to certain criteria, for example the requirement to achieve the levels of performance necessary for $Q = 5$, e.g. [445–448, 450, 451].

Among the predictive simulations, those that achieve the target $Q = 5$ performance, do so either by relaxing the constraint of reaching fully non-inductive conditions [17], or by relying on upgrades of the ITER auxiliary H&CD systems [15, 450], or by relying on a very high pedestal boundary condition (higher than predicted by theory-based models) [15]. These results highlighted the challenge of reaching steady-state $Q = 5$ in ITER using day-1 H&CD tools, and using transport and pedestal predicted by theory-based models.

Simulations where the transport properties are assumed show that $H_{98y2} > 1.5$ is required [446, 450]. Many experiments have shown $H_{98y2} > 1.5$, but the excellent confinement quality in these plasmas may be due to physics that does not extrapolate to ITER. This would be consistent with the challenge found for ITER by theory-based models. For example, several DIII-D studies exploiting the capability to vary the total NBI torque at constant total NBI power [296, 313, 452, 453] have shown that very high $H_{98y2} > 1.6$ in DIII-D often relies on high NBI torque, and the related high plasma rotation, and $E \times B$ shear stabilization of turbulence. The level of rotation predicted for ITER is relatively small compared with

the rotation obtained in current NBI driven plasma, although it might still provide some turbulence suppression [454].

One additional recent ITER simulation is part of the work by McClenaghan *et al* [455], that used theory-based models for the core and the pedestal to project to ITER the DIII-D high f_{BS} scenario (high β_p scenario). This simulation is noteworthy because it is the only one to date that predicts a steady-state $Q = 5$ scenario using day-1 H&CD powers, and using transport and pedestal characteristics predicted by theory-based models (TGLF [322] for turbulent particle and energy fluxes, NEO [456] for neoclassical fluxes, and EPED [183] for the pedestal structure). The predicted scenario has $Q = 5.5$ with $I_p = 8.25$ MA, $f_{NI} = 1.0$, $f_{GW,ped} = 0.97$, $f_{GW,line} = 1.3$, $\beta_N = 3.0$, $P_{Aux} = 73$ MW, and a fast ion fraction contribution to pressure of 5% (key predicted profiles are shown in figure 44).

A significant difference between McClenaghan's simulation and those of table 8, may be the higher line averaged Greenwald fraction (~ 1.3), given that the plasma density is strongly leveraging toward higher fusion performance. In fact, at high pressure, the D–T reaction rate increases with temperature more slowly than it increases with density, once high enough temperature is reached (square of the particle density, assuming fixed D/T ratio). Therefore, a further increase in core pressure is beneficial (for higher Q) if it is obtained with a core density increase. As aforementioned, the Greenwald limit has been shown to be an edge density limit, not a core density limit [28, 387], and core densities significantly exceeding the Greenwald value have been observed in different tokamaks [414, 457, 458]. McClenaghan's scenario exploits the creation of an ITB in the density profile to achieve high core density with a pedestal density just below the Greenwald limit.

Note that McClenaghan's work does not address the time-dependent issues of accessing a high q_{min} scenario and triggering the ITB formation. This class of issues is studied in detail in the papers by Poli *et al* [20, 459]. The first work was using TSC [240] to evolve equilibrium and transport, iterated with the TRANSP code [460] for the profiles of heating deposition and current. In the second work, the transport and H&CD were solved more consistently with a fixed-boundary equilibrium solver in TRANSP. Here the thermal diffusivities are computed with the CDBM model [461] and the pedestal structure is set by the EPED1 model [462]. The time-dependent transport simulations examine the plasma evolution from start-up, through the I_p ramp-up, to sustainment in the flattop, and indicate that the formation of ITBs can be triggered in L-mode during the I_p ramp-up phase, and sustained in H-mode during the flattop phase using the baseline heating configuration in ITER [459].

An upgraded configuration including LHCD is predicted to sustain more expanded ITBs and plasmas with higher global confinement levels compared to the baseline heating mix, at $f_{GW,line} \leq 1$ [20].

Another theory-based ITER simulation is discussed in the paper by Petty *et al* [273], and aimed at extrapolating to ITER the fully non-inductive, 'steady-state hybrid' scenario. In present-day experiments, the steady-state hybrid and the high- I_i scenarios are capable of reaching very high normalized

Table 8. Candidate ITER steady-state scenarios reviewed in [445]. Reproduced from [445]. © 2021, ITER Organization. All rights reserved.

Code	P_{NB} [MW]	P_{EC} [MW]	P_{IC} [MW]	P_{LH}	H_{98}	I_p	Q	f_{GW}	f_{NI}	Core transport	Reference
TOPICS	33	40	17	0	1.86	9	3.5	< 1.0	0.98	CDBM05	[15]
TSC/TRANSP	16.5	0	20	25	1.63	8	5.1	< 1.0	0.98	GLF23	[15]
ONEWTWO	33	20	20	0	1.42	9	5.7	1.18	1.01	GLF23	[15]
CRONOS	5.9	0	16.3	20	1.60	8.4	6.9	< 1.0	0.72	GLF23	[15]
CRONOS	0	20	20	13	1.70	8	6.5	0.90	0.90	Specified diffusivity	[448]
Case A	33	20	20	0	1.37	9	4.4	0.97	0.82	GLF23	[449]
Case B	0	53	20	0	1.37	9	4.45	0.97	0.72	GLF23	[449]
Case C	0	73	0	0	1.37	9	4.3	0.97	0.71	GLF23	[449]
Case D	33	40	0	0	1.37	9	4.38	0.97	0.83	GLF23	[449]
FASTRAN	33	20	20	0	1.49	8	3.31	0.85	1.06	GLF23	[17]
TOPICS	33	20	20	0	1.48	8	3.32	0.85	1.03	GLF23	[17]
TRANSP	33	20	20	0	1.43	8	3.31	0.85	0.99	GLF23	[17]
CRONOS	33	20	20	0	1.30	8	3.8	0.85	1.06	GLF23	[17]
ASTRAi	33	20	20	0	1.36	8	3.34	0.85	1.04	GLF23	[17]
ASTRAk	33	20	20	0	1.50	8	3.11	0.85	1.05	GLF23	[17]
FASTRAN	33	20	20	0	1.49	8	3.36	0.85	1.02	GLF23	[17]
FASTRAN	33	20	10	0	1.50	8	3.38	0.85	1.01	GLF23	[17]
FASTRAN	33	20	5	0	1.53	8	4.25	0.85	1.01	GLF23	[17]
TRANSP	33	20	20	0	1.57	7	2.4	1.00	1.00	Specified diffusivity	[450]
TRANSP	33	20	20	0	1.64	7.4	3.6	0.99	1.00	Specified diffusivity	[450]
TRANSP	33	40	20	0	1.52	9	3	0.84	0.98	Specified diffusivity	[450]
TRANSP	33	40	20	0	1.61	9	3.8	0.96	0.95	Specified diffusivity	[450]
TRANSP	33	20	0	20	1.63	8.85	4.3	0.95	1.00	Specified diffusivity	[450]
TRANSP	33	20	0	20	1.64	8.85	5	1.05	1.00	Specified diffusivity	[450]
CORSICA	16.5	20	0	20	1.23	9	3.06	0.80	0.67	Specified diffusivity	[451]
CORSICA	16.5	20	0	20	1.34	9	3.64	0.80	0.72	Specified diffusivity	[451]
CORSICA	16.5	20	0	20	1.46	9	4.32	0.80	0.79	Specified diffusivity	[451]
CORSICA	16.5	20	0	20	1.60	9	5.14	0.80	0.87	Specified diffusivity	[451]
CORSICA	24.75	20	0	20	1.59	9	4.59	0.80	1.00	Specified diffusivity	[451]
CORSICA	33	40	0	0	1.61	9	4.24	0.80	0.99	Specified diffusivity	[451]
ASTRA	33	22	0	0	1.66	9	5	0.90	1.00	Specified diffusivity	[446]

fusion performance, mainly enabled by very good energy confinement quality. However, as discussed above, an excellent confinement quality ($H_{98y2} > 1.5$) in plasmas with monotonic q -profile and $q_{min} \sim 1$ has been connected to the turbulence stabilization effects of large $E \times B$ shear and/or large fast ion fraction. Theory-based modeling of ITER does not find these effects to play a significant role, although it is also possible that these effects have not been included accurately so far. The ITER simulation by Petty *et al* [273] used FASTRAN [463] with self-consistent calculation of transport, equilibrium, and H&CD sources. The simulation predicts that 50 MW of NBCD power and 56 MW of ECCD are needed to drive all of the plasma current, that is a significant upgrade to ITER's day-1 H&CD powers. A fusion gain $Q = 4.6$ is predicted, and the steady-state plasma profiles from the FASTRAN modeling are shown in figure 45. Note that a pedestal pressure raised by 25% over the EPED1 [462] prediction is used in the modeling, mimicking the ratio between the experimental pedestal height for the RMP ELM-suppressed, steady-state hybrid discharges on DIII-D ($\beta_{N,ped} = 1.0$) and the corresponding EPED1 prediction ($\beta_{N,ped} = 0.8$). A fusion gain $Q \sim 3$ is predicted using the actual ITER pedestal height predicted by EPED1.

6.4. Outstanding issues

As stated in the IRP [10], the demonstration of fully non-inductive $Q = 5$ for 3000 s is the ultimate goal of the ITER project. Such demonstration requires a highly integrated solution encompassing many research areas of tokamak plasma physics. To date, many of the candidate solutions are still unripe. The challenges include but are not limited to:

- High global confinement quality: from the simulation results reviewed in the previous section, the predicted $Q \geq 5$ cases all require $H_{98y2} \geq 1.6$ at $0.8 \leq f_{GW} \leq 1.05$, except if higher f_{GW} can be achieved, e.g. $f_{GW} \sim 1.2$ in which case the requirement in H_{98y2} can be relaxed to 1.4. It is critical to recognize the gap between the achieved candidate operational points in present tokamaks and the requirement for ITER steady-state $Q = 5$ scenario. As mentioned in sections 4 and 5, deeper questions also lie in whether the high confinement quality achieved in present experiments can be extrapolated to ITER conditions, e.g. low toroidal rotation, strong electron heating, $T_i \sim T_e$, low CF, etc. There are urgent needs for experimental explorations in present

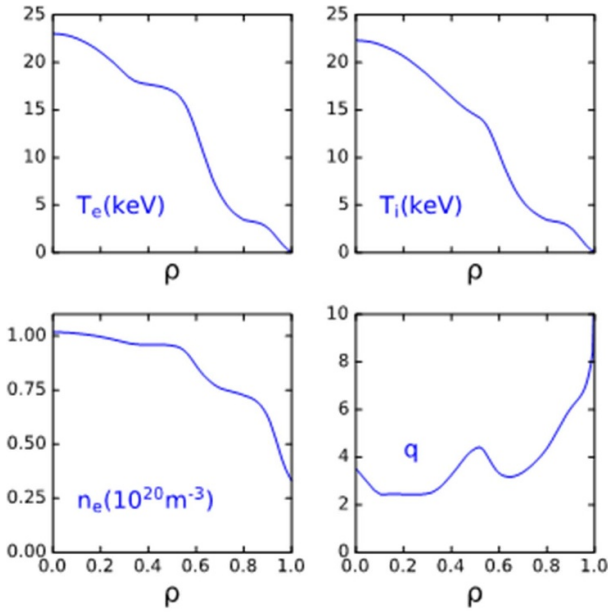


Figure 44. Radial profiles of electron temperature, ion temperature, electron density, and safety factor corresponding to steady-state $Q \sim 5$ in ITER, predicted using the STEP module in OMFIT. Reproduced from [455]. © 2020 IAEA, Vienna. All rights reserved.

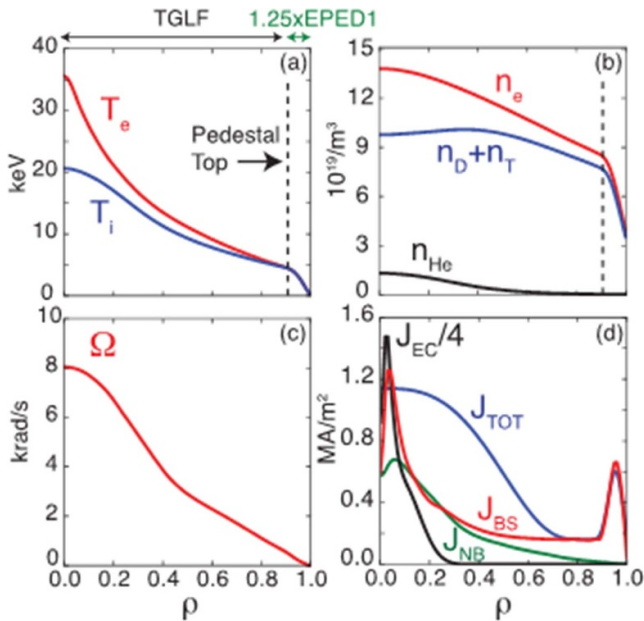


Figure 45. Self-consistent profiles for ITER steady-state $Q \sim 5$ case from FASTER simulation; (a) electron and ion temperatures, (b) electron, fuel ion and helium ash densities, (c) toroidal angular frequency, and (d) ECCD, NBCD, bootstrap, and total current densities. Reproduced from [273]. © 2017 IAEA, Vienna. All rights reserved.

tokamaks, not only pushing operational boundary towards simultaneous higher density and higher confinement quality, but also trying to understand the governing physics and the compatibility with reactor conditions. The high β_p scenario

could be a promising example that has demonstrated operation with simultaneous $1.3 \leq H_{98y2} \leq 1.8$ and $1.0 \leq f_{GW} \leq 1.25$ [464].

- High non-inductive current fraction: ideally, this fraction should be 100% in the candidate ITER steady-state $Q = 5$ scenarios. From table 8, most $Q \geq 5$ cases require additional LH power for off-axis CD, highlights the challenge of achieving $Q = 5$ steady-state using day-1 H&CD systems. Further optimizations or even a substantial change in the operational target in the modeling should be considered. In present experiments, long-pulse demonstrations all go along with reduced performance [398], below the requirements for ITER steady-state $Q = 5$. It is important to note that present experiments tend to have significantly higher electron collisionality compared to ITER's values. When projecting the experimental scenario to ITER conditions, one should expect higher bootstrap current fraction (at fixed β_N and q_{95}) due to the lower collisionality in ITER. On the other hand, if a present experiment achieves fully non-inductive operation, it is very likely that the H_{98y2} would be higher in its ITER counterpart with similar normalized parameters, because less CD power would be needed in ITER (to be confirmed by predictive simulations).
- MHD activities associated with high β : the ITER steady-state $Q = 5$ scenario is expected to have higher β_N than the $Q = 10$ inductive solution at higher plasma current. The question is how high and how close this β_N will be to the no-wall and ideal-wall MHD β_N limits. Review of the previous modeling results suggests $\beta_N \geq 2.9$ for successful $Q \geq 5$ predictions. Such a value is found to exceed the $n = 1$ no-wall β_N limit in many cases. Both passive stabilization and active control of the potential RWMs are important physics issues from plasma operation in this condition. Demonstrations of stable operation with similar dimensionless parameters and profile shapes in present experiments are highly needed. Another potential challenge is the NTM. This can be especially important for the scenario that has low q rational surfaces, e.g. $q = 1.5, 2$. Robust and efficient active controls need to be developed in relevant candidate scenarios: ideally, both direct control of the unstable mode, and control of the equilibrium profiles to reduce the instability drive.
- Fast particle confinement and control of associated MHD instabilities: experiments show that, at low density, the high β_N , high q_{min} , steady-state scenario is vulnerable to degradation of fast ion confinement, due to AE activity. The fast ion losses to the wall not only reduce the global energy confinement quality, but also increase the risk of first-wall damage, and lead to additional source of impurity influx from the wall, further reducing the performance of the core plasma. Although confinement enhancement by interaction between MHD activities and fast ions is observed in some experiments as reviewed in [465, 466], optimizations of the scenario parameters or control approaches are needed to minimize the impact on the fusion performance.
- Core-edge integration: edge solutions that mitigate the transient heat load to the main wall and divertor from ELM instabilities, and divertor solutions that mitigate the steady

heat and particle load on the divertor plates are extremely important to machine survivability in the 3000 s, steady-state $Q = 5$ operation of ITER. Both the leading ELM and the leading divertor solutions for ITER (RMP ELM suppression and divertor detachment) tend to degrade the edge pedestal, and reduce the global plasma confinement quality. In addition, international experimental database analyses show high-Z impurity influx leads to reduced confinement quality with metal wall in comparison with data from carbon wall [192]. Thus, core-edge integration is a serious challenge for $Q = 5$ steady-state operation requiring excellent energy confinement quality, $H_{98y2} \geq 1.4$. The high f_{BS} scenario approach has demonstrated the compatibility of a very high confinement core ($H_{98y2} \sim 1.5$, $\beta_N \sim 3.0$) with fully detached divertor and ELM mitigation, albeit with carbon wall. Progress is expected in the near future since more tokamaks are converting to a metal wall.

Beside the high-level challenges discussed above, there are different outstanding issues in association with specific candidate scenarios in present experiments for the ITER steady-state $Q = 5$ goal:

- High f_{Aux} scenario with elevated q_{min} : due to the inverse proportionality between auxiliary CD efficiency and plasma density, this approach favors lower density to achieve higher f_{Aux} . Using off-axis CD broadens the current density profile and results in an elevated q_{min} . Although many experiments following this approach have achieved high non-inductive current fraction and high plasma performance, the use of high power co-current direction NBI leads to high external torque in a low density plasma and results in high toroidal rotation compared to the rotation expected in ITER. Even at lower performance, the role of toroidal rotation in the achieved high confinement quality has not been investigated thoroughly. The features of low density and high q_{min} also yield some challenges in this type of scenario, namely substantial AE activities that lead to fast ion anomalous transport, and lower global energy confinement [467]. This issue may be further amplified in a $Q = 5$ environment, where fast particles (fusion helium) become the major heating source. Demonstrations of core-edge integrated solutions in this type of scenario is lacking.
- High f_{Aux} scenario with low q_{min} : the steady-state hybrid scenario, high- l_i scenario, and the new FIRE mode are the examples of this scenario with q_{min} close to unity with flat or monotonic q -profiles. Steady-state hybrid scenario has been achieved by extending the standard hybrid scenario towards lower plasma current. ELM suppression has been successfully integrated with high normalized fusion performance. However, high toroidal rotation is found to be important to maintain the high confinement quality observed experimentally, and theory-based extrapolation to ITER suggests an upgraded H&CD power system is needed to reach $Q \sim 3$. The steady-state hybrid scenario also needs to be demonstrated in long-pulse operation. High- l_i scenario has transiently achieved high f_{BS} ($\sim 80\%$) at very high $\beta_N \sim 5$ and $H_{98y2} \sim 1.8$, with $l_i \sim 1.4$ and has been sustained at

$\beta_N \sim 3$ with $l_i \sim 1.0$. Optimization for steady-state operation leads to $l_i \sim 1.0$ and more moderate performance, quite similar to the steady-state hybrid scenario. Core-edge optimization incorporating ELM mitigation also needs to be explored. FIRE mode features H-mode confinement quality with a non-ELMing L-mode or I-mode edge, and long duration with very high core T_i and no impurity accumulation. The good energy confinement quality is explained via multiple toroidal-rotation-independent turbulence stabilization mechanisms, mainly due to fast ion stabilization. Key issues to address are the requirements for low density and for input power below the L–H transition threshold. The latter one could be addressed by combining the FIRE mode with a negative triangularity shape, if available, to robustly avoid H-mode at high power. Regarding the former, it may be challenging to achieve in ITER a sufficiently high fraction of fast ions to enable a FIRE-like mode. However, in the core region of ITER's burning plasma, the presence of alpha particles, in addition to NB-driven fast ions, may contribute to thermal ion dilution, and help mitigate turbulence. Investigations to further understand these effects are ongoing.

- High f_{BS} scenario: this approach favors high β_p , since $f_{BS} \sim \epsilon^{0.5} \beta_p$, where ϵ is the inversed aspect ratio, and high density, since density gradients are more efficient at driving bootstrap current than temperature gradients. Research has shown various reactor-relevant features in this scenario, including toroidal rotation independent turbulence suppression mechanism, excellent confinement quality at density above the Greenwald value, high β_N fully non-inductive operation, high confinement core compatible with fully detached divertor and small ELMs. However, key outstanding issues remain, including: access path to the large-radius ITB state, and stable operation above the no-wall β_N limit. Formation of ITB at large radius is the key to the success of this scenario. However, neither the ITB access nor the operation above the no-wall β_N limit have been thoroughly investigated in ITER-like conditions including dominant electron heating, low toroidal rotation, and low collisionality. In addition, it needs to be demonstrated in long-pulse operation. These demonstrations will also be important for assessing the controllability of integrated performance in a highly self-organized configuration. At high bootstrap fraction and low recirculating power, control of the current profile may rely primarily on control of the pressure or stored energy. In ITER, this will have to be done with limited diagnostic capabilities.

6.5. Conclusions

To achieve its goal of stationary fully non-inductive operation at $Q = 5$, ITER will most likely require a plasma energy confinement enhancement $H_{98y2} > 1.5$, simultaneous with line averaged density Greenwald fraction $f_{GW} > 1$. Up to now, only few tokamaks have demonstrated access to this area of parameter space [464]. Some upgrades to the ITER FPO phase H&CD tools are predicted to reduce the physics challenges, by enabling plasmas with higher global confinement levels compared to the baseline heating mix.

Ultimately, present-day tokamak experiments should try to assess the feasibility, access, and controllability options for steady-state scenarios with $H_{98y2} > 1.5$, $\beta_N \geq 2.8$, and $\beta_t \geq 2.2$, operating with simultaneous conditions of low input torque, high Greenwald fraction, detached divertor, small or no ELMs, and a metal wall. Then, theory-based integrated modeling can be used to understand the projections of these candidate scenarios to the ITER conditions.

Beyond what has been done and summarized in the previous subsections, the modeling needs to address the key challenge of multi-species transport analysis. Since very high core confinement quality is required for ITER's steady-state $Q = 5$ goal, will the outward transport of the helium ash be sufficiently large, in the absence of core instabilities such as sawteeth expected in the baseline inductive scenario? Conversely, will the inward transport of tritium and deuterium be sufficiently large to maintain the fueling of the DT reaction in the deep core of ITER? Pellet penetration is expected to be insufficient to reach the deep core ($\rho \leq 0.6$). Thus, fueling the deep core will rely on turbulent and neoclassical inward transport. High-fidelity, self-consistent multi-species transport modeling is only beginning to be feasible [468]. The growing efforts towards establishing whole-device modeling frameworks should be able to provide significant contributions in developing reliable ITER steady-state $Q = 5$ scenario predictions in the near future.

7. Actuators for integrated scenario operation

Coordinated by R.V. Budny

7.1. Introduction

Actuators play crucial roles for developing plasmas to conditions appropriate for successful tokamak fusion reactors. These conditions include sustained durations of large rates of fusion energy production in burning DT plasmas. This subsection reviews research for actuators for ITER and other future tokamak fusion reactors. The topics covered are the needs and uses of actuators, development and application of computer codes for modeling and predicting actuator effects, benchmarking, and experimental results for verification and validation of these codes, and quantifying actuator effects. It does not address engineering or other technical details of actuators.

It focuses on actuators listed in a PCS matrix [469] showing the overlap of actuators for controlling plasma parameters for optimal tokamak operation. These were grouped as actuators dedicated primarily for density, such as PLCS, temperature control with HSs, with CD, and with coils for plasma shaping. Each of these actuators affect multiple categories. The HS actuators can affect not only the thermal energy, but also the plasma density, current, and momentum. This subsection also include discussions of some research on topics not included in the ITPA JAs, but which have important bearing on the development of actuators. Examples are discussions of results from DT experiments, a topic especially relevant for ITER and beyond.

Alpha heating is essential for standard approaches to fusion energy production. This heating results from the slowing down of confined alpha particles emitted by DT fusion reactions. Experiments in TFTR and JET have quantified or put bounds for the fusion power P_{α} . The ultimate goal is to use this power as the dominant heating source. This could become more of a control challenge for reactors. Here we focus on auxiliary HSs which can be directly controlled.

The H-mode plasma, assumed to be the IBL (see section 5) is predicted [157, 192] to be the most likely mode for achieving the desirable conditions. Large amounts of heating power are required to achieve this mode. Scalings for threshold powers have been derived from an extensive database of experiments [157]. Predictions for ITER gives a threshold power of ~ 52 MW in a deuterium plasma at a line averaged density $n_e = 0.5 \times 10^{20} \text{ m}^{-3}$. At the nominal ITER H-mode density, $n_e = 1.0 \times 10^{20} \text{ m}^{-3}$, the threshold power estimated is ~ 86 MW.

New operating modes in DT have been developed. These include a hybrid mode with lower plasma current and higher pressure [311] discussed in section 5, and steady-state modes discussed in section 6. Also successful control of tungsten from the wall was achieved, as well as ELM control. These results were achieved with synergistic enhancements of the NB and ICR actuators, which is expected to play a smaller role in ITER than in current machines.

Our consideration mainly covers issues addressed by the ITPA JA and JEX carried out by the ITPA IOS and the Steady-State Operation (predecessor of the IOS TG) [470, 471] TGs published after the latest IPB [11]. It includes an overview of progress of relevant studies carried out since 2007, and issues needing further studies.

Besides control of density and temperatures, both PLCS and HS affect the energy confinement and rotation, affecting the regime of operation. Global confinement has been studied in ohmic [472], L-mode [473], H-mode [33], Long-pulse inductive scenarios (Advanced Tokamak and hybrid operation [219, 311, 474]).

The energy confinement depends directly on plasma density and on heating power, P_{heat} which depends, for instance, on the power to the SOL, $P_{\text{SOL}} = P_{\text{heat}} - P_{\text{rad}} - \dot{W}_{\text{th}}$. The radiation, P_{rad} , in turn depends on the impurities, such as ions resulting from seeding by PLCS and from wall material (W) which can diffuse to the plasma core.

The heating techniques considered for reactor projects are termed as H&CD systems, since, besides control of temperature through heating and plasma heat diffusivity χ , they typically drive non-inductive current, j_{CD} , which affects also plasma stability, transport, and length of discharge.

The H&CD systems considered for future reactors include LH, EC, IC, and NBI. The IC and NBI H&CD contribute to plasma rotation through polarization and direct NBI, as well as generation of suprathermal ions, thus affecting plasma transport, including stabilization of turbulence [466] and instabilities such as NTMs [475, 476], sawtooth [477] (see also [478] and chapters 4 and 7 of this issue). The NBI contributes also to plasma fueling, although very little is expected in ITER due to the high energy, and thus low density of the beam ions.

Mutual impacts of actuators within a group dedicated to the same primary control are called ‘synergistic effects’. Examples of synergy are LH and ECR H&CD [479], NBI and ICR H&CD [171, 179, 480]. The synergy of HS could make control more complicated, although [171] predicts that synergy effects between NBI and ICR are small.

Another issue for H&CD systems is absorption of the injected power by the plasma. For the ICR and LH H&CD, coupling of the injected power to the plasma depends on plasma parameters inside and outside the last closed flux surface [481], thus depending on the PLCS. The fraction of power absorbed by the plasma in a single path depends on plasma parameters in the plasma core. Unabsorbed power affecting the PFC needs mitigation to reduce the PFC degradation, and thus, requires coordination of the HS and PLCS [149, 171, 482].

The status of heating and other actuators is discussed in [483]. The issues related to H&CD are discussed in more details in section 7.2. The issues related to PLCS are discussed in section 7.3.

Computer models of actuator actions play key roles advancing fusion research. One such role is projecting plasma regimes to future experiments. A large portion of this chapter discusses potential plasma regimes resulting from these projections.

Another class are control models discussed in the scenario control section 8. There the emphasis is on predicting the control of performance by simulation. The ITER PCS design, requires the development of a large set of control-oriented models. These automatically survey the operating space and find optimal settings, for example to achieve the highest possible performance within the physics and engineering constraints.

An important foundation for modeling is predicting plasma parameters such as plasma profiles. Majority advances in the accuracy of modeling have come from gyrokinetic-based approximations. These are discussed in chapter 2. Since that publication, a major review of codes is given in [484]. The theory, development, and validation of reduced quasi-linear models of gyrokinetic turbulent transport in the closed flux surface core of tokamaks is given.

Physics models are used to understand results from experiments and to extrapolate physics results to the parameters anticipated for future experiments. These uses of models are described in detail in the proceeding subsections and in chapter 4. Actuator models are used for designing actuators and assessing the abilities of actuators in new situations.

An important part of the JAs of SSO and IOS TGs was dedicated to benchmarking codes used for modeling. It is important for assessment of accuracy of both predictive and interpretative simulations.

The subsections are organized as follows:

7.2 Auxiliary heating, torquing, and CD

7.2.1 NB

7.2.1.1 NB models

7.2.1.2 Benchmarking NB models

7.2.1.3 Experiments corroborating NB models

7.2.2 IC

7.2.2.1 IC models

7.2.2.2 Benchmarking IC models

7.2.2.3 Experiments corroborating IC models

7.2.3 EC

7.2.3.1 EC models

7.2.3.2 Benchmarking EC models

7.2.3.3 Experiments corroborating EC models

7.2.4 LH

7.2.4.1 LH models

7.2.4.2 Benchmarking LH models

7.2.4.3 Experiments corroborating LH models

7.2.5 Synergies between heating and CD

7.3 Density

7.3.1 Density modeling

7.3.2 Benchmarking density modeling

7.4 Integrated modeling

7.5 Outstanding issues

7.6 Discussions and conclusions

Many of the papers cited have contributions in more than one area so they are discussed in multiple subsections.

7.2. Auxiliary heating, torquing, and CD

In tokamak plasmas there is an intrinsic bootstrap current I_{BS} driven non-inductively, for instance by NB ions [485], and intrinsic heating due to the fraction, $f_{ion} < 1$, of P_{fus} , carried by fast ions born in the fusion reactions, $f_{ion}P_{fus}$. To reach the desirable plasma performance it is necessary to use auxiliary external tools for plasma heating power, P_{Aux} , and CD. The efficiency of a reactor is expressed as a fusion gain, $Q = P_{fus}/P_{Aux}$, and the efficiency of the external CD is proportional to I_{CD}/P_{Aux} . In this subsection we consider techniques of the H&CD used in present machines and potentially useful in future reactors. Applications include heating, torquing, CD, and shine-through control by impurities and rotation.

Some of the actuators described in these subsections probably will be less important in the burning phase of a fusion reactor with large Q , however, all of them are important in such tokamaks as ITER with a goal of $Q \sim 10$ as well as in future fusion reactors to provide access to the burning phase. Control becomes increasingly difficult with increasing Q , even when Q is still finite. This is a huge challenge. The status of heating and other actuators was discussed in [483].

7.2.1. NBI. NBI will be an important actuator in ITER. Besides enhancing fusion yield by heating, its role in fueling, torquing, and destabilizing Alfvén and other energetic driven eigenmodes is expected to be very different from its role in the present-day experiments, but it probably will not be used for future tokamaks with higher P_{fus} . In those tokamaks, access into the tokamak and beam penetration into large, dense plasmas is problematic. These will depend on alpha heating and use other actuators for fueling, CD, and control.

The system for ITER is large and complex for accelerating negative deuterium ions, neutralizing them, and injecting the fast neutrals into ITER. An overview of the system is given in [149]. The current status of the source development are reported in [170, 486]. Recently, 90% of the ITER target

for the extracted current density was achieved in hydrogen for 600 s.

The NBI system will be similar to the negative deuterium ion systems used in JT-60U and JT-60SA, but will be very different from the positive D and T ion based injection used in current tokamak experiments. In those positive D and/or T ions were accelerated to energies E_{beam} in the 10–140 keV range, neutralized, and injected into the tokamaks.

ITER foresees both H and D NBI, but likely will operate mostly (perhaps only) with H NBI. The beam ions will be accelerated up to E_{beam} in the MeV range. This high energy, needed to penetrate to the core, is considerably larger than in current experiments. Also the core density is expected to be larger. The maximum NBI power planned in ITER is in the range $P_{\text{NBI}} = 33.0\text{--}49.5$ MW, close to the maximum range (30–40 MW) of current experiments. In MeV range of energies, multistep ionization dominates the NBI stopping cross-section, in contrast with charge exchange in present machines [487, 488].

An accurate model of multistep ionization of NB atoms is important for calculating safe maximum beam injection power and minimum plasma density in order to not damage the FW facing the beams. The excited states beam stopping of hydrogenic and helium beams in target ions with $Z = 1$ to 10 is calculated in TRANSP [29] using the density dependence of beam stopping data from ADAS [489].

The beam stopping and shine-through of high energy neutral hydrogen beams in high density plasmas is studied in JT-60U [488]. Legacy atomic cross-section data for stopping in fully stripped impurity ions of He, Li, Be, C, N, O, and Fe are fitted. Approximate agreement of beam shine-through in JT-60U is obtained.

Detailed predictions of shine-through in ITER are presented in [186], where ITER operational limits are presented with a simplified 0.5D plasma transport solver METIS [340], a fixed-boundary magnetic equilibrium solver CHEASE [490], and NEMO [491]. The BBNBI [492] and NEMO beam ionization codes are compared. BBNBI has been recently upgraded to use ADAS cross-sections, though not yet implemented in the ITER IMAS [491] BBNBI version.

A Coordinated Research Project on the topic of fundamental atomic data for NB modeling was conducted by the International Atomic Energy Agency (IAEA) ran from 2017 to 2022; this project brought together ten research groups. A new cross-section for the proton-impact ionization of H was recommended. Results are reported in [493]. This paper reviews and compares theoretical methods for calculating ionization, excitation, and charge exchange cross-sections applied to several important processes relevant to neutral hydrogen beams. Coefficients for an empirical fit function to this cross-section and to that of the first excited states of H are provided and uncertainties estimated.

The maximum CF from NBI, $S_b \sim P_{\text{NBI}}/E_{\text{beam}}$, will be an order of magnitude smaller than in present machines. The total fueling rate in the core D and T by the beam, pellet, and gas puffing is estimated to be of order of replenishment of the fuel burned in the DT reaction, but insufficient to fuel plasma to

the densities required for high-Q operation [494]. These estimates assume of a perfect particle balance, when the wall is in equilibrium with the plasma, releasing and absorbing the same number of particles per second. If this equilibrium is violated (globally), the fuelling and pumping system will need to transport a corresponding amount of particles to or from the torus. A detailed analysis of the wall effect in the particle balance is therefore an important issue.

In reactors that require H-mode operation to reach full fusion performance (such as ITER), it would be necessary to heat plasma ions to increase alpha heating to a level that keeps the plasma above the L-mode to H-mode, P_{LH} threshold. In current experiments this threshold increases with density so low density is desirable if above the roll-over density, but a lower bound for the density is that it cannot allow excessive NBI shine-through to opposite walls. There is an optimal density for minimum L–H threshold and it depends on the plasma effective mass [495]. That paper compares data to power threshold scalings, used for extrapolation to future devices such as ITER and DEMO.

During low density phases of ITER DT operation, the fast ion pressure is expected to be a substantial fraction of total pressure. This could significantly reduce the excitation drive of the Alfvénic modes and turbulent transport [255]. For the range of energies, $E_b \sim 1$ MeV the dominant ion heating is still possible locally in the plasma center if the electron temperature is sufficiently high (in the 25 keV range depending on the energy, NBI and plasma species), but for the range of temperatures predicted for the main volume the NBI heating of the electron component will dominate.

7.2.1.1. Neutral beam modeling. For current and previous generations of tokamaks, such as TFTR and JET, NBI was required for achieving the highest tokamak fusion rates due to beam-thermal reactions and dominant ion heating.

There are three methods for computing NB parameters. Pencil-like beams are the simplest, and least accurate; Fokker–Planck methods are fast to compute, but do not incorporate as many of the physical phenomenon as Monte Carlo methods. Thus, the Fokker–Planck approach is very useful for rapid scans, while Monte Carlo methods are more suitable for intensive studies with more realistic physics.

The TRANSP [29] Monte Carlo code NUBEAM model [496] has played an important role in studying of NBI and alpha parameters in DT experiments. Examples are given in [22, 25, 497–499]. The peak neutron emission rates were much higher than those achieved in these experiments without NBI. The core ion temperatures, ion densities, and rotation rates are much higher with NBI. The beam-thermal reaction rates were higher than the peak thermonuclear fusion rates. In contrast, the beam ion fraction predicted in the core of ITER is two orders of magnitude lower, so many of the beam ion parameters will be much lower [16]. This is expected to be the case for large future fusion reactors.

Integrated predictions using TRANSP and other codes are discussed in section 7.4.

7.2.1.2. Benchmarking neutral beam modeling.

Benchmarking codes is a verification process important for finding bugs and increasing confidence in their validity. Many benchmarking studies of predictive codes are published. Examples of publications focusing on NBI predictions are:

The TRANSP Monte Carlo NUBEAM model was benchmarked with the original TRANSP Fokker–Planck code [496]. Good agreement was found. NUBEAM was used to benchmark the beam calculations of the beamlet-based ASCOT code [492]. Test particle ensembles from TRANSP-NUBEAM results for NBI beam-beam fusion reaction agreed well with the produced by PENCIL for JET. AORSA was used to benchmark NBI ASTRA, DBEAMS, FPP, and NBEAMS. Approximate agreement was found.

Two new 1D Fokker–Planck solvers have recently been implemented within ETS: StixRedist and FoPla [500]. V&V efforts of JET shots are discussed.

Reference [501] presented a bounce averaged Fokker–Planck solver integrated into the integral transport code TOTAL. This was used for analyzing NBI H&CD. The distribution functions of the collisional pitch angle scattering operator are discussed. In steady-state, the results agreed with those obtained by the ACCOME and the OFMC [502] codes.

A rapid Fokker–Planck solver, RISK incorporating the CRONOS [325] transport suite into the JAMS framework was developed [503]. This showed the strength of running simulation codes in a modular and machine independent environment for simulating fusion plasmas. Outputs for JET and ITER steady-state operation were compared with those from SPOT [504].

Benchmarking of NBCD with the Monte Carlo codes was studied using OFMC [471]; using NUBEAM; and with NEMO/SPOT [503]. Agreement is observed in fast ion source profiles calculated with different beam stopping models including atomic excited states corrections. Heating profiles by the Monte-Carlo codes used generally agrees. The partitions to electrons and ions are different between OFMC and NUBEAM, while SPOT and NUBEAM give quite close partitions. The orbit loss as fast ions cross the separatrix causes an erroneous result. OFMC calculates larger fast ion density and fast ion current than NUBEAM, while SPOT gives a smaller one than NUBEAM. Collision may be weaker in OFMC and more pitch angle scattering may occur in SPOT.

Comparing the Fokker–Planck codes, ASTRA calculates larger j_f in $\rho > 0.4$ than ACCOME. This can result from the non-bounce-averaged beam ion velocity and the simplified NBI geometry. The electron shielding models used agree within 2% in $\rho < 0.6$. In the peripheral, however, the models derived for the banana regime deviate from the NUBEAM model using Hirshman's coefficients applicable to the entire collisionality regime, although the total NBCD is almost unaffected due to small fast ion current there. The NBI-ASTRA solver with improved description of the trapped ions demonstrated good agreement with the NUBEAM predictions [447].

Benchmarking of NBCD with Fokker–Planck codes was studied [505] with ONETWO coupled with FASTWAVE and TORAY, and the Fokker–Planck codes ACCOME and ASTRA

codes. The focus is off-axis NBCD. Differences in the fast ion source arising from different beam models are small for the ITER.

7.2.1.3. Experiments using dominantly neutral beams for heating and/or CD. Validation of NBCD in AUG, DIII-D, JT-60U, and MAST are studied [506]. Plasmas without MHD (e.g. sawteeth, fishbones, NTMs, AEs except ELMs) were chosen since MHD can cause redistribution of fast ions, thus modifying the NBCD. Current profile broadened by off-axis NBCD was evaluated using MSE diagnostics and good agreement between the measured and calculated NBCD profile was observed. Comparisons of measured and calculated CD profiles suggested the requirement of anomalous transport of fast ions by turbulence in modeling to agree with measurements.

Off-axis NBCD experiments on AUG and JT-60U was studied in [507]. It shows significant changes of the current profile by either on-axis counter or off-axis co-CD. The isotopic mass scaling of heat transport in AUG L-modes is studied [508] with the GENE code [509]. The isotopic dependence of core pressure in H and D plasmas with NBI and ECH is analyzed using TGLF [510]. Results strongly suggests that the observed isotope dependence in L-mode is not dominated by a gyroradius effect, but a consequence of the mass dependence in the collisional energy exchange between electrons and ions.

The transport of fast beam and alpha ions in most routine DT plasmas appears to be consistent with classical according to TRANSP analysis [25, 330, 331, 511]. Indirect evidence for classical beam ions is the approximate agreement of simulated and measured neutron emission rates. Exceptions are reported to require anomalous transport for agreement. Anomalous beam-ion loss in TFTR reversed magnetic shear plasmas was reported [512].

Isotopic mass scaling and transport were compared in JET deuterium and tritium L-mode plasmas with matched dimensionless parameters in [513]. Higher scaled energy confinement time is found in the tritium plasma. Negligible difference in the particle transport was seen. A significant fraction of the isotope scaling and reduced heat transport observed in the Tritium plasma is not captured in the GENE and ASTRA-TGLF-SAT2 simulations.

In experiments in JET with NB heating with trace tritium injection, evidence of anomalous diffusion in ELMy H-mode plasmas at high density was reported [514]. The argument for fast ion anomalous transport in JET is questioned in [25].

7.2.2. Ion cyclotron range of frequency. ICRF waves are used to heat either the main ion species at the harmonics or their cyclotron frequency or a minority species at its main cyclotron frequency. Minority ions are introduced on purpose in typical concentrations of a few percent. ICRF waves may also be used for CD via a combination of electron Landau damping and transit-time magnetic flux pumping. ECCD induced magnetic flux pumping is discussed in section 7.5.

The ITER ICRF system is a high power continuous wave RF system operating at 40–55 MHz. It will be used for H&CD

as well as for ion cyclotron wall conditioning (ICWC). In the current baseline (2016 Staged Approach), the ITER ICRF system features two identical antennas [150], and [515], coupling each 10 MW to the plasma. The coupled power is upgradable to 20 MW per antenna. The ITER project is however currently undergoing a rebaselining process [1] in which one equatorial port is dedicated to the ITER ICRF system for coupling 10 MW to the plasma in the first operation phase of ITER. Depending on the ICRF results in this phase, the ICRF system could be upgraded to 20 MW while keeping one antenna.

The ITER ICRF system, owing to its flexibility, the design of its antennas, the local gas injection, etc can be operated with excitation limits which allow high power operation (up to 10–20 MW/antenna), while keeping low ICRF-specific sputtering compatible with operation with a high-Z (tungsten) wall.

7.2.2.1. Ion cyclotron range of frequency modeling.

Modeling of ICRF H&CD is subtle due to complications such as the strong coupling between the antenna and the scrape-off plasma region, coupling to impurities and fast ions, and the variety of wave modes which can be excited [515].

The performance of the ITER ICRF system using TOPICA and ANTITER II modeling is reported in [481]. This paper also discusses optimization of ITER antenna design. More recent ICRF coupling results are obtained on TOPICA, COMSOL, ANTITER II and Petra-M and reported in [517–520].

Results for both the Fast and Slow modes in the SOL using the ANTITER IV code [481] are shown in [520]. The new modeling capabilities suggest mode conversion and parasitic slow mode absorption near the peripheral LH resonance can be tailored by steepening the density profiles locally near the resonance. This can be achieved by reducing the low- $k_{\parallel}$ content of the radiated spectrum. This can be performed by setting the right correlation of excitation with amplitude and phase for the phased array antenna.

ICRH scenarios for DEMO are investigated in [521]. It proposes second harmonic tritium ion heating scheme in the core, assisted by fundamental ICRH of ^3He in the early phase of the discharge. Three ion resonance heating and NB-ICRH synergy are discussed.

ICRF processes anticipated in the SOL of ITER:

- Many of these processes are hard to diagnose, and the present measurement techniques are hardly envisaged for ITER. Some processes are poorly characterized experimentally in present devices, e.g. ‘far field’ sheaths or coaxial modes. Most of the edge effects are studied in D plasmas while in its nuclear phase ITER will work with a DT mix as majority species.
- Power coupling [481, 517, 518–519], and the reduction of the near parallel electric fields are addressed in [515]. Excitation of coaxial modes and power losses at LH resonance are estimated in [520].
- Isotopic effects relevant for the ICRF-specific SOL processes are being analyzed in 2023–2024 JET experiments [522]. The magnitude of these processes depends on far SOL

parameters. The main hurdle in the quantitative extrapolation via modeling is a large uncertainty on what the far SOL profiles will be in ITER. ICRF simulation tools are further evolving towards more integrated SOL+core modeling incorporating RF sheaths.

- The change of the FW to tungsten requires a reassessment of the impact for the PWI, especially near the ICRF antenna. Of special concern is ICRF-specific erosion of W from the antenna port sides [523]. This reference uses the multi-physics finite element solver COMSOL and SSWICH-SW which computes RF wave propagation from the antenna mouth to the SOL. It gives an assessment of ICRF-specific PWI in the new ITER baseline.
- Several of the points above are now demonstrated by modeling for the ITER ICRF antennas using the well-developed tools which have been applied on the existing devices and validated with experimental data. For example, the boosting for the coupling using local gas injection in ITER is demonstrated in [518], the reduction of the near fields using proper excitation of the antenna is demonstrated in [515], etc.
- Maximization of ICRF power by SOL density tailoring with local gas injection to enhance ICRF power coupling is discussed in [524]. ICRF coupling improves with local injection, in comparison with similar gas amounts puffed from the divertor region. The effect is local and vanishes as the injection point moves away toroidally from the radiating straps. Too large gas rates degrade the core plasma confinement, and may also impact performance in the ELM regime. It was however demonstrated experimentally that boosting the coupling can be achieved at rates which do not degrade the plasma confinement [524].
- ICRF heating schemes for the ITER non-active phase are tested in [525] and studied in [526]. These studies show that it is possible to use ICRF heating schemes for 1/3 toroidal field with either H-majority or He-minority in H plasmas, but not for 1/2 field in H plasmas.
- Three-ion ICRF mode conversion was observed in TFTR [527] and [528]. This is discussed in section 7.2.2.3. The physics and applications are discussed in [529] and [530], where the physical principles behind three-ion ICRF scenarios and the large variety of experimental observations are reviewed. A careful mix of two majority species transforms mode conversion electron heating into efficient heating of a third (minority) ion species.
- Beyond on-axis and off-axis heating, the scenario discussed above provides a localized source of high-energy ions for CD and to explore alpha particle physics in current experiments. These ICRF scenarios hold promises for ITER to extend the H-mode operational space in its low activation phase and to provide bulk ion heating of D–T plasmas using intrinsic impurities as minority species. For the non-active phase, ITER could use ^4He – ^3He –H 3-ion scheme, as has been successfully tested experimentally recently on AUG and JET, although the resonance is far from the axis.
- Recent studies show that ITER ICRF antenna with their four toroidal columns of straps can be fed such as to reduce the near fields [515]. Some antenna excitation laws even allow reducing the near fields without compromising the coupled

power (this can in particular be achieved while using local midplane gas injection) [531].

- Modeling of ITER ICRF antennas with the TOPICA code is reported in [517]. The performance of the antenna has been documented in terms of input parameters, power coupled to plasma and electric fields.
- ICRF coupling to ITER antenna with non-axisymmetric cases are studied.
- Three D full wave modeling of realistic HHFW antenna have been performed [532]. The Petra-M code, which is a newly developed state-of-the-art generic electromagnetic simulation tool for modeling RF wave propagation based on MFEM: (a free, lightweight, scalable C++ library for finite element methods) [533].

7.2.2.2. Ion cyclotron range of frequency benchmarking.

ICRF modeling results for ICRF codes in ITER plasmas are benchmarked [534]. The plasma predictions were generated using TRANSP and GLF23 [535] for an H-mode baseline and three half-field, half-current H-modes which are of interest for the low activation phase. Results from the full wave-solvers AORSA [536], CYRANO [537], EVE [538], PSTELION [539, 540], TASK/WM [541], and TORIC [30] are compared for H&CD. Approximate agreement was achieved by four of the solvers for the heating power partitions for the DT and ^4He cases. Factor of two or more disagreements were found for the heating power partitions for the cases with second harmonic ^3He heating in bulk H cases. The PSTELION results had considerably lower electron heating and higher minority ion heating. The CYRANO results had higher electron heating and lower minority ion heating. Three of these codes achieved approximate agreement simulating the ICCD 1D profiles.

Benchmarking was performed between the antenna codes TOPICA, RAPLICASOL and Petra-M for the ICRH ITER antenna [519]. An excellent agreement in S-, Z-matrices, amplitude of the wave field in front of the antenna and coupled power for different antenna feeding schemes was found.

ICRF coupling modeling results for ITER ICRF have been benchmarked using tools such as TOPICA, COMSOL, Petra-M [542]. Petra-M (Physics equation Translator for MFEM).

7.2.2.3. Ion cyclotron range of frequency experiments.

Three component ICRF mode conversion can be an efficient means of achieving energetic ions. This was first observed in DT experiments in TFTR [527]. The resonating ions were D, T, and ^7Li (from pellet injection). The absorption was calculated using a variant of the 1D full-wave code FELICE [543]. The single-pass absorption of ^7Li was computed to rise rapidly with the tritium fraction. Strong ion and electron heating was observed. This scheme is discussed further in [544].

This scheme has been investigated in C-Mod and JET [545] and reviewed in [529] using the 1D full-wave code TOMCAT [546]. The three-ion ICRF scenarios with ^3He resonant with deuterium and hydrogen studied resonant heating of ^3He .

Results from the recent JET DTE2 campaign are summarized in [547]. The ($n = 2$) D heating scheme is being reassessed for DT operation in ITER at half magnetic field [548]. The main uncertainty in the case of $n = 1$ or $n = 2$ D ICRF heating is the spurious heating of the alphas (while the fusion yield remained modest in JET).

Second harmonic ($n = 2$) T, possibly combined with fundamental ($n = 1$) ^3He or H minority is considered as a reference heating scheme for ITER DT baseline. Experiments for this scenario were tested in TFTR DT [528], JET DTE1 [549], and JET DTE2 [550]. The JET DTE2 modeling indicates that even low intrinsic concentrations of ^3He lead to significant changes in ICRF power partitioning during second harmonic heating of T due to absorption of up to 30% of the wave power by ^3He .

H minority is outside the frequency range for full field operation in ITER so this is not a reference scheme for DT, unless done at half-field. In DTE2 50:50 D:T plasmas, up to 35% and 5% larger fusion power and diamagnetic energy content, respectively, were obtained with second harmonic heating of T as compared to H minority heating at comparable total input powers and gas injection rates.

Fundamental ($n = 1$) D heating with D-NBI is demonstrated to be the most fusion-power efficient in JET DTE2 [551]. Experiments showed that in tritium rich DT plasmas both the thermalized D ions and the fast D-NBI ions are accelerated to energy ranges which are optimal for the DT reaction cross-section, boosting the fusion yield. Simulations of such scenario for ITER have been done with the integrated modeling and analysis suite IMAS H&CD workflow discussed in subsection 7.4.

A multi-machine review of RF sheath effects is given in [552]. Non-linear interactions of RF in the sheath are dominantly rectification. RF-enhanced wall sputtering might increase core plasma contamination with high-Z impurities.

Heat loads onto the wall are locally increased during ICRF and were documented from surface temperature measurements. Besides other SOL transport processes (turbulence, parallel losses), the RF-induced $E \times B$ convection redistributes the local density in the antenna vicinity. In general, the scenarios are devised so that the resonant wave/particle interactions occur near the plasma core. The accelerated resonant species re-distributes the power gained from the waves to the other plasma constituents via collisions.

Methods for boosting the ICRF coupling are now very well developed. Experiments under the coordination of the ITPA with AUG, JET, and DIII-D are reviewed in [524]. Large increases in the antenna loading were observed when injecting deuterium in ELMy H-mode plasmas using midplane inlets close to the powered antenna instead of divertor injection in DIII-D, JET, and AUG [524, 553–555]. This will probably be employed on ITER. The methods for reducing the RF-specific impurity generation are also very well developed (see results on AUG [556, 557]).

Minimization of the ICRF/SOL interaction by proper RF feeding of the 4×6 strap array on ITER is studied in [558], based on successful proof-of-principle experiments on AUG,

C-Mod, and JET. Multiple observations confirmed the reduction of the impact of ICRF on the SOL and on total impurity production when the ratio of the power of the central straps to the total antenna power is in the range $0.6 < P_{\text{cen}}/P_{\text{total}} < 0.8$. Near-field calculations indicate that this fairly robust technique can be applied to the ITER ICRF antenna, enabling the mode of operation with reduced PWI. On the contrary, for the A2 antenna in JET-ILW the technique is hindered by RF sheaths excited at the antenna septum. Thus, in order to reduce the effect of ICRF power on PWI in a future fusion reactor, the antenna design has to be optimized along with design of PFCs.

Robust guidelines (i.e. resilient to profile changes) are needed to mitigate undesired SOL interaction in ITER. An example is to measure SOL density profiles down to typical peripheral R-cutoff densities. Avoid excitation of low- $k_{\parallel}$ components in the radiated spectrum ($|k_{\parallel}| < \omega/c$) favor ICRF heating scenarios with large single-pass absorption; tailor the edge density profiles near the antenna using gas injection to reduce the strap cut-off distance; steepen the profiles near the LH resonant layer to reduce the LH absorption; and tailor the strap excitation to minimize image currents induced on the ICRF antenna box.

7.2.3. Electron cyclotron heating and CD. A system for ECH and ECCD is planned for ITER. H&CD applications include plasma initiation and start-up, control of the equilibrium current profile, and control of MHD instabilities such as sawteeth and NTMs. The ability to accurately predict the location and profile of ECH and ECCD is critical for ITER.

Active research is being carried out for many years on EC-assisted breakdown and current ramp-up. For instance, in KSTAR and DIII-D. This is discussed in the section 2 on plasma initiation.

ECCD induced magnetic flux pumping is discussed in section 7.5.

7.2.3.1. Electron cyclotron heating and CD modeling. The single-pass absorption is predicted to be low in some scenarios with the ECH. The operational constraints and the H&CD capabilities for 1/3 field (1.8 T, 5 MA) operation in the ITER PFPO phase are studied in [188]. These scenarios require low density in hydrogen and helium plasmas. To achieve single-pass absorption of 99% from the equatorial launchers relatively high central electron temperature ~ 8 –10 keV is required.

For half-field (2.65 T, 5 MA) operation proper absorption requires change of the EC wave polarization from ‘O’ to ‘X’ in the current flattop phase, which is accompanied by rather skinned power deposition profiles due to parasitic third harmonic absorption near the edge [171, 482]. Both codes predict that absorption can be made more peaked by increase of the magnetic field to 2.75–2.9 T.

The $J \times B$ force produced by ECH in HSX Plasmas has been estimated [559]. This paper evaluates the toroidal torque generated by radial return current which cancels the current by

radial flux of energetic electrons. This paper presents results for helical plasmas. This code is already applied to tokamak plasmas.

7.2.3.2. Electron cyclotron heating and CD benchmarking. Codes for ECH and ECCD in ITER conditions are benchmarked in [470]. It compares predictions for several representative cases using TORAY-GA [560], GENRAY [561], CQL3D [562], and BANDIT-3D, and ray tracing with TORBEAM, OGRAY, and GRAY [562]. The agreement between codes for off-axis application is excellent.

7.2.3.3. Electron cyclotron heating and CD experiments. ECH and ECCD are useful for controlling NTMs by precisely depositing localized current at the magnetic island effectively stabilizing it and preventing disruptions; essentially, ECCD ‘patches’ the missing bootstrap current within the island, mitigating its destabilizing effect. Success relies on rapidly and accurately determining the presence of NTM growth, accurately directing the H&CD, and the absence of turbulence diffusion of the injected waves.

Detection of NTM growth is an important first step for amelioration. Algorithms have been developed which use the NTM amplitude to control ECCD deposition and stabilize NTMs. They have been tested in AUG [562–564].

Recent AUG experiments assessed third harmonic ECRH, in view of 1/3 field operation in ITER PFPO1 phase [187, 255]. In order to avoid stray radiation on the inner wall, it was proposed to tailor the magnetic field and keep the second harmonic layer off-axis at the HFS of the discharge.

Quasi-optical modeling of wave beams with and without mode conversion has been studied in [565]. Numerical studies of waves in dissipative media simulate propagation and absorption of electromagnetic waves including EC waves in plasma. This paper presents a simple model of field configuration. This code has been applied to tokamak equilibria.

ECCD feedback control of locked modes was studied in DIII-D [566]. The phase of the perturbation was optimized in real-time, in feedback with magnetic measurements, in order for the mode’s phase to closely match a prescribed phase, as a function of time. Synergies between heating and CD actuators are discussed in section 7.2.4.

7.2.4. LHCD. LHCD [567] is not currently being considered for ITER; however, it has the very attractive properties for CD by being steerable, and might be used in later experiments. The launchers can be swiveled or use a phased array to vary the location of the driven current. This can be used to tailor the total current profile and thus adjust the safety factor profile, and add to the bootstrap current.

7.2.4.1. LHCD modeling. One issue for the usefulness of LHCD in fusion reactors is that the waves can be absorbed by the high-energy alpha particles. LHCD-alpha interactions

are studied in [568]. Upper bounds on the density of supra-thermal ions are found. The higher frequency, 5 GHz waves had negligible absorption. Effects of the LHCD-plasma current interactions are studied in [569].

Predictions for LH absorption by fusion alpha particles are discussed in [570]. The models which were used combined 2D velocity space Fokker–Planck solvers with toroidal ray tracing packages. These models also predicted the highest LHCD efficiencies with 2.0–2.6 MA of driven current for 30 MW of coupled LH power.

7.2.4.2. LHCD modeling benchmarking. Benchmarking LHCD codes and predictions for LH absorption by fusion alpha particles are discussed in [570]. Predictions of CQL3D-GENRAY, DELPHINE, ACCOME, SPOT, DKE, and LSC are discussed and compared for an ITER-like steady-state scenario plasma. Differences in the magnitude of the driven current are predicted, with the more complete 2D (velocity space) Fokker–Planck-ray tracing models predicting the highest and the response function codes predicting lower.

7.2.4.3. LHCD experiments. A passive-active-multi-junction LHCD launcher in Tore Supra long-pulses is validated [571]. It assessed LHCD at high density for extrapolation to ITER-like advanced scenarios. A very good agreement between the experimental results and the coupling code prediction is obtained.

Experiment results from C-Mod, EAST, FTU, JET, and Tore Supra are summarized in [572]. These experiments occur in electron temperature regimes where the LH wave is not fully absorbed in a single-pass, thus challenging analysis. This paper argues that reducing wall recycling of neutrals increases the edge temperature and the effects of parametric instability and collisional absorption are reduced, thereby increasing the CD efficiency.

A full-wave model for the LH wave electric field vector with synthetic turbulence on C-Mod is discussed in [573]. Models for filaments, holes, and periodic fluctuations are used to assess their effects on LH wave electric fields. The most efficient LHCD experiments are achieved at low densities. This paper shows that in addition to parametric decay instabilities and collisional losses, LH wave scattering by SOL fluctuations is also a serious candidate to explain the loss of CD efficiency observed at high density (large Greenwald fraction), at least in C-Mod. The changes in LH wave field magnitude and direction are also measured by dynamic Stark effect spectroscopy.

Modeling of advanced tokamak scenarios with LHCD in C-Mod is presented in [574]. Conditions of stable operation near ideal stability were identified. They have high bootstrap current fraction. Excellent wave accessibility and absorption were still found in the presence of an H-mode-like edge density barrier. However, the presence of these barriers resulted in about a 10% decrease in the stability limit.

Results from JET, Tore Supra, and FTU are reviewed in [575]. JET achieved long ITBs with current profile sustained and controlled by LH acting under real-time feedback together with first LH control.

7.2.5. Synergies between heating and other actuators.

EAST has several electron heating sources. Simulations of LH power at two frequencies were studied [576] using TRANSP, GENRAY, and CQL3D. The expected synergy is when the absorption of one RF wave increases the fast electron population, which allows a second, simultaneously injected RF wave to drive current at a higher efficiency. This theoretically predicted synergy has not yet been observed experimentally on EAST.

Simulations of NBI-ICRH synergy in EAST scenarios using TRANSP have been discussed [576]. Fast NBI-D ion are accelerated far past the NBI energy of 55 keV, up to 350 keV. This could intensify fusion in burning plasmas.

LH waves can provide the highest CD efficiency (defined as the ratio of the driven current to the injected wave power), although with limited control capability. Conversely, EC waves drive highly localized currents, and are therefore particularly suited for control purposes, but their CD efficiency is much lower than that of LH waves (typically, by an order of magnitude in present-day experiments, but with the order of magnitude at reactor-relevant temperature) [577].

The reason for this difference in CD is related to the different interaction mechanisms of the two waves with the electrons; parallel velocity diffusion associated with Landau damping for LH waves and perpendicular velocity diffusion associated with cyclotron damping for EC waves (parallel and perpendicular directions are defined with respect to the equilibrium magnetic field). LH waves can efficiently drive electrons from low to substantially higher parallel velocities, making them poorly collisional and insensitive to trapping, thus carrying a larger current than the slower electrons interacting with the EC waves. For these reasons, the idea of combining the two CD systems has been proposed and investigated since the early 1980s.

Fast ions (protons and ^3He) originating from NBI and ICRF systems can increase neutron production during the PFPO phase of ITER in presence of Be impurity. The ASTRA-NBI [578] and TORIC-SSFPQL codes were used to study them in [179]. This shows non-linearities with minority fraction.

The CD efficiency and the synergistic effect for ECCD and LHCD and the possibility of long-pulse operation in ITER are discussed in [479]. It discusses extending the pulse length by the increase in the CD efficiency due to the synergistic effect for combined ECCD and LHCD, which was predicted to be small for the LH and EC H&CD systems designed for ITER.

Advances in the steady-state hybrid regime in DIII-D to a fully non-inductive, ELM-suppressed scenario for ITER are discussed in [273] (see section 5). It suggests high confinement in the ITER similar shape have achieved zero surface loop voltage for up to two current relaxation times using efficient central CD from ECCD and NBCD. Anomalous widening of absorption/CD for central ECCD is discussed. During ECCD, AE activity is replaced by a more benign fishbone-like mode, reducing anomalous beam ion diffusion by a factor of two.

Synergies between EC and LH waves in Tore Supra are studied in [258, 577]. Improvement up to a factor of four of the ECCD efficiency in plasmas sustained by LHCD has been

demonstrated in stationary conditions on the Tore Supra tokamak. This was made possible by feedback control of zero loop voltage, constant plasma current, and constant density. This effect, predicted by kinetic theory, results from a favorable interplay of the velocity space diffusions induced by the two waves; the EC wave pulling low-energy electrons out of the Maxwellian bulk, and the LH wave driving them to high parallel velocities.

There are also effects of the H&CD on the plasma configurations, such as 3D stable configurations and magnetic flux pumping, discussed in subsection 7.5, which changes the absorption of the heating power and shape of the driven current profile and distribution of fast particles and pressure.

7.3. Density actuators

Density control is crucial for plasma performance. In this subsection we discuss the PLCS relevant actuators, as well as limitations of PLCS control connected with physical limitations and interplay with HS and other control systems.

The fusion rate depends on the deuterium and tritium ratio in the core. This can be controlled by the gas fueling rate and pellet injection. The ITER particle pumping and fuelling system has been studied [494]. Penetration of particles from the edge sources in the reactor size plasmas is well screened by the opaque SOL. Thus, the density control requires integrated approach because of the impact of other actuators, discussed below.

The deuterium and tritium fueling needs to be separately controlled. Possible isotopic mass effects with increasing tritium could be significant, either beneficially or detrimentally. There have been extensive studies of the isotopic mass effects of density and temperature profiles and of enhanced energy and confinement times following observations in TFTR of enhancement with tritium. A recent reanalysis of the TFTR results indicates that the fast NB tritium ions had enhanced energy questioning the assumption that the thermal DT plasma had enhanced confinement [579].

Examples of theoretical studies indicating mechanisms that might improve confinement above gyroBohm, are [580], that attributed enhanced confinement in TFTR to radial electric field shearing; [581, 582], to non-adiabatic electrons altering the transition from the ion-dominated core to the electron-dominated edge; [583], effects of trace impurities in residual zonal flows [584], predicting DD and burning DT plasmas to compare isotopic effects. This paper describes the interplay and competition between zonal flows shearing, electromagnetic effects, and the torque applied, which can reverse the sign of gyroBohm scaling.

A system for creating and controlling spin polarizations of the deuterium and tritium fuel could be a new type of actuator for fusion power plants [585–588]. The cross-section for $D + T \rightarrow \alpha + n$, would be increased by 1.5 if deuterium and tritium were injected with parallel polarization. The polarization dependencies of fusion reactions are summarized in [589]. That reference studies simulations of the impacts of polarized fuel from alpha heating in a large-scale tokamak, such

as ITER. It outlines a detailed study of *in situ* polarization survival test experiment using pellets.

These references demonstrate that using spin-polarized DT fuel with more deuterium than tritium can increase tritium burn efficiency by at least an order of magnitude without compromising fusion power output, compared to unpolarized fuel. This could reduce the requirement of off site tritium.

The polarization of deuterium and tritium, whether along or perpendicular to the toroidal direction, results in different distributions of neutron emission directions. Control of the direction of the neutron emission could significantly reduce the neutron damage and shielding requirements as well as being used to optimize the tritium breeding blanket design.

Small adjustments to the polarization scheme (and therefore the DT cross-section and alpha+neutron emission directions) might also provide plasma and power control over a range of time scales. In the event of ignition thermal runaways, rapid depolarization of the enhanced cross-section polarization scheme could be achieved by launching RF waves into the plasma with a frequency equal to the deuterium and tritium nuclear magnetic moment precession frequency. By depolarizing the fuel, the DT fusion cross-section would instantly decrease, thereby reducing the alpha self-heating through fewer emitted alpha particles. Such a scheme is analogous to a fire extinguisher for burning plasmas, which is challenging for unpolarized fuel.

Another effect of spin-polarized fuel currently being investigated is the effect of anisotropic alpha emission on the alpha heating profiles, the plasma current, and fast-particle instabilities. By changing the relative weighting of trapped and passing alpha particles by changing the DT polarization, there could be significant implications for plasma stability, the fusion power density profiles, and the fusion gain.

Various methods for polarizing deuterium, tritium, and helium-3 for spin-polarized fusion are discussed in [589]. Methods developed for polarizing ions in nuclear and particle physics experiments can produce nearly full polarization. Another potential method could be pellet injection of polarized lithium-deuteride.

Recycled deuterium and tritium can become depolarized in the FWs. Various mechanisms of depolarization and choices of the wall that could prevent depolarization are discussed in [590]. If the fusion reactor is sufficiently large that the polarized fuel in the core has a small probability of reaching the wall that depolarization might not be a problem. A research program for studying the lifetime of spin polarized fuel has been proposed [591].

The lifetime of spin polarized fuel has been studied in DIII-D with D polarized pellets [592]. Experiments to study unpolarized D-NBI with polarized D fuel reactions are evaluated in DIII-D.

The PFCs will be a source and sink of fuel, ash, and other impurities. The choice of wall materials can strongly affect plasma performance, even changing confinement scalings that are relied upon for extrapolation to future devices. Effects of changing the JET wall have been studied [317]. The replacement of the carbon wall by a Be/W ILW affected the plasma

energy confinement. It was found that the power degradation of thermal energy confinement with the ILW is weaker than predicted by the H-mode confinement scaling $\tau_{E,98y2}$ [192] and resulting in an increase in normalized confinement from $H_{98y2} \sim 0.9$ at $\beta_N \sim 1.5$ to $H_{98y2} \sim 1.2$ – 1.3 at $\beta_N \sim 2.5$ – 3.0 as the power was increased. This paper discusses whether this is a reduced power degradation or a positive beta effect. It is also shown that at higher gas puff, the power degradation is similar to H_{98y2}/H_{20IL} .

Progress with power and particle control was summarized in [593]. Significant progress has been made in the areas of energy and particle transport in the SOL.

There are two types of actuators dedicated for CF and CMIS, the GIS and PIS. Helium isotopes, ^4He and ^3He can be delivered to plasma through gas puffing, and possibly by ^3He -D pellet injection. Fusion alpha ash will accumulate and needs to be considered. Hydrogen isotopes, H, D, T can be delivered to the plasma both through gas puffing and as solid pellets. Hydrogen can be delivered also by diagnostic NB and by heating H^0 -NBI.

The deuterium and tritium fuel can be delivered by D^0 -NBI and T^0 -NBI. T^0 -NBI is not planned for ITER, provided these tools are used for diagnostics and heating. It should be noted that the ^4He , ^3He , H, T isotopes are produced and D, ^3He , T consumed as the result of fusion reactions in presence of deuterium. The core density control in many cases affects the MP.

The location of mass deposition is deeper for bigger pellets with higher velocities and depends on the local plasma parameters [594]. Therefore pellet parameters should be adjustable to provide desirable deposition in evolving plasmas.

The primary goal of the CF and CMIS is the multipurpose control of the core density, n including control of the plasma species, n_H , n_D , n_T , $n_{4\text{He}}$, $n_{3\text{He}}$. A large part of the flux of the species to the plasma volume can come from outgassing of the walls. In reality, every change in the plasma-wall contact during the discharge can disturb this balance, requiring an adjustment of the fuelling system performance. In comparison with current devices, the particle throughput in ITER is expected to be much lower with respect to the recycling flux in a fusion reactor where the ratio of pumped to recycling flux will be ~ 0.01 compared with ~ 0.1 in smaller machines [494, 595]. Similar pumping predictions for DEMO are given in [594, 596].

Pumping out the particle exhaust remains an important actuator of the PLCS. During DT operation the CF is used to control the DT fuelling mixture including the fuelling replenishment with T. Therefore, together with plasma heating the CF provides the control of the rate of fusion reactions (for DT operation also referred as burn control) [597–600].

The CF (implicitly) and CMIS (explicitly) together provide the IC power coupling to the plasma and IC heating using the minority heating scheme. Thus, together with the IC power the CFS and CMIS control the IC heating [481].

The net particle throughput in ITER discharges includes the particle outflux from the confined plasma, due to the

transport processes, G_{SOL} , the particle outflux associated with the ELMs, G_{ELM} , the particle flux required for ELM control by pellet pacing, G_{LFS} , the gas fuelling, G_{aux} , used for control of divertor detachment and power loads (for example, gas puffing, G_{div}), and the gas injection for ICRF coupling, G_{IC} , if necessary:

$$G_{\text{DT}} = G_{\text{SOL}} + G_{\text{ELM}} + G_{\text{LFS}} + G_{\text{aux}} + G_{\text{IC}} \quad (18)$$

Quantifying this quantity via modeling would be important.

Major aspects related to the modeling of density control including PLCS actuators are addressed in the IOS-JA11 ‘Particle transport benchmark’ with results summarized in [23] (see section 4.1.2). The JA of the pellet source code benchmarking and validation on the basis of the experimental data is ongoing.

To avoid duplication in the present subsection we simply list the aspects of the integrated simulations related to PLCS actuators described in detail in [23] leaving the PLCS relevant issues, which require further analysis to section 7.3. The IOS-JA11 addressed both PIS and GIS code benchmarking.

For PIS it comprises comparison of pellet fuelling for LFS and HFS pellet injection. It includes the sensitivity studies on pellet size and velocity. For GIS the sensitivity studies were carried out for L-mode and H-mode operation in a cylindrical slab with a 2D neutral gas description. The benchmark has shown that the depth of fuelling for reactor size plasmas predicted by different solvers differs by a factor of two.

Therefore, for integrated PLCS it is necessary to derive a parametric operational space, where the control is possible, together with the dependencies of the efficiency of the actuators on plasma state and the impact of other controllers (like RMPs, pellet fueling, and pellet pacing). The boundaries of the operational space can be determined by MP, design features of diagnostics and actuators (range and dynamics of variation), and other controlled parameters. Let us consider the operational space for density control as an example.

Note that there are phases of the discharge, such as plasma breakdown, burn-through, transition from the limiter to divertor configuration, and ELMs, where the events are too fast to be actively controlled by the PCS during these events. Before such phases (events), when plasma parameters are still controllable, they should be set to the level which provides secure operation within the limits of the operational space of the events. For example, before plasma breakdown and burn-through the pre-fill gas pressure should be taken above the level of the electron runaway limit and below the maximal density for burn-through. Parameters of ELM pacing or/and RMPs should be chosen to provide MP and to avoid the loss of H-mode operation [603]. For the cases where real-time control is possible, there are the boundaries for density control operation space listed below.

For MP, the plasma density should be kept above the run-away limit:

$$n > n_{\text{ra}}(E_{\parallel}, T_e, Z_{\text{eff}}). \quad (19)$$

The lower boundary for possible density control by fuelling is the separatrix density, which, depending on the scenario can be determined by other control needs, such as heating, divertor detachment control, ICRF coupling control, ELM control, plasma current, safety factor, and mode of operation:

$$n > n_s(P_{\text{SOL}}, P_{\alpha}, G_{\text{SOL,diffusive}}, G_{\text{SOL,ELM/RMP}}, G_{\text{puff}}, G_{\text{LFS,pellet-pacing}}, S_{\text{eng}}, q, I_p, n_{\text{imp}}, L/H - \text{mode}). \quad (20)$$

Here the power to the SOL depends on the auxiliary heating and core radiation, $P_{\text{SOL}} = P_{\text{Aux}} - P_{\text{rad}}$. Engineering pumping speed is limited by S_{eng} , particle fluxes from the core (diffusive and ballistic) are used for ELM pacing and puffed for ICH coupling, fueling $G_{\text{SOL,diffusive}}$, $G_{\text{SOL,ELM/RMP}}$, $G_{\text{LFS,pellet-pacing}}$, G_{puff} , are limited by design of the ITER pumping system:

$$G_{\text{SOL,diffusive}} + G_{\text{SOL,ELM/RMP}} + G_{\text{puff}} + G_{\text{LFS,pellet-pacing}} \leq G_{\text{pump}} \quad (21)$$

where G_{pump} is the design pumping limit.

Pedestal density should be less than the Greenwald density (which can be controlled by HFS pellet fuelling as shown in [604]):

$$n_s < n_{\text{ped}} < n_{\text{GW}} (= I_p / \pi a^2 [10^{20} \text{ m}^{-3}; \text{MA}, \text{m}]), \quad (22)$$

$$n > n_{\text{ped}}. \quad (23)$$

Density should exceed the NBI shine through limit when NBI is used (supposed to be controlled by PIS and GIS):

$$n > n_{\text{shine,NBI}}(A_i, A_{\text{NBI}}, E_{\text{NBI}}, Z_{\text{eff}}, T_e). \quad (24)$$

Density control should keep the operational mode (supposed to be controlled by PIS and GIS); for H-mode operation (H-mode threshold density),

$$n < n_{\text{H,max}}(P_{\text{SOL}}, A_i) \quad (25)$$

for L-mode and ohmic operation (for optimal L–H transition) (supposed to be controlled by PIS and GIS):

$$n \geq n_{\text{min}} \quad (26)$$

with n_{min} determined by avoidance of the lock modes [605], and minimization of power, required for an L–H transition [606].

Potentially, there is another limit for pedestal density, connected with the possibility of type-I ELM suppression, to be studied in further experiments [607] (see section 7.3):

$$n_{\text{ped}} < n_{\text{ELM}} \quad (27)$$

Therefore, all control options should leave the operational space for density control, keeping

$$\max(n_{\text{min}}) < \min(n_{\text{max}}) \quad (28)$$

as well as to derive the efficiency of different actuators in different conditions of plasma operation.

In JET, a lower bound for the minority species concentration was imposed when using high power ICRF. This was done to ensure good single-pass absorption and reduce fast particle losses. The latter may not be so important in ITER because of the large dimension and larger plasma current.

Integrated simulations of H-mode operation in ITER including CF, divertor detachment and ELM control are studied using EPED1 [183] and SOLPS scaling [603]. These show the operational space where the ELM mitigation might not be needed for ITER H-modes.

The ongoing activities and outstanding issues, which require the further studies are discussed in subsection 7.5.

7.3.1. Density modeling. To avoid duplication in the present subsection we simply list the aspects of the integrated simulations related to PLCS actuators described in detail in [23] leaving the PLCS relevant issues, which require further analysis to subsection 7.5. The IOS-JA11 addressed both PIS and GIS code benchmarking.

Efficiency of different actuators also depends on the plasma state and other actuators (see below figure 16 from [23]). In particular, for reactor scale plasmas, GIS is efficient only at ohmic and L-mode operation. With increase of plasma heating sufficient to H-mode operation the separatrix density increases to the level $n_{s,H}(P_{\text{SOL}}) > n_{s,HL}$, where the increase of puffing cannot increase plasma density [184, 601, 602]. The density control at H-mode requires application of PIS to increase the density to desirable level, but the core density control is limited by $n > n_{s,H}(P_{\text{SOL}})$.

7.3.2. Density model benchmarking. Major aspects related to the modeling of density control including PLCS actuators are addressed in the IOS-JA11 ‘Particle transport benchmark’ with results summarized in [23]. The codes benchmarked therein are ASTRA, CRONOS, ETS, FASTRAN, JINTRAC, RAPTOR, TASK/TR, TOPICS, and TRANSP/PTSOLVER. Simulations are performed with identical sources, sinks, transport coefficients, and boundary conditions prescribed in the flattop H-mode phase. Comparisons of the transport solvers and particle flow and density profiles are shown. Origins of the discrepancies are discussed in section 4.1.2.

The JINTRAC suite of codes has been used to benchmark EDGE2D and SOLPS 4.3, and found agreement [189]. EDGE2D was modified to model helium species and compared with SOLPS 4.3. Predictions of plasma parameters in ITER L-mode and H-mode are shown. Approximate agreement is found. Particle transport in the SOL and divertor are studied in detail in chapter 5 in this review.

The ongoing JA13 of the pellet source modeling is benchmarking and validating on the basis of the experimental data.

7.4. Integrated modeling of ITER performance with multiple actuators

This subsection gives an overview of broadly integrated modeling of burning plasma performance using multiple actuators, including aspects discussed in section 7.2.

TRANSP was the first integrated (meaning comprehensive) code used to predict ITER performance. TRANSP predictions gave the most accurate predictions of DT fusion rates in TFTR before the DT campaign [331, 511]. This indicates the predictive abilities of TRANSP.

For ITER predictions, electron density profiles near the Greenwald limit were assumed. Waveforms for the electron density profiles are ramped up to achieve the L–H transition starting NB injecting into relatively low density plasmas.

TRANSP using the GLF23 transport code [206] was used to compute ITER temperature profiles in IBL plasmas [16, 22, 497, 534, 609, 610]. TRANSP with the GLF23 and MMM95 [611] codes were also used to study IBL plasmas [612]. These show optimistic results for the fusion gain goals for the IBL H-mode plasmas, assuming high density, low impurity, benign MHD, etc. They also predict conditions where the fusion alpha power becomes dominant and the auxiliary heating power can be reduced.

In steady-state conditions, the ash (thermalized alpha particles) will accumulate to a constant profile depending on the fusion power, ash transport, and recycling rates. Predictions for this are given in [16] and [610]. Similar modeling ignoring ash accumulation, such as those in [614] predict Q continuing to increase as P_{Aux}^2 .

The TGLF transport model [243] was improved and verified against a database of 400 collisionless non-linear gyrokinetic simulations from the GYRO code [510] and validating against DIII-D, JET H-mode and hybrid discharges, and TFTR L-mode discharges. TGLF was used in a fast transport code XPTOR [613] to compare with experiment data. Density and temperature predictions were performed for 500 transport runs from 96 tokamak discharges. Sensitivities to various contributions are discussed.

Profiles are in excellent agreement with the measured profiles. The new model, TGLF has been used to make predictions for ITER [614]. These predictions show slightly less optimistic than GLF23 predictions for the fusion gain goals of ITER.

Predictive modeling [199] using the JINTRAC framework [189] studies the L–H transition in JET high current plasmas, and applies the results to ITER. An electron density waveform and plasma rotation are assumed. The transport is assumed to be NCLASS [615] in the core and either BgB [195] in L-mode or GLF23 in H-mode. The wall material is assumed to be carbon or tungsten. To agree with JET experiments, the core and edge transport rates need to be reduced at different rates as the power flowing across the separatrix increases.

The European ITM effort is described in [616]. The EFDA ITM-TF simulation framework aims to provide a standardized platform and an integrated modeling suite of validated numerical codes for the simulation and prediction of a complete plasma discharge of an arbitrary tokamak. It was used

for equilibrium reconstruction and linear MHD stability simulation chain which was applied to the analysis of the edge MHD stability of AUG type-I ELMy H-mode discharges and ITER hybrid scenario, demonstrating the stabilizing effect of an increased Shafranov shift on edge modes.

Interpretive simulations of a JET hybrid discharge were performed with two electromagnetic turbulence codes within ITM infrastructure showing the signature of trapped-electron assisted ITG turbulence. A successful benchmark among five EC beam/ray tracing codes was performed in the ITM framework for an ITER inductive scenario for different launching conditions from the equatorial and upper launcher. This found good agreement of the computed absorbed power and driven current.

This ITM framework has been replaced by the integrated modeling & analysis suite, IMAS to provide a standard framework for supporting scenario preparation and plasma operation at ITER [171]. This reference gives a description and application to H&CD sources for the different phases of a plasma discharge, including possible synergistic effects between the heating sources. A study of electron cyclotron heating absorption for an ITER helium PFPO scenario at 7.5 MA/2.65 T is also presented with a discussion on the edge parasitic absorption which arises under specific conditions.

A phenomenological model of the power threshold for the L–H transition developed by combining the energy equipartition between electrons and ions, and the L-mode scaling of the energy confinement is presented in [606].

ELMs can be controlled with RMP fields. Optimization for various ITER scenarios based on linear and quasi-linear figures of merit was studied in [617]. Optimal coil configurations differ for different scenarios. Effects of NBI rotation on ELM mitigation are discussed. Full utilization of three rows of ELMCCs in ITER is found to be essential to ensure both flexibility and robustness of ELM control.

For discharge modeling to be fully integrated it should include the plasma core, scrape off, and actuators, including the external coils. Plasma parameters such as temperatures, densities, pressures, and currents since plasma parameters are strongly coupled. If the modules used are validated and verified the model could give a reasonable guess of feasible plasma conditions achievable.

These codes are reduced in the sense that they use simplified models for tasks, such as computing equilibria and bootstrap currents. They should rely on gyrokinetically based codes, such as TGYRO and GENE to compute plasma parameters, such as temperature profiles. Iteration is often needed since the gyrokinetic codes need parameters from the integrated modeling codes, and vice-versa. The outputs are predictions of plasma parameters and performance, and their outputs can be combined with actuator characteristics to facilitate the design and development of actuators. Synergies involving NBI and other actuators are discussed in section 7.2.5 above.

Framework or workflows are being used to modularize various tasks such as plasma equilibria and transport. An example is FENIX [618] which is being developed on AUG. This

describes a use of simplified physics and plant system ‘control’ models combined with a representation of the AUG DCS to allow for fast simulation runs with reasonable prediction quality.

Other frameworks and workflows are being developed to couple disparate regions. The JINTRAC [189] suite of codes couples core with EDGE2D. Predictions of the transition from L-mode to H-mode in ITER plasmas are discussed [199]. JINTRAC includes optimization of the plasma density evolution. The transport assumed in the core is NCLASS and BgB in L-mode and GLF23 in H-mode. Anomalous transport is assumed in the edge, and is reduced with heating power.

The model is validated with JET carbon-wall experiments, and used to predict the L to H transition in ITER. The modeling finds optimal fuelling schemes. The ion particle transport in two mixed H/D plasmas in the 2016 JET campaign were studied [619]. TRANSP interpretive analysis was performed for one. Large diffusion rates are observed and confirmed by quasi-linear gyrokinetic simulations with QuaLiKiz [244]. ITM, and its successor the IMAS framework also have core and scrape-off regions coupled.

This IMAS study sketches an effort to build the high-fidelity plasma simulator which might become the reference for future predictions of ITER scenarios, and later for supporting ITER operation. In [171] the transport is calculated using the JINTRAC transport solver described in [202]. It focuses on the effect of NBI and fusion-born alpha particles on IC absorption and heating.

To study this effect, the NBI modeling is carried out by the NEMO beam deposition code associated with the RISK ion Fokker–Planck code. The fusion-born alpha particles are simulated by the SPOT orbit following Monte Carlo code, the IC wave absorption of all the plasma species is computed by the CYRANO code and the collisional power transfer from the IC-accelerated (thermal, NBI and fusion-born alpha particles) ions is simulated by the FOPLA 1D Fokker–Planck solver. The transport was the one used in JINTRAC. Modest synergistic effects between NBI ions, fusion-born alpha particles and IC waves are predicted in ITER DT scenarios.

There are many uncertainties about plasma responses. An example of an unknown is the extent of alpha ash accumulation in the core plasma discussed above.

The hybrid operating mode is simulated using the GLF23 and CDBM05 energy transport models [15] as discussed in section 5. Both fixed and free-boundary 1.5D transport evolution codes including CRONOS, ONETWO, TSC/TRANSP [240, 329], TOPICS [620] and ASTRA [335] were used. The simulations project to fusion gains of 5.6–8.3, β_N values of 2.1–2.6 and fusion powers ranging from 350 to 500 MW.

Integrated modeling including momentum transport, NTMs, and ELMs were also conducted for the ITER hybrid mode [18]. Predictions of the long-pulse operation in ITER with GLF23, BgB, SBM [184], MMM7.1, Weiland, CDBM [461] core transport models, EPED1-SOLPS pedestal, and boundary conditions were carried out by JINTRAC, ASTRA, and TOPICS codes [184].

Integrated modeling of steady-state scenarios and H&CD mixes for ITER comparing predictions for steady-state

mode predictions were studied in [17] using five codes: FASTRAN/ONETWO, TRANSP, CRONOS, TOPICS-IB, and ASTRA as discussed in section 6. One of the scenarios assumed weak magnetic shear scenarios integrate the plasma core and edge by combining a theory-based transport model (GLF23) with scaled experimental boundary profiles. This achieved with fusion gain $Q = 4.3$, is fully non-inductive fraction with bootstrap current fraction $f_{BS} = 63\%$, normalized beta $\beta = 2.7$ at plasma current $I_p = 8$ MA, and toroidal field $B_t = 5.3$ T using ITER day-1 H&CD capability. These codes demonstrate good agreement in power absorption and driven current density profiles. Note that the difference between the measured and simulated profiles is less than the difference between different code predictions, and therefore has physical rather than computational reasons.

The results of integrated modeling of plasma parameters and H&CD, fueling and magnetic control systems required for steady-state operation in ITER for actuators designed for ITER is presented in [447],

Integrated modeling of the current profile in steady-state and hybrid ITER scenarios was studied in [323]. This paper considered NBCD, ECCD, and LHCD. The simulations indicate that generation and control of internal and edge barriers to access and maintain high confinement will be a major undertaking for future simulations, as well as a challenge for the ITER steady-state and hybrid regimes.

The ITER low activation phase operation scenarios are also studied in [182] (see section 3). It discusses simulations by the JINTRAC suite of codes including TRANSP, CORSICA with EPED1 pedestal, and SOLPS boundary conditions for ITER discharges in hydrogen and helium. This paper concludes that hydrogen pellet fueling into He plasmas is needed for optimizing ICRF power absorption, NB shine-through density limit and H-mode.

The impact of H&CD in the ITER hybrid scenario using CRONOS and GLF23 was studied [346]. Various choices of H&CD, and assumptions for the pedestal temperatures were optimized for improved confinement.

Operational constraints and the H&CD capabilities for ITER scenarios in hydrogen and helium plasmas were studied to assess their viability, and how to address with them [526]. The modeling results show that 5 MA/1.8 T scenarios are viable and will allow the exploration of the H-mode physics.

High-density H-mode operation by pellet injection and ELM mitigation with the new active in-vessel saddle coils in AUG is discussed in [604].

In general, different types of control could be tightly connected. In particular, control of H-mode operation in ITER requires integrated approach taking into account the CF, divertor detachment, and ELM pacing control [603].

There are many other types of actuators which are not directly relevant to the H&CD and density issues discussed above. Examples are magnetic control coils discussed in detail in section 8. Since this is important for integrated models it is discussed here. The IBL scenario’s end-to-end start-up and shutdown coil specifications were modeled by [19] using the CORSICA [197] for the free-boundary equilibria and DINA [52] for the thermal plasma transport. These self-consistent

models include the CS and PF coils to provide CD and feedback control of the plasma current, position, shape, and stabilization of unstable motion. CORSICA is used to study performance sensitivities while DINA is exploring the use of fast controlled current ramps. This paper concludes that the ITER design provides control needed for baseline discharges.

The results of integrated modeling of plasma parameters and H&CD, fueling and magnetic control systems required for fully non-activation operation in ITER for actuators designed for ITER are presented in [445]. Transport and source modeling suites (METIS, ASTRA, and CORSICA) and ideal MHD stability analysis codes (KINX and MISHKA) are applied to combined analysis and scenario development. The CORSICA scenario modeling suite has been applied to study access to the target plasmas as well as to develop a candidate operation scenario.

The ITER PCS will make use of about 45 different diagnostic systems and 20 different actuator systems to determine relevant plasma conditions and produce the necessary coil current waveforms, heating effects, fueling, etc. Methods for active current profile control and tearing mode suppression by pulsing gyrotrons to drive current within a magnetic island have been studied [621].

A new approach to current control is to use ML to construct a controller. An architecture for tokamak magnetic controller design which autonomously learns to command the full set of control coils is described in [622]. This was applied to control currents in TCV.

7.5. Outstanding issues

A key issue for plasma control is the predictability of the impact of actuators on plasma parameters (plasma response). We group the issues relevant to H&CD and PLCS controls, giving references to relevant papers and sections of this review with more detailed discussion on the issues.

Classes of issues are:

(1) The models used for developing DT plasma regimes and their actuators need lots of improvements. Lots of efforts in the past have been devoted to reinventing existing models, and often using simplistic physics bases. Much more effort is needed in incorporating more realistic physics, such as gyrokinetics and Monte Carlo methods, improved atomic data, and in validating and verifying the models with experiments.

Examples, related to interplay of control, are related to PLCS and H&CD are listed [597] in the subsection above, and mentioned in the description of the synergetic effects between different H&CD actuators, [479, 480, 623]. There is also impact of H&CD on particle transport such as pumping of particles from the core with ECRH [270].

(2) The plasma control is nontrivial in the case of synergies between actuators and alpha heating, and the interplay with actuators related to different control. The validation of predictability of the integrated control with interplay of different actuators, as well as operational boundaries is one of the remaining issue for H&CD and PLCS, and requires further investigation.

An issue related to integrated control is compatibility of density control with the ELM mitigation/suppression. Experiments show that operation without ELMs or small ELMs is possible in some certain range of pedestal densities, lower than in ELM type-I operation both with gas puffing and pellet fuelling. Such decreases of density can be provided by different means including active control by RMPs [607, 624], vertical kicks [625], reduction of gas puffing [625], Neon puffing [235].

Such limitation of density rise with ELM-free operation and can be caused by some turbulence, which affects the pedestal density growth, like in the quiescent H-modes [270], long inter-ELM operation, [626, 627], EHO, etc. It was found that when the density was increased in ELM suppressed plasmas by gas puffing or pellet injection to a certain level, the type-I ELMs come back. Conditions for ELM-free operation are possible such as operating with low B_t [270] or in a narrow range of q_{95} . Understanding such phenomena, required to design the integrated schemes of PLCS and ELM suppression control, is under investigation.

(3) There are cases when application of an actuator causes an anomalous plasma response, mentioned in sections 7.1–7.3. Such anomalous response requires theoretical explanation with further validation of predictability.

One of the most impressive phenomena of this class is the anomalous response of equilibrium and driven current on H&CD actuators. In many experiments with $q_{\min} \sim 1$ as discussed in section 5, it was found that there are some stable saturated helical structures, which in general require 3D equilibrium, stability, and transport simulation.

The interpretation of H&CD obtained in such experiments in the 1D transport approximation requires assumptions on the anomalous radial diffusion of fast ions [298] and ECCD [273]. The existence of the stable 3D helical core was reported also in AUG experiments with off-axis NBCCD [507, 628] and off-axis ECCD [373], as well as in JET experiments with stable 3D helical core [629].

There are a few physical mechanisms considered to explain the existence of stable 3D structures with high central pressure without sawtooth. They include saturated NTMs, 3D stable snake-like structures [630], partial reconnection with stochastic layers, fishbones, dynamo effects, stabilized kinks or interchange modes, etc (see [219, 273, 297, 298, 507, 628, 629, 631, 632] (partial reconnection (stochastic) in AUG), [373], etc).

It is natural to expect that in presence of fast toroidal rotation of such 3D helical structures would be visible in such experiments, the NBI and ECH absorption should be spread radially over the area swept by the rotating saturated island or snake. Such spreading was not addressed yet in the analyses of such experiments. It is likely that such a ‘sprinkler effect’ of radial spreading of the source at the phase of absorption will reduce the anomalous diffusivities of fast particles, required for interpretation of experimental data in a 1D approximation.

For such advanced confinement high temperature operation with strong auxiliary heating, the population of energetic particles is higher than for the baseline H-mode operation. Fast particles can create and stabilize the 3D equilibria including

snakes and affect stabilization of 3D equilibrium by fast ion anisotropy [633]. Fast particles can also affect microturbulence and so the radial transport of thermal particles [466, 584, 634, 635]. Such phenomena are addressed in chapter 7.

Another caveat is that high level of confinement, which looks sufficient for fully non-inductive operation in ITER ($H_{98y2} \sim 1.6$) is achieved on DIII-D with very high rotation and rotation shear, obtained with high NBI torque injection. Experiments with variable injected torque have shown strong correlation between confinement improvement and the speed of toroidal rotation [296]. It looks necessary to clarify how such rotation affects the confinement improvement, how that confinement improvement affects the rotation and shear of rotation and how it could be extrapolated to reactor scale plasmas with higher torque, confinement and plasma mass, but lower ρ^* and ν^* .

(4) Extrapolation of the phenomenology related to control by H&CD and PLCS actuators to reactor scale plasmas. Further analyses of such issues require better theoretical understanding and extension of experiments on present machines as close as possible to reactor scale parameters, and comprehensive integrated simulations, discussed below for pellet injection and ITBs.

For efficient H&CD by NBI it is necessary to increase the NBI energy to MeV-scale energy [149]. For such energies, multistep processes play a major role in beam ionization [487, 488]. This increases the efficiency of the NBI absorption, reducing the NBI shine-through loss, and thus extends the operational space to lower densities, beneficial to transition to good H-mode confinement [255]. Cross-sections of such processes are based on theoretical consideration which require experimental verification, especially for the high-Z impurities where the theoretical approximations have less accuracy.

Another example of this type of issues is related to plasma fuelling. In contrast to present-day experiments plasma fuelling from the NBI will be very small for high NBI energies as well as the role of gas puffing in the H-mode operation [479, 636], making the pellet fuelling the most efficient fuelling technique. According to simulations in H-mode reactor size plasmas, a pellet is ablated within the pedestal [23, 479]. However, in present-day experiments, the ELM triggering by pellets starts when a pellet reaches half of the pedestal width. It is not clear if this applies to HFS pellet fuelling. This could affect the final deposition and fuelling efficiency, in particular density pump out by ELMs triggered by the pellets. The predictive capabilities of the pellet-simulation codes for the case of fusion reactors, such as ITER or DEMO, require further investigation.

An example is extrapolation of single main ion experiments typical for present machines to two-main ion deuterium and tritium, essential for a reactor. Simulations predict strong mutual impact of D and T ions for neoclassical and turbulent transport in different dominant turbulent transport mechanisms. Non-linear gyrokinetic simulations show tritium to have better confinement in ITG-dominated and TEM-dominated regimes [581]. References [598] and [599] study differences in the deuterium and tritium densities in ITER. The first considers anomalous transport due to ITG and trapped electron drift

instabilities. The second, calculations using a 2D fluid model with the code package B2-EIRENE (SOLPS4.3) to study densities in the ITER divertor region. It takes into account the different charge-exchange rates between ions and neutrals in the scrape-off and divertor regions, and finds reduced transparency for tritium atoms.

According to predictions for reactor scale plasmas, the SOL will be opaque for gas penetration to the core [494], which requires fuelling by pellets. In this case, density control at the pedestal can be done independently from that at the separatrix. Such simulations predict for the reactor scale plasmas, a difference in collisionalities at the pedestal and separatrix, which corresponds to the high confinement, $\nu_{ped}^* \ll 1$, and low ELM loss, $\nu_{sep}^* \gg 1$, similar to [235, 637, 638]. The possibility of such operation should be studied in further experiments and needs development of predictive models. Further understanding of the role of the edge collisionality of plasma performance is required [639].

The role of collisionality in predicting the density profile in ITER is extrapolated from dedicated density scans in AUG [640]. Modeling these discharges with ASTRA using TGLF quasi-linear turbulence predicts relatively flat electron density profiles.

Another example is the extrapolation of empirical phenomena, observed in the machines with dominant gas fuelling, such as confinement saturation [472] and LOC to SOC transition [641] in the ohmic phase at some density, n_{sat} , H-mode density limit [458], confinement degradation with n_{ped}/n_{sep} [642], on the reactor scale plasmas. For pellet fuelling experiments the saturation density and energy confinement increase [643].

(5) Operation beyond the boundaries of controllable operational space occurs, when regular control by H&CD and PLCS actuators becomes impossible for reasons independent of the design of the actuators.

An example of this class is the operation in fusion plasmas with the ITBs. Note that there are two branches of steady-state operation, type-I, with ITB and type-II without the ITBs, [11], (see also discussion therein). In present-day experiments on DIII-D with weak and weak reversed magnetic shear both types of operations are possible [298, 644] and provide high energy confinement $H_{98y2} \sim 1.6$ sufficient for steady-state operation in a reactor (ITER) [447]. Note that extrapolation of the ITB cases to reactor conditions is not straightforward. In reactor plasmas, the presence of the ITB could cause self-supported growth of fusion power [636] which would require special actions by systems like DMS. In a reactor scale plasma, in contrast to present-day machines core fuelling from the NBI is negligible [494].

Thus the ITB could cause isolation of the core from fuelling and cause locking of the He ash in the center with further degradation of plasma performance. If such potential problems are inherent for ITB operation, it would be necessary to develop a control to avoid the ITBs or provide an actuator, which can destroy an ITB if appears. Steering a heating source such as ECH away from the core or using ECCD to alter the q -profile might alleviate this problem. Current profile tailoring requires further development.

Issues for ICRF include developing satisfactory ICRF scenarios for hydrogen operation at 2.65 T. This may be compensated by optional schemes at 3 T and 3.3 T with fundamental minority ^3He heating in a $\text{H-}^4\text{He}$ mixture (three-ion heating scheme). Another issue is the interplay of ICRF scenarios, fast ions, confinement (turbulence), and MHD (AEs). The fast ions may sometimes stabilize the ITG modes [466].

Another issue is control of the minority species in the heating scenarios of ITER, including measurement of their concentration, feedback control schemes, and possibly minimization of the ^3He budget (a scarce resource). For some scenarios, the envisaged minority species are intrinsic impurities (not controlled). The elimination of Be FW material in ITER removes this as a worry.

There is the potential to control sawteeth with ICRF (in case ECCD is late). Two possibilities are:

- (1) Off-axis ICRF (close to $q = 1$) to destabilize the sawteeth should work well but you have less peaked ICRF heat sources [645].
- (2) On-axis ICRF modulation to trigger sawteeth should work well with moderate beam power even with low alphas densities. The ICRF wave deposition is predicted to be much more peaked than the alpha power profile. It will not work in active phase due to strong alpha stabilization. This requires real-time control based on EC H&CD or soft x-ray emission measurements [646].

Further investigation is needed of the use of ICRF to ease the plasma start-up and for wall conditioning in the presence of a permanent toroidal field.

Controlling the plasma conditions in future commercial fusion power plants when the alpha heating power is larger than the auxiliary heating ($Q \gg 1$) is an important issue beyond our present consideration. It is foreseen that the H&CD techniques considered above could be still important to provide safe trajectories in the operational space from small Q to the stable branch of burning with $Q \gg 1$, and then back to plasma damping. They also could be still useful for fully non-inductive operation and for current profile tailoring to avoid or stabilize NTM, sawtooth, and other modes affecting the performance. In current experiments, adjusting the heating power is a strong lever for control.

An important application of ECCD is magnetic flux pumping as discussed in subsection in section 5. This has the potential of increasing the non-inductive current, needed for steady-state operation. A self-regulating magnetic flux pumping mechanism which maintains the core safety factor near unity, thus preventing sawteeth, is analyzed in non-linear 3D magnetohydrodynamic simulations using the M3D-C1 code [372]. In these simulations, the most important mechanism responsible for the flux pumping is a quasi-interchange instability generating an effective negative loop voltage in the plasma center via a dynamo effect. It is shown that sawtooth-ing is prevented in the simulations if β is sufficiently high.

Experimental studies of magnetic flux pumping in ASDEX-U have been studied [373, 563]. These show that widening

of absorption/CD for central ECCD clamps q_{min} close to, or above unity eliminating sawteeth and resulting in higher confinement. Similar studies in DIII-D are described in [564].

Experimental evidence of magnetic flux pumping in the DIII-D steady-state plasmas is given [297]. The hybrid regime overcomes the need for off-axis CD efficiency, taking advantage of poloidal magnetic flux pumping which is believed to be the result of a saturated 3/2 tearing mode. This allows for efficient CD close to the axis, without deleterious sawtooth instabilities. This is also discussed in section 5 as a mechanism to form a q -profile in hybrid scenario.

Other self-generated current sources have been observed originating from turbulence [376–378] or an unknown mechanism [379]. Micro-instabilities such as TEM or ETG mode can generate or modify the plasma current. In the KSTAR experiment, an L-mode plasma with ECH was studied and simulated with the TRIASSIC integrated suite of codes [647], where an anomalous off-axis current was deduced up to 30% of the total plasma current [379].

7.6. Discussion and conclusions

The history of tokamak fusion research shows that each major new experiment brought online has led to significant, unanticipated discoveries. Some discoveries were beneficial for fusion, and some detrimental. Early examples are the high electron temperatures seen in T-4 and the high ion temperatures seen in PLT. Other examples are sawteeth, the L-mode, and disruptions in PLT, the H-mode in ASDEX, and supershots and enhanced confinement with reversed shear in TFTR.

By extrapolation, we should expect major unanticipated discoveries in ITER and beyond. ITER will be a major technological advance far beyond our current experiments. We have very little experience with DT operations and with high density and high B_t . Thus we should be humble about our expectations. Many of our ‘ITER-relevant results’ will not be relevant. Examples will probably include many of the NBI effects.

A large burden is placed on extrapolating various high confinement regimes beyond present experiments. One method for extrapolating plasma parameters uses dimensionless scaling arguments. Better physics understanding is important for extrapolation.

A major advance in characterizing and understanding turbulence resulted from gyrokinetic perturbation theory. This is being used to build accurate models of non-linear turbulent transport fluxes using flux-driven (global) and quasilinear (local flux tube) models. These non-linear models are gradient-driven, and the quasilinear models use the spectra of linear instabilities. Examples are the TGLF and GENE transport codes. Future major advances in understanding and modeling are expected to result from advances in computers and computer science tools such as ML and AI.

Integrated, validated and verified modeling is important for designing actuators and understanding their effects. Results give the anticipated environments the actuators will experience, and in turn the actuator effects are incorporated as

boundary conditions for the modeling. One outstanding issue discussed in sections 7.3 and 7.4 is improving the physics content of the modeling. Various work flows are being developed for coupling various models, but frequently simplified physics assumptions are used. Some of the references do not define the model being used. Arbitrary models can produce arbitrary predictions.

The discussions of actuators above are separated into sub-sections containing (1) generalities, (2) model benchmarking, and (3) model experimental verification.

Highlights are:

- (1) A major advance in codes for simulating fast ions was the introduction of Monte Carlo methods in TRANSP, and later in the standalone NUBEAM. The same techniques for simulating fast beam ions are used for simulating fast alpha effects. Multiple fast beam ion codes have been benchmarked with NUBEAM.
- (2) The NBI shine-through calculations have been improved using these codes with newer atomic cross-section data. Multi-step ionization processes are computed to have significantly reduced threatening beam ion shine-through at low plasma densities.
- (3) Experimental verification of beam driven current calculations is important for conforming its computed contribution to the q -profile, which in turn significantly affects the plasma confinement and MHD stability.
- (4) Ion cyclotron range of frequency provides an alternative H&CD actuator having the potential of being more reactor-relevant than beam heating. Large advances have been made in modeling the antenna and plasma interactions. Significant advances in the ICRF wave modeling and benchmarking have occurred. The 3D aspects of the antenna and the plasma in the scrape-off region of the antenna are being coupled. The tungsten wall being constructed for ITER is challenging since ICRF waves can create energetic ions capable of sputtering or otherwise eroding the antenna and FW materials. Various minority species and plasma regimes are being studied for ITER. A revived interest in three-ion heating modes is being pursued.
- (5) EC H&CD are planned for ITER. Heating in the ITER pre-fusion phase are being studied. These hydrogen or ^4He plasmas with one-third or one-half toroidal field are expected to obtain the H-mode only with low density. The requirements of low NB shine-through set a lower limit ($\sim$ half the Greenwald limit), and the desired high single-pass wave absorption sets a lower limit on the electron temperature.
- (6) ECCD is considered for magnetic flux pumping by periodically altering the plasma current. This could help maintain the plasma current for long durations. Experiments have demonstrated this, but the mechanism is not well understood.
- (7) EC H&CD can be useful for controlling NTMs by heating and/or CD inside islands.
- (8) LHCD is not planned for ITER, but it has the very attractive properties for providing steerable CD.

- (9) Traditional actuators for controlling the plasma density include gas puffing, pellet injection, and NB fueling. The NB fueling rate in ITER will be ten times lower than that in current tokamaks since the beam energy will be ten times higher. A new actuator that might be useful for DEMO and beyond is spin polarizing the deuterium and tritium fuel. If they are polarized in parallel their fusion would have a 50% higher than with unpolarized fuel since their spins of 1 and 1/2 would add to the spin 3/2 of the ^5He resonance which dominates the fusion amplitude. This process was studied in the 1980s but the methods for spin polarizing gas were not efficient. Since then, efficient methods have been developed for high energy physics accelerator experiments. Besides the cross-section enhancement, polarization can alter the angular distribution of the fusion products, perhaps reducing neutron damage to the inner wall and/or directing neutrons toward tritium breeding blankets.

8. Scenario control

Coordinated by E. Schuster

8.1. Control needs and issues

The greatest risks to ITER and future fusion reactors come from potential machine damages caused by uncontrolled transients leading either to disruptions or to heat and particle exhaust excursions above safety boundaries imposed by the PFCs. In particular, the loss of control leading to disruptions arises from the plasma simultaneously crossing stability and controllability boundaries. The violation of these safety, stability, and controllability boundaries happens as performance demands push operation closer to these boundaries. Robust, simultaneous, integrated control of a large number of plasma properties in the core, SOL, and divertor is a critical technology needed to operate within these boundaries and to guarantee safe and disruption-free tokamak operation.

The development of robust, integrated control solutions for high performance DT plasmas presents a formidable challenge. While control strategies are often developed and tested in present-day low activation plasmas, significant uncertainties arise when extrapolating these methods to the vastly different conditions expected in a burning DT plasma. Variations in plasma physics, diagnostic accessibility, and actuator efficacy, coupled with the critical need to prevent disruptions in a tritiated environment, necessitate rigorous reassessment of control system reliability. The inherent differences between these operational regimes underscore the need for advanced modeling, diagnostic development, and control algorithms capable of adapting to the complexities of high-power DT fusion. Addressing these challenges is paramount to ensuring the successful and safe operation of future fusion power plants like ITER.

The ITER PCS consists of a large and complex set of functions that enable the controlled execution of all the envisioned ITER operation scenarios. The set of functions can

be broken down in a basic list, grouping the functions as those performing:

- (1) magnetic control
- (2) kinetic control
- (3) control and mitigation of instabilities
- (4) disruption prevention and avoidance
- (5) FW heat load control
- (6) control of RF wall conditioning methods
- (7) exception handling.

However, the creation of any clear breakdown of tasks can be challenging, and there is always some degree of overlap between. Previous publications already detailed the requirements of ITER plasma control and proposed concept control solutions and discussed simulated control performance [33]. Here a brief outline is provided, emphasizing some key issues and how these can be resolved by applying the correct design approach.

Each of the above groups of the functions can be subdivided into a multitude of sub-functions. Magnetic control involves the control of the magnetic configuration, using the PF coils, the plasma current, and the detailed plasma equilibrium. In early ITER operation, kinetic control implements the basic control of the bulk plasma density but later evolves into kinetic profile control to optimize plasma performance and its stability, eventually also regulating the fusion burn. Instability control focuses on the control, both continuous and event-based, of key instabilities like vertical instability, error fields, ELMs, NTMs, as well as RWMs and fast particle-driven instabilities at the later stages of ITER operation, when high beta operation in the presence of large amount of alpha particles is expected. The ITER PCS will have to be designed such that the control is robust against disturbances, preventing loss of control leading to disruptions, for example, due to actuator glitches or diagnostic failures. In addition, PCS exception handling should provide adequate measures to avoid disruptions when such disturbances become too large. The proper design of exception handling, which is often overlooked when considering tokamak control, is a task that in size can easily exceed work to design continuous functions [648]. Therefore, it is listed separately above, even though one could argue that it does not constitute an individual function. PCS exception handling will set the basis to reduce the overall ITER disruption rate, while those disruptions that slip through this first safety net are to be mitigated by the ITER DMS. First-wall heat load control ensures that the pulse is executed such that the heat loads on the ITER blankets and divertor remain within bound, both from steady-state loads and transient loads, such as those due to ELMs.

The fact that the behavior of many PCS functions is coupled complicates their design. This coupling can originate due to a combined response of the plant, by different control functions. For example, changing the current into one of ITER magnets also has to consider the other magnets as well. It can also be due to the fact that the same diagnostic input is used by multiple control functions. While some ITER control

functions have to share actuators. For example the ITER ECH system is utilized to carry out multiple actuation roles, such as NTM control, temperature profile control, and auxiliary H&CD [151]. A strong functional coupling, often overlooked, is the need for a coordinated response of many different functions in case of an exception. For example, a request for the termination of the tokamak discharge demands a coordinated ramp-down of the plasma current, the thermal energy, the particle content, while at the same time considering various operation constraints and avoiding operation limits [649].

There are a few important differences between the design and deployment of the ITER PCS, compared to those systems on current devices. Firstly, the size of the ITER plant and the length of the discharge will impose more constraints on control. For example, long guide tubes for gas and pellets result in a considerably delayed actuator response and the plasma itself may also react slower to the injection of any particles, which must be taken into account when developing the fueling control system. Furthermore, diagnostics should be capable of measuring significantly larger ranges than assumed in current devices, while they need to provide high-quality data over long-duration plasma discharges. Secondly, although many of the required functions for ITER have been tested individually on current tokamaks, ITER will be the first device that will have to deploy all these functions simultaneously, placing constraints on actuator-sharing as well as highlighting the fact that issues related to the coupling between all functions have to be resolved. Thirdly, for some functions, ITER will operate in a range, no other tokamaks have operated before. This is especially true for PCS functions that will control the fusion burn. But also for those controlling various instabilities: NTM and ELM control will have to be deployed with a higher quality of performance, during most of its operation, and thus not only be developed for tests purposes as if often done in current experiments. And finally, the design of the control functions is to be prepared for the future. In current experiments, some new control functions are designed, simulated, implemented, tested, and optimized by a single team. However, as the ITER project will develop over decades, PCS control functions will have to be designed now for the future, and will likely be implemented and tested by others.

At ITER, many control functions cannot be heuristically developed by trial-and-error on the actual machine [621]. A clear prior demonstration of a control function meeting its design requirements is therefore key. For example, for VS control, it is obvious that ITER will have to prove it has a robust function that is capable to control the vertical plasma position during planned operation scenarios. However, the same is true for all other functions; even for ELM and detachment control, for which no control models are yet available. Hence, a much greater emphasis on predicting the control performance by simulation, for the ITER PCS design, requires the development of a large set of control-oriented models. A systematic approach, that emphasizes how to design the PCS and places a large effort to simulate and assess all aspects that might affect the control function, is therefore needed. The ITER PCS design has progressed over the last decade from

the conceptual to the preliminary design level, and the first detailed final design for ITER First Plasma Operation has been completed [140, 650]. The design follows the ITER staged approach [10], hence, PCS functionality is developed based on the need of the next ITER operation stage. Hence, the proposed physics operation scenarios for a given stage, function as important stakeholder requirements that define the design. The ITER PCS design follows a system engineering approach that helps to manage the complex design description linked to the coupling between the many PCS functions [651, 652]. It also employs a PCSSP on which all functions are designed and their performance is simulated also integrated with others [653].

Plasma control research within the ITPA is carried out by different TGs. This section provides an overview on recent progress in IOS-relevant plasma control topics. This review, which does not intend to be fully comprehensive, explicitly excludes plasma control topics covered in chapter 4 such as magnetic control, disruption mitigation, NTM/RWM stabilization, and runaway electron control, as well as chapter 5 such as ELM control and divertor control. Moreover, this review does not cover diagnostics and actuators, two key components of control systems. Diagnostics-related research is dealt with in chapter 8. On the other hand, actuator-related research, which is indeed carried out by the IOS TG, is separately addressed in a different section of this chapter (section 7). The section is organized as follows. The role of controls in the robust realization and sustainment of different plasma scenarios, as well as the constraints imposed by the control algorithms on scenario development, are both discussed in section 8.2. The need of control-oriented modeling for scenario development and control is stated in section 8.3, where the features and different applications of control-oriented models are also presented. Different key components of scenario control are introduced in section 8.4, which include profile control (section 8.4.1), density and β control (section 8.4.2), burn control (section 8.4.3), initiation control (section 8.4.4), and termination control (section 8.4.5). It is worth mentioning at this point that the physics aspects of plasma initiation are covered in detail in section 2. The challenges associated with control integration and actuator-sharing are addressed in section 8.5. Exception handling is discussed in section 8.6, while disruption prevention and its connection with MHD stabilization are presented in section 8.7. Challenges associated with long-pulse operation, focusing on core/divertor integration, are addressed in section 8.8. Outstanding issues are discussed in section 8.9 and a brief conclusion is presented in section 8.10.

8.2. Scenario development

Important requirements for the design of ITER control functions are set by the physics operation scenarios. These are detailed use cases of how the ITER tokamak plasma discharge is to evolve during a so-called tokamak pulse. It presents a time-ordered set of objectives or plasma states that define the intended pulse or discharge development. Many detailed physics operating scenarios of ITER discharges have been

simulated, often already taking certain concept control and operation limits into account. Designing an ITER operation scenario also takes into account the capabilities of power supplies, heating, fueling, and other plant systems [21, 171, 182, 255]. The IRP identifies specific operating scenarios to be achieved per ITER operation stage [10]. Importantly, these operating scenarios are important stakeholder requirements for the PCS design, outlining what means and how it needs to achieve.

The attention usually lies on ITER full performance operation, that is, plasma discharges with a plasma current of 15 MA, full auxiliary heating power, in H-mode, achieving $Q = 10$, thus with the plasma heating dominated by α particles. However, it is wrong to assume that the PCS is only challenged during full-performance scenarios. Often control can be more challenging, when operating at lower performance, also with fewer actuators or lower actuator capacity available, for example during the early stages of ITER First Plasma Operation. Physics operation scenarios also indicate important constraints on the plant capabilities, or operation limits that should be taken into account for the design of the PCS as well. A variety of physics operation scenarios should be considered, from which detailed control required are to be derived. The PCS design will have to show that it meets these requirements by simulating its behavior in achieving physics operating scenarios.

Regarding ITER physics operation scenarios, it is important to consider the step by step expansion of the operation range, initiating operation at low plasma currents (i.e. a few MA), that is increased together with the levels of auxiliary heating and plasma energy, as ITER operation progresses. Not all PCS functions are required from the start of ITER operations, e.g. ELM and detachment control are only needed when H-mode operation of certain levels of exhaust power is achieved. On the one hand, this allows one to test some PCS functions individually before integrating more of them as ITER operation progresses. On the other hand, it means that these control functions may need to be tested in these early physics operation scenarios, and the question is if this is feasible. Furthermore, it means that in the early stages of ITER operation, actuator and diagnostic capacity might be restricted which has to be taken into account in its design as well.

8.2.1. Constraints on scenario development by control approaches.

The difference in simulations to design a physics operating scenario and simulating the PCS to achieve this scenario is that the former often relies on high fidelity behavior models, often focusing on the plasma or the plasma magnetic equilibrium while ignoring detailed simulation of actuator and diagnostic systems. The latter, makes use of control-oriented models (section 8.3), focusing on the integrated simulation of all control functions, including detailed behavior of actuator and diagnostic plants, including fault responses. On the other hand, control design integration is often needed for physics scenario design as well. The capability of certain control functions is essential in the understanding of what physics operation scenarios are feasible. And this is not simply due

to the limitations of the available actuator capacity, which is usually taken into account when designing physics operation scenarios but also due to the complex coupling between some of the scenario assumptions and the control performance. For example, VS control; its capabilities and limits should be known to understand what operation in ITER is feasible. The VS control is not only determined by the capability of the coils that provide the actuation but also by the assumed plasma equilibrium (e.g. its elongation and internal inductance) and its proximity to the stabilizing wall [654]. To a lesser extent this is also true for other control functions, such as equilibrium and position control, density control, and more complex control functions, such as ELM control, burn control or NTM control. Hence, some of these control issues are to be covered early on, during the ITER tokamak conceptual design. Therefore, an iterative approach is needed, developing and designing possible physics scenarios and control designs.

Thus, besides that physics scenarios can determine control function requirements, control approaches may in turn set constraints to physics scenarios. In the first place, this is evident when developing the plasma breakdown and initiation scenario [91]. The scheme that is to be deployed by ITER depends strongly on how to control the pre-fill pressure and on how to control the CS and PF coils that generate the toroidal electric field and the magnetic field configuration that allows the plasma to form. But the capability to control the plasma equilibrium and initiate the controlled current ramp-up (section 8.4.4) will also impact how the transition is made. Similarly, the start of the current ramp-down, or termination phase (section 8.4.5), needs to consider how to optimize the radial position control, in view of the pressure drop associated with the H–L back-transition, while also maintaining VS control at increasing internal inductance (due to the reducing plasma current) by reducing simultaneously the elongation. A final example is the controlled transition into a controlled fusion burn (section 8.4.3), which will require not just a transition from L- to H-mode, but one that ensures an optimized increase of the amount of α power produced by the fusion reactions. The optimization should avoid too large an overshoot of the fusion power, complicating its regulation. But it can also undershoot, with the risk affecting the L to H transition, which in turn can start a reduction in α power (a so-called fizzle). Thus, the simulation of physics operating scenarios is an iterative process and has to progressively add more complex control functions as well, to better predict the discharge behavior.

8.3. Control-oriented modeling for scenario development

The cost and complexity of a single burning plasma discharge on a tokamak like ITER warrants intensive use of model-based methods for preparing and designing tokamak scenarios. These methods allow to design, test, and validate tokamak discharges ahead of time, including the design and testing of both the reference waveforms and feedback controllers in the PCS needed to achieve the experimental objectives of the discharge.

There are multiple reasons to use model-based approaches for developing and designing plasma scenarios, as opposed to,

for example, trial-and-error tuning of scenario waveforms and control parameters in the experiment. The first is the possibility to start with an already optimized plasma scenario, with associated control parameters, based on prior knowledge of the expected plasma behavior. Also, the risk of human error is decreased by automated validation of discharge parameters based on model simulations of the closed-loop behavior of the PCS with the tokamak. Finally, use of model-based scenario development tools allows to automatically survey the operating space and find optimal settings, for example to achieve the highest possible performance within the physics and engineering constraints included in the model.

Any trial-and-error required on the experiment itself is thereby restricted to that due to mismatches between the model used for designing the scenario/controllers and the true behavior of the system. This also points to the need to develop semi-automated methods to account for these model-reality mismatches and iterate rapidly. Also these methods can use model-based approaches, data-driven approaches, or a combination thereof.

For these reasons, so-called control-oriented models of tokamaks have been developed. This field has seen great progress over the past two decades, with many examples of their use in existing devices and simulations for ITER and other future devices.

8.3.1. Features of control-oriented models. Control-oriented models differ from physics-oriented models mainly in their purpose. Physics-oriented models are traditionally oriented towards understanding and interpretation of physical behavior of a plasma, and attempt to include as much as possible the physics effects deemed responsible for this behavior. On the other hand, control-oriented models are aimed at simulating only the specific aspects of the plasma behavior that are relevant for scenario and controller design, validation, optimization or similar engineering/operations purposes. These require fast execution times, on the order of seconds or minutes for a representative simulation, in order to be compatible with iterative workflows such as controller design, optimization, validation and testing [141].

This means that, to a greater extent than physics models, control-oriented models must make strongly simplifying assumptions regarding the physics of the system being simulated. Often, they rely on ad-hoc formulae replacing first-principles physics models, or use external parameters tuned on experimental data rather than attempting to simulate a given effect self-consistently. This being said, there have been significant recent efforts to bridge the gap between these two model domains.

Additionally, control-oriented models are often structured in specific mathematical forms that render them particularly useful for control/scenario design and optimization purposes. For so-called ‘forward’ models, used for controller design and simulation there is usually a very clear distinction between input quantities representing actuators affecting a physical process, internal states representing the internal (plasma) evolution, and outputs being both the plasma state and expected

diagnostic measurements. These represent the inputs/outputs required to then close a feedback loop with a controller. Other control-oriented models can be formulated in ‘inverse’ mode where the input is the desired (plasma) state and the output is the actuator settings (possibly in time) to be applied to obtain this state.

Control-oriented models can be categorized as data-based or physics-based. Data-driven models rely on data (from experiments or simulations) to extract a model of the time-dependent behavior system being studied, which is referred to as *System Identification*. These models are typically valid around an operating point and are used to derive (linear) controllers. An example thereof is the use of simulations and/or experiments to derive linearized models of the 1D profile evolution dynamics for designing profile controllers [655]. Another example is shown in [656], where a model obtained from a small number of System Identification experiments was used to design a controller for the radiation front on TCV. Physics-based control-oriented models, instead, begin with known physics from which the simplified dynamic equations for a control-oriented model are derived. In some cases, data-driven approaches are used to identify some of the unknown parameters in the physics-based model, for example [657, 658].

8.3.2. Applications of control-oriented models. Control-oriented models can serve a multitude of applications which can be categorized as forward simulation, pulse schedule or scenario design, real-time observers or real-time prediction. These are described in more detail below.

- Forward models for feedback controller design and validation:
 - Dynamic models of the to-be-controlled plasma quantity are an essential starting point of controller design. In some cases, feedback controllers can be synthesized directly from models of the dynamics of interest in a suitable form, typically linear or linear-time-varying models, using model-based design techniques such as Loop Shaping, Robust control, MPC, or LQR (optimal) feedback control. In such a workflow, the controller is synthesized and then tested on the design model itself, allowing the control designer to tune any control design parameters.
 - Then, increasingly complex models can be used to validate the controller to an increasing level of physics fidelity, possibly including validation with several control loops simultaneously to check for interactions.
 - Models that simulate an entire plasma scenario can be used for the application of pulse design validation. Such a simulator receives the pulse schedule and simulates the expected evolution of all the key discharge quantities. Such a simulation does not strictly need to include all the feedback loops, but should verify that the planned discharge is feasible and stays within operational envelopes for both the plasma state and any engineering/operation limits.
- Ultimately, the entire PCS can be tested in closed-loop with a control-oriented model of the entire device behavior. Such closed-loop simulations, sometimes referred to as ‘tokamak flight simulators’ [659] and are distinct from the previous above in the sense that they include all the relevant processes to assert the correctness of the PCS implementation and configuration. In this sense, the tokamak model used in a flight simulator must behave as much as possible like the true tokamak. Inputs to this model should only consist of command waveforms to actuators, and outputs to the PCS are solely the measured signals simulated by synthetic diagnostics.
- Pulse schedule design: this entails using models to prepare a plasma scenario, i.e. compute a time-dependent evolution of controller reference and feedforward signals to obtain the plasma behavior desired, within the constraints of the actuators. Examples of this in axisymmetric control are inverse equilibrium codes such as [660] to compute PF coil currents for a desired magnetic equilibrium, and more recently [661–663].
- Scenario design: this deals with determining the optimal scenario to obtain certain experimental goals. This is distinct from Pulse schedule design in the sense that the latter assumes the target scenario (the desired plasma evolution) is given, while the scenario optimization problem deals with designing an optimal scenario. There are obvious overlaps as the scenario optimization problem also needs to take actuator and physics constraints into account, and a sufficiently detailed scenario design workflow could, in principle, directly output a pulse schedule. Scenario optimization has been approached using control-oriented core transport models as part of automated (numerical) optimization routines, for example [664–666].
- Observers: as mentioned in [140] in ITER the task of estimating the to-be-controlled plasma state from diagnostic carried out in a so-called input support function, separate from the control algorithms. For plasma state estimation, so-called dynamic state observers have been proposed. These combine predictive (forward) models of the to-be-controlled plasma dynamics diagnostic data to form the best possible estimate of the plasma state with given all available information. A core element of an observer scheme is a one-step-ahead prediction of the to-be-observed system. This prediction step must run faster-than-real-time in order to be complete when the next diagnostic data sample becomes available. This poses a challenge to control-oriented models, therefore it so far only been possible to deploy observers based on highly simplified models. Examples include Kalman Filtering for vessel current estimation [667, 668], real-time density observers based on particle inventory +1D core transport models [669] and q -profile and/or electron temperature estimation [670, 671].
- Forecasting/prediction: an even more challenging application is that of real-time forecasting and prediction. In this case, a model runs several times faster than real-time to compute the expected time trajectories of the plasma evolution under a number of scenarios, including the nominal case and possibly several alternative cases. A recent example of this

application is the real-time prediction of the NTM evolution, including a prediction of the power required to stabilize it [672].

8.3.3. Control-oriented models for different simulation domains. We conclude this section by providing an overview of some examples of existing control-oriented models in the fusion community and their application, illustrating the way these models can be used for ITER. Some of these examples are discussed in more detail in later sections.

- Regarding control-oriented models for axisymmetric control simulation, including here only models with a simplified description of core plasma evolution, we can distinguish linearized models with both deformable and rigid plasma assumptions, as well as fully non-linear free-boundary evolution models. Examples include CREATE-L/NL [673], FEEQS [674], GSolve [675], NICE [676], MECS [677]. Applications include magnetic control development and testing for ITER [678], WEST [679], and JT60-SA control [680]. Electromagnetic breakdown/burn-through simulators such as DYON [36], which combine electromagnetic model for the current-carrying structures with a 0D model for the early plasma phase, or the simpler model used for model-based breakdown optimization in [681] lie in a similar category.
- Control-oriented models for core plasma transport are typically able to solve the coupled evolution of the core plasma current density and several kinetic profiles (i.e. temperatures, densities of various species), using a variety of transport and source/sink models. Several simplifying boundary condition assumptions are adopted to couple the plasma core with the edge, SOL and divertor, as well as to model the dynamics in the pedestal region. Contrarily to models used primarily for integrated physics simulations, control-oriented models typically have links to optimization routines, features to provide local linearizations around an operating point and/or are capable of running in real-time. Examples include RAPTOR/RAPDENS [665, 669, 682], COTSIM [683], METIS [340]. RAPTOR and COTSIM in particular have been applied to state observers [669, 671, 684], iterative learning control [685], non-linear optimization of q -profiles and scalar core quantities [665, 666, 686], controller design using MPC [686, 687], optimal control techniques [688], or non-linear feedback control methods [689–691].
- Integrated tokamak simulators for control applications can be used to test individual control schemes in closed-loop, as well as testing the entire PCS. For example, the TRANSP code has been used to test control schemes by deriving linearized control models from simulations, designing a controller, and then testing it on full non-linear simulations for non-inductive operation of NSTX-U [692]. TRANSP has also been used for scenario optimization [693]. Another example is FENIX [659, 694] which is a flight simulator coupling a full PCS with a tokamak simulator based on

ASTRA and SPIDER. Other examples of integrated modeling suites used for scenario design and control include CRONOS (used for profile control simulations in [695]), JINTRAC (used in [189] for ITER fueling & divertor power flux control), and METIS+FEEQS (used for scenario design in [696]).

To enable the rapid execution of control-oriented models while retaining sufficient physics fidelity, significant effort has also gone to development of fast modules for key physics effects that have been identified as bottlenecks for simulation. These models either use a specific formulation of the physics equations that still retains sufficient accuracy while being sufficiently fast, or alternatively relies on ML surrogates to emulate the input-output behavior of a more computationally intensive code. Examples of the former include TORBEAM [697] for EC absorption, RABBIT for NBI [698]. Examples of the latter include NUBEAM-NN for NBI heating [699], QLKNN [700] and MMM [701] for turbulent core transport, and TGLF-NN and EPED1-NN for combined core-pedestal simulation, which are used together in [702]. More recently, fast models that integrate the plasma SOL have been developed as well, for example DIV-1D, IMEP [703]. As is often the case, these models are initially developed to accelerate integrated modeling, and then integrated in control-oriented simulators for control applications. Altogether, acceleration of physics models for scenario development & control simulations is an emerging and rapidly growing field that continues to attract significant attention, also in synergy with computational sciences and ML.

8.4. Key components of scenario control

8.4.1. Profile control. A fundamental control problem arising in tokamaks is the regulation of several properties of the core plasma such as toroidal current, density, temperature, and rotation since these properties are strongly linked to advanced modes of operation [648]. These advanced scenarios are characterized by a high confinement state with enhanced MHD stability, which yields strong improvements in plasma performance (see section 6). Such improvements are quantified by increases in energy confinement time, plasma pressure, and fusion power density. Moreover, in these advanced modes of operation a dominant fraction of the plasma current is self-generated by the neoclassical bootstrap mechanism [704], which reduces the requirement on externally driven non-inductive current for steady-state operation. The dynamics of these core plasma properties depend on both time and space, and are governed by non-linearly coupled PDEs. Therefore, the realization and sustainment of advanced scenarios rely heavily on the active control of the plasma spatial profiles, i.e. the spatial distribution of the plasma properties from the magnetic axis to the boundary. Due to actuation limitations in tokamaks, the profile control expectation for a given plasma variable may need to be reduced to its control at a finite number of spatial locations. Furthermore, in some cases only the control of a plasma variable at a single point or integrated over the plasma volume may be feasible. On the positive side,

the combination of profile and scalar control may be enough for achieving advanced operation. Due to the high dimensionality (infinite-dimensional PDE system) and non-linearity exhibited by the plasma profile dynamics in burning plasmas, its control arises as one of the most challenging control problems in tokamaks.

Several aspects of this advanced scenario control problem must be explicitly taken into account during the control-design process. First, the strong coupling between magnetic and kinetic profiles, which are usually characterized by very different time scales. Second, the relationship of both magnetic and kinetic profiles with MHD stability. Third, the strong interlacement of control objectives since actuators affect more than one plasma variable. More than one decade ago now, the plasma control community identified a model-based control design approach as the only possible way of tackling the complexity of this challenging problem. Therefore, a significant component of the profile control problem is the development of plasma-response models suitable for control design. This demands the modeling of diffusivity and source terms in the non-linear transport PDE's that are accurate enough to correctly predict the plasma evolution in response to the different actuators but simple enough to keep the control design problem tractable.

8.4.1.1. Current profile control. Shaping the current profile, usually defined in terms of the safety factor q - or rotational transform ι profiles, has been demonstrated to be a key condition for realization of advanced plasma scenarios characterized by MHD stability, improved confinement, and possible steady-state operation [705]. The achievement of some types of current profiles favors generation of ITB [706], which improves confinement and facilitates steady-state operation by enhancing bootstrap current. It has been demonstrated that current profile control can slow down, and possibly stop, temporal evolution of current profile peaking [707], which tends to develop due to the fact that the plasma temperature is higher in the core and plasma resistivity is inversely related to temperature. Reduction of this peaking effect also allows avoidance of some instabilities or disruptive events related to presence in the plasma of magnetic flux surfaces where q is a rational number, such as NTMs.

Initial efforts towards current profile control focused on demonstrating real-time feedback control of scalar parameters characterizing the current profile by using non-model-based, SISO, control laws. In [708], the internal inductance parameter, l_i , a measure of the current profile shape, was controlled using LHCD on Tore Supra, while in [709], the width of the LH power deposition profile was regulated. In [710], feedback control of either $q(0, t)$, the safety factor at the magnetic axis of the plasma, or $q_{\min}(t)$, the minimum value of the safety factor profile, was achieved on DIII-D by modifying ECH or NBI. Additional efforts based on non-model-based, real-time control of the current profile include the work carried out at JT-60U [711, 712]. Performance limitations of this type of control solutions arising from the complexity of the problem described above demanded, however, a model-based control approach to

accommodate this complexity by embedding the known physics within the design. By capturing the response of the system to the available actuators in a control-oriented model, such control designs can achieve improved performance without the need for extensive trial-and-error tuning. Control-oriented models of the current profile dynamics can be divided into two groups; those created entirely from experimental data and those that are driven by plasma physics.

The evolution in time of the toroidal current profile in tokamaks is related to the evolution of the poloidal magnetic flux profile, which is modeled by using a first-principles, non-linear, dynamic PDE referred to as the magnetic flux diffusion equation. The magnetic flux diffusion equation is derived from Gauss's law, Ampere's law, Faraday's law, Ohm's law, and an equilibrium balance. The dynamics of the poloidal magnetic flux can be modified through three different mechanisms; (i) the plasma resistivity η (*diffusivity control*), which can be 'actuated' by controlling the electron temperature profile by using different RF heating mechanisms such as ECH; (ii) the non-inductive sources such as NBI or RFCD, which may include ECCD, FW CD, LHCD, etc; (iii) the plasma current I_p (*boundary control*), which can be 'actuated' by controlling the inductive component of the plasma current by transformer action. Data-driven modeling techniques based on linear system identification have been successfully used to model the plasma transport dynamics both from experimental data and synthetic data generated by physics-based predictive codes. This modeling approach, which has the potential to obtain low-complexity dynamic models, lacks however the ability of arbitrarily deciding on the level of simplicity, accuracy and validity of the model since it directly produces linear models based on ODEs. As the identified models are linear, they are only valid around the reference plasma state adopted during the system identification experiment. Therefore, the effectiveness of the controllers synthesized based on these models may be limited when the plasma state moves away from the reference state. Moreover, as these models are device-specific, dedicated system identification experiments are needed in each device, and potentially for each control scenario, to develop model-based controllers. Nevertheless, active profile controllers synthesized from these models have shown a great deal of effectiveness. By using input/output diagnostic data, the current profile dynamic model has been identified for AUG [713, 714]. At JET, a two-time-scale linear model has been identified from data to describe the dynamic behavior of the magnetic and kinetic profiles around certain quasi-steady-state trajectories [715], and a feedback controller has been designed based on two loops operating on the magnetic and kinetic time scales respectively [716]. Dedicated experiments have also been carried out at DIII-D to generate input/output diagnostic data for identification of the current profile dynamics [655]. These models enabled the design of several current profile controllers [717–720]. Physics-driven modeling techniques combine the magnetic flux diffusion equation with control-oriented models for the electron temperature, the plasma resistivity, and the non-inductive CDs. The control-oriented models employed to close the magnetic flux diffusion equation model, either in the form of empirical correlations

or analytical laws, are based on physical observations of the plasma response to the control actuators, which are not unique to any one machine, e.g. the observations of increasing plasma confinement with increasing plasma current and of decreasing plasma confinement with increasing total injected power. Therefore, physics-driven models are more easily adaptable than data-driven models to different tokamaks because the model can be closed by leveraging already existing experimental data from the tokamak of interest. Moreover, the plasma-response-model structure for various tokamaks operating in different equilibrium configurations remains unmodified. This approach produces a strongly non-linear PDE model describing the evolution of the poloidal magnetic flux profile in response to the electric field due to induction, the auxiliary heating and CD systems, and the plasma density [670, 721–723]. The complexity of this physics-driven model needs to be reduced to facilitate design of compact and reliable control strategies. During this control-oriented model reduction process, there is always a trade-off between the simplicity of the model and both its physics accuracy and its range of validity. Physics-driven modeling provides the freedom of arbitrarily handling this trade-off and deciding on the level of simplicity, accuracy and validity of the model depending on the control design objective. This leads to a plethora of current profile control designs as it will be discussed below.

The infinite dimensionality associated with the physics-driven model (PDE model) has been tackled in two different ways. On the one hand, the DTA approach synthesizes an infinite-dimensional controller based on the PDE model and then introduces an approximation to reduce the dimensionality and obtain a finite-dimensional controller that is useful for practical implementation. However, a controller based on the DTA approach is much more complex to synthesize in most practical scenarios due to the infinite-dimensional nature of the underlying model. Nevertheless, linear control algorithms based on the DTA approach have been developed and successfully implemented for safety factor regulation in tokamaks [690, 724]. Work on current profile regulation avoiding the linearization of the model includes sliding-mode [725] and Lyapunov-based [726] control algorithms. On the other hand, the ATD approach reduces the PDE model to a set of ODE's before synthesizing a finite-dimensional controller. Most of the work on model-based current profile control has followed this approach and the rest of the discussion in this section will focus on this approach. The performance of the finite-dimensional controller, both in the DTA and ATD approaches, when implemented to regulate the original infinite-dimensional plant, depends on the model approximation scheme used and is hard to quantify mathematically in advance.

With the exception of a few pieces of work such as the non-linear robust controller proposed in [727] by using feedback linearization and Lyapunov techniques, most of the controllers designed by following an ATD approach rely on linearized models. In these cases, the linear feedback controllers are complemented by non-linear feedforward controllers [664, 665, 693, 728–731]. Linear current profile feedback controllers

have been designed based on a variety of techniques, including robust control [686, 729, 732–734], optimal control [688, 735–739], backstepping control [691], passivity-based control [740], and MPC [741–745]. Among these techniques, only MPC can take into account the saturation and rate limits of the actuators *a priori*. All the other types of controllers can, however, be potentially augmented by anti-windup compensators to cope with these saturation and rate limits *a posteriori*. Feedback linearization controllers [727] can also handle actuator limits during the control synthesis. However, both MPC and feedback linearization controllers rely on real-time optimization, which could be too computationally expensive for some present devices. As an alternative, Lyapunov-based techniques have also been proposed to handle the nonsymmetric actuator saturation of the individual CDs [746].

Most of the work cited so far pursues a global control objective, where the whole profile is regulated either by controlling the profile value at several spatial locations or by minimizing the tracking error of the whole profile integrated over the spatial domain. Local control algorithms, on the other hand, are designed to regulate the scalar properties of the profile at fixed locations, such as the safety factor at the magnetic axis $q(0)$ [747], or at moving locations, such as minimum safety factor $q_{\min}$ [748, 749] or the safety factor gradient around a predefined rational surface [750]. The control of local properties of the current profile at moving locations benefits from the use of the actuators with the capability of varying the spatial location of the current deposition in real-time such as the ECCD. Current profile controllers exploiting this additional actuation capability have been recently proposed [745, 750].

The H&CD sources used for current profile control also affect the temperature profile, and therefore the pressure profile, with the risk of triggering MHD instabilities or undesirable L–H/H–L transitions. Therefore, it is straightforward to conclude that an approach to current profile control that does not integrate some level of kinetic control will not be practical. In addition to using the temperature and density as mechanisms to regulate the current profile, these plasma properties must be closely monitored and regulated, if not tightly, at least within a region that prevents the triggering of these undesirable events. Recognizing these constraints, most of the proposed current profile control solutions simultaneously regulate the normalized β or the plasma internal energy to prevent the controller from triggering MHD instabilities or L–H/H–L transitions in its aggressive effort to achieve the desired target profile. Figure 46 shows the performance of active control in DIII-D shot #163832. The goal was to drive both q and thermal stored energy W (defined as a volume-integrated quantity below), from a given initial state at $t_i = 400$ ms (end of plasma formation phase) to a target final state at $t_f = 3000$ ms (start of physics-study flat-top phase) by controlling individual on-axis and off-axis NBI powers, total EC power, and total plasma current. This target final state is characterized by a minimum value of q , denoted as $q_{\min}$, of around 1.85, and a value of q at $\bar{\rho} = 0.95$, denoted as q_{95} , of around 5.5. A primary goal for the DIII-D research program is to develop the physics basis for a high q ($q_{\min} > 1.5$ –2.0), high W , steady-state operation mode

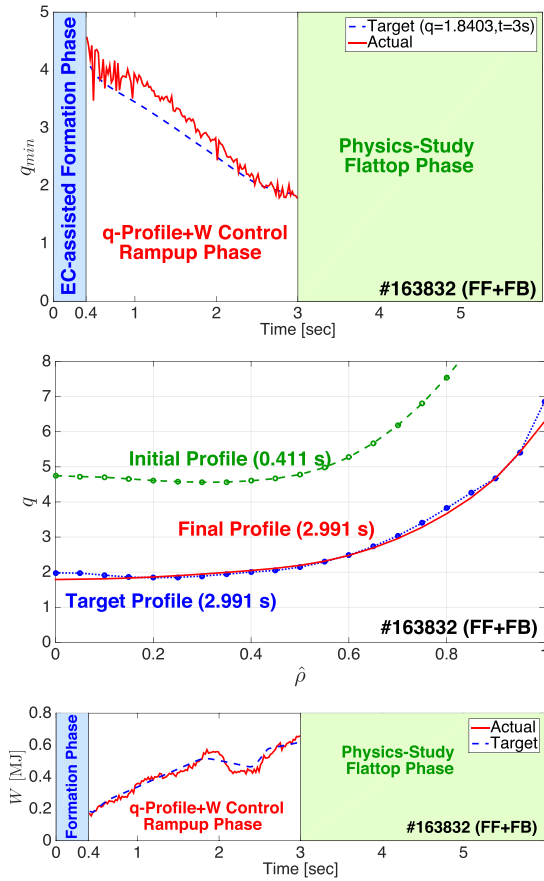


Figure 46. Experimental test of feedforward + feedback q -profile controller based on off-line+on-line model-based optimization. Target profile ($q_{\min} = 1.85$) achieved at $t_f = 3000$ ms while tracking a target W [648].

that can serve as the basis for future steady-state burning plasmas. The tight regulation of both q and W is made possible by combining off-line (feedforward control) [664, 665, 693, 728–731] and on-line (feedback control) [741–745] model-based optimization (i.e. MPC). At the core of both optimizations are physics-based, control-oriented models of different complexities (off-line optimization: Non-linear model, on-line optimization: Linearized model) capturing the dominant physics of q -profile and W responses to actuators.

8.4.1.2. Rotation profile control. Control of the toroidal rotation profile has recently attracted much attention because of its connection with MHD stabilization and advanced mode access. As an alternative to active stabilization of RWMs by magnetic feedback control (chapter 4), experiments have shown that RWM passive stabilization is also possible by plasma toroidal rotation coupled with an energy dissipation mechanism. It has been shown that the critical rotation speed for RWM stabilization is a function of the rotation profile shape, implying a radially distributed stabilizing mechanism. RWM stability therefore depends on both β (a measure of plasma pressure, defined below) and toroidal angular velocity profile. Unfortunately, RWM stabilizing rotation is often dissipated by drag torques generated by different mechanisms,

including plasma fluid viscosity, and interaction between the plasma fluid with magnetic field perturbations. Modification of the rotation profile through active control [751–754], in order to affect energy confinement, can improve stability of not only RWMs but also NTMs (chapter 4). Other applications that might call for rotation profile control include; (i) modification of the tearing mode rotation frequency to facilitate ECCD stabilization of the NTM as it rotates toroidally in the plasma, (ii) establishment of conditions favorable for access to advanced operation modes (e.g. free of certain MHD instabilities). NBI, and non-axisymmetric magnetic field coils (the same coils that are used for RWM feedback stabilization) that allow modulation of the error fields for magnetic braking (controlled drag torque), provide torque sources for control of the rotation profile in tokamaks. It is expected, however, that the sources of torque may be too limited in tokamaks of the size of ITER for effective rotation profile control.

8.4.1.3. Density, temperature, and pressure profile control.

The control of density, temperature, and pressure profiles in tokamak plasmas is challenging because of limited actuation capabilities and a relatively poor understanding of particle transport phenomena. In principle, these profiles can be controlled by fueling mechanisms such as puffing of gas and injection of frozen-fuel pellets and by heating mechanisms such as NBI and different sources of RF waves. It is believed that the gradient of the pressure profile around magnetic flux surfaces where q is a rational number can play a critical role in triggering certain types of MHD instabilities chapter 4. Moreover, certain density and temperature profiles favor the self-generation of the non-inductive bootstrap current [704], which favors steady-state operation.

Plasma density regulation in present tokamaks is routinely achieved by gas puffing. Relatively simple PID controllers actuate gas valves based on the deviation between requested and measured plasma density. Nevertheless, fueling the plasma core by gas puffing will not be suitable in future reactor-grade tokamaks like ITER because of the large time-delays introduced when actuating the gas valves due to the machine size (large distance between valves and vessel) and the inability of the neutral fueling gas to penetrate the plasma region due to higher plasma temperatures and densities in the core. Fortunately, pellet injection is expected to be the primary fueling system for reactor-grade tokamaks as discussed in section 7. However, this fueling approach, which is largely unexplored and not routinely used in present devices, will offer its unique challenges, such as the effects of the pellet size and velocity on the spatiotemporal dynamics of the plasma, the discrete-time nature of the actuation, the ablation and penetration dependence on the plasma conditions, and the possibility of pellet drops. Moreover, non-linear effects arising from both plasma and actuator dynamics will be even more severe and the control requirements will be much more demanding than in present tokamaks, since effective density regulation, including the isotope-ratio regulation, will be critical for burn control. It is also expected that density control, particularly through gas puffing, will offer higher coupling with other objectives

in reactor operation such as detachment control and ICRF-plasma coupling (see section 7).

Traditionally, 0D multi-reservoir models [755–761] have been used to develop different types of control approaches for density regulation via gas puffing in experimental tokamaks. More recently, driven by the need of controlling the density in ITER by using actuation mechanisms that fuel the plasma at different spatial regions (gas puffing at the edge and pellet injection in the core), control-oriented models have been proposed to capture the 1D dynamics of the core ‘reservoir’ [682, 762] and to develop control and estimation solutions [685, 763, 764] for density regulation in ITER. This effort runs in parallel to a significant research thrust within the fusion community to develop integrated, physics-oriented, predictive models to study the dynamics of the density profile in response to both gas puffing and pellet injection [246, 602, 765] since there is still an urgent need to convert these accepted physics models to a form useful for control synthesis and simulation. This modeling work will enable the development of control strategies integrating gas puffing and pellet injection, while studying the effect of both pumping and strike point evolution and assessing their potential use as actuators to assist active (core) density and (edge) impurity control for different wall recycling conditions. These control strategies must take into account the different actuation time scales (slow (gas) versus fast (pellet)) and incorporate realistic actuator limitations and constraints such as response times, time delays, bandwidths, non-linear efficiency, discrete/continuous time nature, saturation levels, and rate limits. The neglect of relevant reactor-grade actuator dynamics is indeed a deficit of present density control solutions which must be tackled with urgency.

Although previous work on temperature profile control is more limited, it is still possible to highlight previous efforts towards the development of active control strategies [695, 766] with the ultimate goal of regulating simultaneously the electron temperature and safety factor profiles.

8.4.2. Density and β control. Control of the plasma density or β^{25} have often been considered separately from each other in present devices and, therefore, also from emulated burn control schemes (section 8.4.3). Whereas density is usually controlled in feedback by fueling actuators such as gas valves, pellet injectors, or cryopumps, beta control is normally achieved by using auxiliary heating sources. However, both problems are highly interlinked. Just to name a few examples of this relationship, we have the evident influence of the density dynamics on the beta evolution, the impact of the heating-induced L–H transition on the density pedestal, or the dependence in the L–H power threshold with the overall electron density. In addition, other plasma parameters primarily controlled by the tokamak coils, such as shape or total current, can be used as actuation methods to modify both the plasma density and beta simultaneously. Despite the evident coupling between the two

problems, most previous pieces of work make a clear distinction and focus on one of them.

The vast majority of the present tokamaks operating in the world have some kind of feedback control of the plasma density. This is motivated by the goal of attaining and maintaining the required densities for the reproduction of the plasma scenarios of interest for physics studies. Historically, gas valves and cryopumps were the first actuators employed to regulate the plasma density and, therefore, control algorithms for gas puffing are the most extended [757, 767–770]. Gas puffing injects neutrals at the plasma edge, of which a percentage get ionized and/or may make it into the plasma core. With the increase in size of the subsequent tokamak designs and the development of H-mode scenarios, the low fueling efficiency achieved with gas puffing at the plasma edge motivated the use of pellet injection for density control [771–774]. Pellets injected at high velocity have a significantly higher capability to cross transport barriers and penetrate into the plasma core than conventional gas puffing. In ITER, pellet injection is foreseeing as the main fueling technique for the core, specially in H-mode and/or burning plasma regimes (see section 7). Other more exotic approaches to density control include, for instance, the use of 3D fields [775].

Beta control under feedback has played a vital role in the development of plasma scenarios in present devices, some of which are candidates for ITER operation. Some examples are the development of advanced scenarios in DIII-D [776] and EAST [409], long-pulse operation in KSTAR [777], or beta overshooting avoidance in JET to prepare for DT operation [778]. Other work includes low aspect-ratio plasmas in NSTX [779] or I-mode development in AUG [780]). In addition, novel approaches to β_N feedback have lately been explored, such as the use of 3D fields to recover thermal quenches in DIII-D [775] or RL to design feedforward beta control in KSTAR [294, 295].

8.4.3. Core kinetic control (burn control). The regulation of the amount of fusion power produced by future reactors will require precise control over the plasma density and temperature. Therefore, the control of the core plasma kinetic state, usually referred to as burn control, arises as one of the most fundamental problems in nuclear fusion and will be critical to the success of burning plasma devices like ITER [781]. Due to the non-linear coupled dynamics of the plasma, feedback control of the burn condition will be necessary to avoid undesirable transient performance, to respond to changes in plasma confinement, impurity content, or operation conditions, which could significantly alter the plasma burn, and to avoid potentially disruptive plasma conditions due to thermal instabilities. When the control objective is not only to regulate the plasma around a desired fusion power but also to have the capability of determining the operating point in terms of plasma energy, density, and tritium fraction that provides the desired fusion power, a model-based control approach is needed to take into account the high dimensionality and the coupled dynamics of the burning plasma. This control capability will be absolutely necessary in fusion reactors in order to keep the operating

²⁵ The plasma beta serves as a proxy for the plasma thermal energy or pressure.

point far from stability and safety boundaries in order to avoid disruptions and protect the device while regulating the fusion power. Since the primary goal of a burn controller is usually to regulate volume-averaged properties of the plasma, 0D models are appropriate for control synthesis. However, if the goals were different for different regions of the plasma (e.g. core versus edge), dynamic models capturing the spatial dependence will be needed for control synthesis, such as fully 1D models or multi-region 0D models.

Over the years, the physical and technological feasibility of different methods for controlling the burn condition have been studied. Pioneering studies mainly focused on three different types of actuation; modulation of auxiliary power, modulation of fueling rate, and controlled injection of impurities [782–784]. The controllers based exclusively on the modulation of the auxiliary power [785–787] require operation at subignition points where the auxiliary power is nonzero. As the plasma heats up due to a positive perturbation in the temperature, the auxiliary power is reduced by the controller. Similarly, as the plasma cools down due to a negative perturbation in the temperature, the auxiliary power is increased by the controller. Since the maximum reduction is complete shutoff of the auxiliary power and the maximum increase is limited by the available installed auxiliary power, there is a limited range of thermal excursions and quenches that can be effectively rejected by the control system. Moreover, for devices operating at high fusion gain Q , and therefore at high fusion power, the amount of auxiliary heating may be quite small compared to the total plasma heating, such that modifying the auxiliary power may have a limited effect on the plasma temperature. In advanced scenarios, the control authority may be further restricted by the need to use some auxiliary power sources to drive plasma current. This essentially forces the burn control system to respect a nonzero minimum auxiliary heating power constraint. Therefore, operating points characterized by auxiliary heating power requirements very close to either the minimum required for CD or the maximum installed in the device will be difficult to stabilize and control with auxiliary power regulation alone. The controllers based on the modulation of the fueling rate [788–791] allow operation at ignition points where the auxiliary power is zero (or near the minimum auxiliary power required for CD). However, although they can deal quite well with perturbations leading to thermal excursions, they are not very effective for perturbations leading to quenching. Due to disruptive density limits, the plasma density cannot be increased arbitrarily high. The approach is also limited by the decay rate of the density, which can be quite slow when particle recycling rates are high. Controlled introduction of impurities is useful to enhance the radiation losses and in this way cool down the plasma and prevent thermal excursions. For large positive perturbations in the temperature this method requires the introduction of a large amount of impurities. Therefore, after controlling the thermal excursion, additional amount of auxiliary power, with the consequent Q reduction, must be provided in order to compensate the radiation losses due to the impurities until they are completely removed from the reactor. Prior work that combines actuation

concepts by using linear-control techniques includes solutions based on 0D [792–797] and 1D [798–800] models.

The isotopic fuel mix, which is a critical reactor parameter as it has a major influence on the fusion power produced, was more recently suggested as a potential actuation mechanism. The α particle heating power, P_α , from a burning DT plasma is given as

$$P_\alpha = Q_\alpha \gamma (1 - \gamma) n_{\text{DT}}^2 \langle \sigma v \rangle \quad (29)$$

where Q_α is the energy of the α particles produced by the reaction (3.52 MeV), $n_{\text{DT}} = n_{\text{D}} + n_{\text{T}}$ is the DT density, $\langle \sigma v \rangle$ is the reactivity, and $\gamma = n_{\text{T}}/n_{\text{DT}}$ is the tritium ratio, a measure of the isotopic mix. For a constant density and temperature, the fusion heating depends quadratically on the tritium ratio, with the maximum fusion power at $\gamma = 0.5$. Differences in deuterium and tritium transport [801] and fueling efficiency, as well as perturbations introduced by other sources of particles not under the burn controller such as NBIs, may lead to a non-optimal fuel mix in the core even with an optimal 50:50 DT injection, which would reduce reactor efficiency. Additionally, depending on the operating scenario, it may be desirable or even necessary to operate at a lower tritium fraction or modulate the tritium fraction during operation. To control the tritium ratio it is possible to use a method of fueling referred to as isotopic tailoring, in which the relative mix of deuterium and tritium injected by the fueling system is modulated in real-time [802]. The PIS for ITER will include two separate injectors; one with pellets made of primarily deuterium and the other with pellets made primarily of tritium. The GIS will be used to supply deuterium at the edge of the plasma. Together, these systems will allow for fuel mix modulation. More recently, the in-vessel coil system has demonstrated its capability to control the plasma energy in DIII-D [803] by generating non-axisymmetric fields that modify the plasma energy confinement time, τ_E^* .

A well-designed burn controller should be able not only to achieve tight regulation around a desired operating point by rejecting perturbations in temperature and density but also to drive the plasma from one operating point to another during the burning plasma mode (e.g. different Q or fusion power), to access to and exit from the burning plasma mode, and to handle the non-linear coupling with other competing controllers using shared actuators. These control objectives demand not to neglect the non-linear dynamics of the plasma during the model-based control synthesis process. In order to overcome the operability limits imposed by the linearization of the burn dynamics, non-linear techniques for burn control have been proposed to account for the non-local character of the dynamics. Model-based control designs incorporating the non-linear dynamics of the plasma have shown higher levels of performance, stability, and robustness against model uncertainties [804, 805]. These controllers utilize several actuators simultaneously, using auxiliary power modulation to prevent quenching, impurity injection to stop thermal excursions by increasing radiation losses, and fueling modulation to regulate the density. The relative mix of tritium fuel in the plasma

was exploited more recently as a virtual actuator to cool the plasma during thermal excursions [806]. In this way, impurity injection, and the consequent reduction in fusion power because of enhanced radiation, needs to be used only in cases where the control of the relative content of tritium in the core is limited by severe particle recycling conditions at the wall, which is not what is expected for ITER. The in-vessel coils can be used in combination with isotopic fuel tailoring [807] as an additional actuation mechanism for reduction of the plasma stored energy in case of thermal excursions. It is important to highlight that the lack of an isotopic fuel tailoring capability in future fusion reactors will not prevent a well-designed burn controller from regulating the fusion power (note that control designs such as those proposed in [804, 805] do not exploit isotopic fuel tailoring) but it may limit the capability of the controller in terms of simultaneous regulation of other important plasma properties defining the operation point within the operation space due to the loss of actuation capability.

The robustness and adaptiveness of the proposed burn controllers have been pursued and demonstrated. First, simulation studies have been carried out to demonstrate control performance of the proposed non-linear controllers under confinement mode transitions in ITER [808]. Second, the impact of potential fueling limitations on the burn condition for different confinement time and particle recycling assumptions has been studied. Unmeasurable variations over time (biases and drifts) in the tritium concentration of the fueling lines could have an important impact on reactor operation. Limitations in the tritium plant subsystems could result in a lower tritium concentration. Even if the nominal concentration could be initially achieved, it might not be possible to sustain it for the total duration of long-pulses. This not only imposes burn control challenges but also raises concerns over having the fueling capability to sustain long-pulse high- Q operation in ITER. While methods have been successfully proposed to robustify the designed burn controllers against these variations [809], the operational space can significantly shrink. The shrinkage of the operational space, illustrated in terms of plasma operation contour (POPCON) plots in [810], cannot be overcome by any control design and is fundamentally an actuation-related constraint. In other words, while robust controllers can still effectively regulate operation within the operational space, they cannot do anything against the shrinkage of the operation space itself. Limitations imposed by the tritium concentration in the fueling lines together with actuator constraints and operational limits imposed by safe divertor operation [811] will at the end dictate the space where the reactor can operate. Third, to ensure asymptotic tracking of the target operating point despite uncertainties in the burning plasma model used for control design, the proposed non-linear controllers have been augmented with adaptive control schemes designed to improve the model used for control design in real-time [812, 813]. Finally, though the non-linear adaptive control scheme proposed in this work can drive the burn condition to the desired reference (defined by the energy, density, and tritium fraction), the model uncertainty may lead to differences between the resulting outputs, like fusion power or divertor heat load, and those

predicted by the model. For example, differences in impurity or alpha particle confinement could alter radiation losses and DT fuel content in the plasma at the reference values of energy, density, and tritium fraction. With large enough parameter changes, the outputs for a given reference could differ significantly from the predicted values, making it difficult to choose the controller references prior to a discharge. To overcome this issue, an online burn condition optimization scheme has been proposed to alter the controller references in real-time in order to minimize a cost function related to the desired reactor performance [812]. This cost function can be chosen to weight a variety of possible outputs (either measured or predicted from models).

Modeling of specific dynamics and constraints of both actuators and diagnostics in fusion reactors like ITER, as well as the design of control approaches to cope with them, are essential for any practical burn-control solution. Modeling of the actuator dynamics and constraints enables assessment of the capability of the fueling and HSs to regulate the burn condition with short enough delays both when the fusion power needs to be increased and when the fusion power needs to be reduced. In particular, the development of more realistic models for gas valves and pellet injectors could also lead to more efficient density control algorithms, which are indeed needed well before reaching the burning plasma (DT) phase in ITER. An actuator-allocation approach has been proposed to address this challenge by mapping in real-time the request by the burn controller in terms of heating and fueling to the reactor's actuators while taking into account actuator dynamics/constraints and time-varying model uncertainties in the mapping [814–816]. In future fusion reactors, diagnostic systems may be very limited due to the extreme conditions inside the devices, as well as cost considerations. Because the proposed non-linear burn control approach relies on knowledge of the state of the system (i.e. it is a state feedback control scheme), it may not be directly implementable with the available diagnostics. To overcome this obstacle, an observer, combines knowledge from the mathematical model of the system with the available real-time output measurements to estimate the system states, will likely be necessary in practice. Observer design for the burning plasma system is complicated by the presence of modeling uncertainties, the non-linearity of the state dynamics, and the fact that the output map is, in general, non-linearly dependent on the states of the system. This problem has been tackled by employing proportional-integral observation concepts [817].

8.4.4. Control aspects of plasma initiation. A tokamak discharge starts with a process that ionizes a gas to form a tokamak plasma. This is a complex process that requires the tokamak to induce initial ionization (breakdown) by applying a toroidal electric field, and to generate a toroidal plasma current, forming a tokamak discharge (see section 2). For the plasma current to be sustained the plasma temperature needs to increase, such that the plasma resistivity decreases throughout the process. The latter means that the plasma heating exceeds the losses, especially those due to radiation (i.e. it achieves

burn-through). This process can be supported by applying waves, either assisting the ionization process or by providing additional heating to benefit the power balance [31, 91] as discussed in section 2. An underestimated factor in the plasma initiation process is the control aspects required to achieve it. Failure of the plasma initiation process is often attributed to breakdown or burn-through problems, however, control errors can also be an important factor [91]. The plasma initiation phase also should initiate several important control functions, firstly radial and vertical position control, followed by plasma current control, density control, and eventually control of the plasma shape and VS control. This initiation of control can either simply be timed by the pulse schedule or better event triggered. With all these control functions active the plasma initiation phase can transition to the current ramp-up phase, allowing the current and density to be increased to its flat-top value.

For superconducting devices, plasma initiation has to be done at lower toroidal electric fields, and in the case of ITER, will have to be achieved in a very large vessel [91] as discussed in section 2. This makes the whole process more fragile and puts a great emphasis on optimizing the control. The control scheme should allow flexible adjustment of the timing of all aspects of the plasma initiation process. Thus the timing of the application of the radial electric field, the gas fueling, the application of waves, and many other features, but also when certain control functions start to act. The initiation of the control functions should assist the burn-through and current rise, and not counter-act it. For example, radial position control should not lead to a reduction in plasma volume, such that it compromises the burn-through. Or the density control should provide by both feedforward and feedback action with exactly the right amount of fueling that avoids on the one hand the density decreasing after breakdown, possibly leading to the generation of start-up runaway electrons, or too high a density that leads to a failed burn-through or one may hit the density limit [115, 818]. A final point would be the preparation of exception handling, thus how the control system responds under exception events. Handling policies should be carefully chosen [140]. Switching of actuators, including gas fueling in case of any alarms, may end the control of the density, and allowing a ramp-down of the plasma current increases the electric field, both actions that again can enhance the generation of runaway electron generation. Assessment of the proposed ITER control schemes (i.e. controllers, exceptions and their configuration for plasma initiation) by simulation or experimental tests is crucial to optimally prepare for ITER's First Plasma Operation.

8.4.5. Control aspects of discharge termination. The controlled termination (or 'shutdown') phase of a tokamak discharge is especially dynamic, including the risk of approaching a variety of controllability and stability limits in parallel, and often facing competing or even contradicting requirements. An international research effort has been going on in recent years with a special look at ITER, including simulations [359, 649, 819, 820], experiments on multiple

devices such as JET [649, 821–825], DIII-D [826–828], TCV [829], AUG [829, 830], C-Mod [226], KSTAR [831], and cross-device, multi-machine studies [649]. In addition, optimized ramp-down scenarios for a DEMO burning plasma have been modeled [832]. Several major control challenges during termination have been identified by the community and are discussed here.

8.4.5.1. Flux consumption and coil limits. Minimizing the flux consumption in ramp-down inherently means more inductive flux available to support the ramp-up and flat-top phases. This is aided by faster plasma current ramp-down and staying in H-mode for longer periods in the ramp-down, but both of these have other implications for control. Several studies have estimated the flux consumption used during ramp-downs associated with scaled ITER-like scenarios to determine constraints on the I_p ramp-down rates [59, 827, 833], and simulation studies have been done to ascertain the range of I_p ramp-down rates that ITER will be capable of handling in different conditions [19, 820]. Under normal, controlled conditions, the fastest possible ramp-down phase on ITER is expected to take at least 50–60 s [19].

8.4.5.2. VS control. As the plasma current is decreased, the internal inductance (l_i) will typically increase, and with it the risk of reaching the limits of the VS control system. The rate at which l_i increases can be reduced by delaying the back-transition to L-mode, and the VS can also be improved by reducing elongation (κ) and/or the distances to stabilizing conducting surfaces [59, 226, 821, 827, 834], but both of these have other repercussions and risks.

8.4.5.3. ELM control. ELM control has been identified as particularly important and challenging during shutdown. As I_p is reduced, q_{95} necessarily is increasing, which changes the coupling of external 3D coils to the edge. If the plasma shape is evolving to aid VS control, this also changes the distance of 3D coils to the edge. The time in ramp-down with type-III ELMs has been found to affect the energy decay rate during the H–L back-transition [835]. ELM control continues to be critical during the discharge termination for control of the accumulation of high-Z impurities [835].

8.4.5.4. Impurity accumulation and radiative power. Impurity control faces several additional complications during ramp-down. As the fueling and density is reduced, impurities can accumulate and exacerbate the radiated power fraction. As already mentioned above, ELM control has been shown to be critical to controlling the impurity accumulation and radiated power fraction [649, 823, 824, 835]. Simultaneously, the heating power is being reduced in order help to ramp-down the stored energy.

8.4.5.5. Density limits. The density typically reduces on a slower time scale than the plasma current [649], which inherently means an increase in Greenwald density fraction. This

can lead to unintended H–L back-transition and eventually Greenwald limit disruptions [28].

8.4.5.6. H–L back-transitions perturbations. When the H–L back-transition occurs, the change in plasma pressure leads to a large perturbation in the radial position control, and risks inward movement to the point of touching the inner wall and causing a large impurity influx or damage to the wall [208, 649, 819]. Likewise, the radial motion changes the gaps between the plasma and the stabilizing conducting surfaces, challenging the vertical control further.

8.4.5.7. Tearing modes and locked modes. As I_p drops, q_{95} rises, which can help with the risk of tearing instabilities chapter 4. However, if the elongation is reduced to aid with VS control, the expected rise in q_{95} will be delayed and slowed, prolonging the risk of tearing instabilities into ramp-down. Additionally, impurity accumulation and high core or edge radiation levels can lead to tearing-unstable profiles [251, 825], which can often lock and lead to disruption during ramp-down [825].

8.4.5.8. Control integration in ramp-down. Despite the challenges in this dynamic phases, ramp-down scenarios for ITER which avoid most of these various issues have been simulated and/or demonstrated in scaled ITER-like scenarios on devices such as JET [13, 822, 823], DIII-D [826, 827, 833], and C-Mod [226]. The RAPTOR code [670] has been applied to the design of optimized ramp-down scenarios to manage many of these challenges with solutions demonstrated on both TCV and AUG [829]. It has been also applied to optimize ramp-down scenarios for DEMO [832], aiming to find fast termination strategies that satisfy VS control limits by optimizing the time traces of plasma current and elongation, while considering the interplay with density limits, radiated power limits, flux consumption, and the effect of the H–L transition timing when ramping-down from burning plasma conditions. An extensive set of AUG ramp-downs has been modeled in RAPTOR to validate that the effects of plasma current, auxiliary heating, and plasma shaping are correctly captured by the model [830]. Integration of these tools and scenario developments should be leveraged for robust, real-time algorithms to intelligently manage scheduled ramp-downs on ITER.

8.4.5.9. Early, emergency termination (for disruption avoidance). The early, controlled shutdown of the discharge has been common technique for minimizing disruption risk after the detection of uncontrolled instabilities. This is done to both attempt to safely shutdown before a disruption might occur, as well as to begin reducing I_p (and therefore magnetic energy) to reduce potential damage risk. Simulations of fast, emergency ramp-downs of 15 MA DT ITER scenarios [820] have been able to reduce the plasma current from 15 MA to 5 MA in as short as 17 s before reaching coil limitations. Stable ITER-like ramp-downs have been simulated on DIII-D

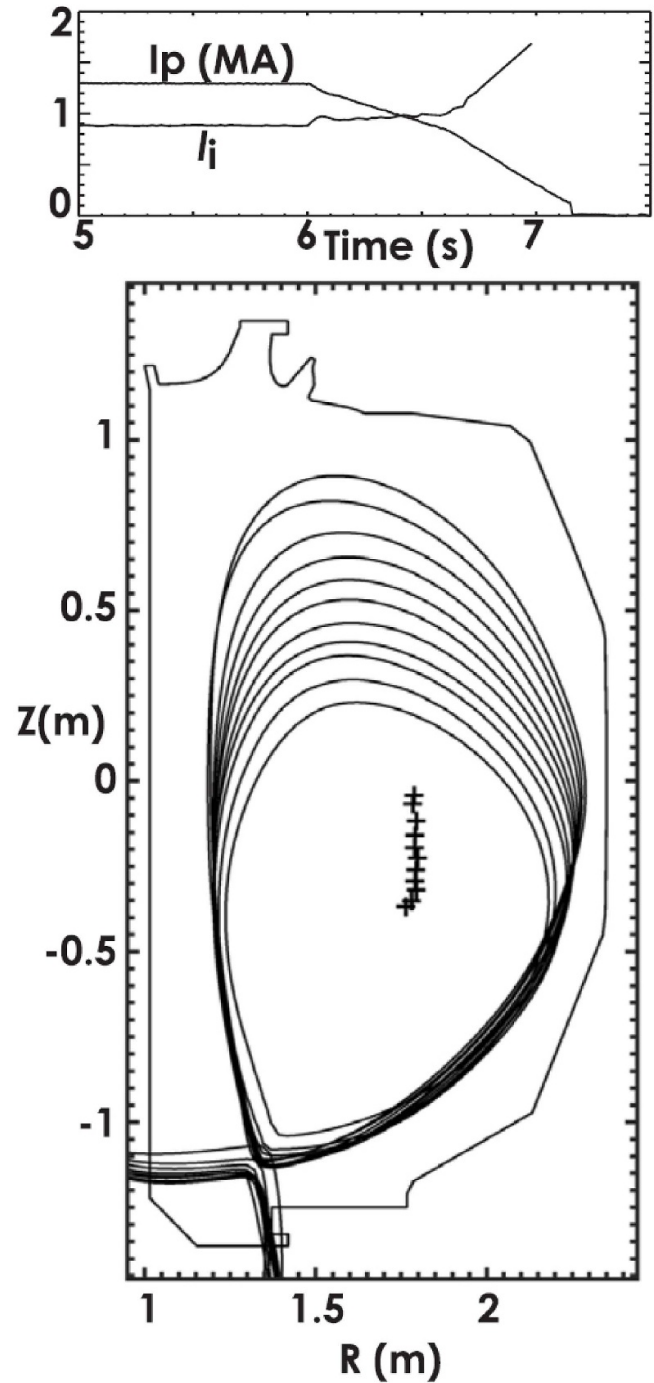


Figure 47. Example ITER-like ramp-down with X-point configuration performed on in DIII-D scaled to mimic a 60 s ramp-down on ITER. Reproduced from [836]. © 2019 IAEA, Vienna. All rights reserved.

up to ramprate which scales to the fastest possible on ITER (approximately 60 s), and an example is given in figure 47. Additional work on DIII-D [828] has also shown that fast, emergency ramp-downs can be optimized to reduce disruptivity after large tearing and locked modes are detected, but that fast ramp-downs alone may be insufficient to reduce the disruption risk in high normalized current ($I_N = I_p/aB_t > 1$)

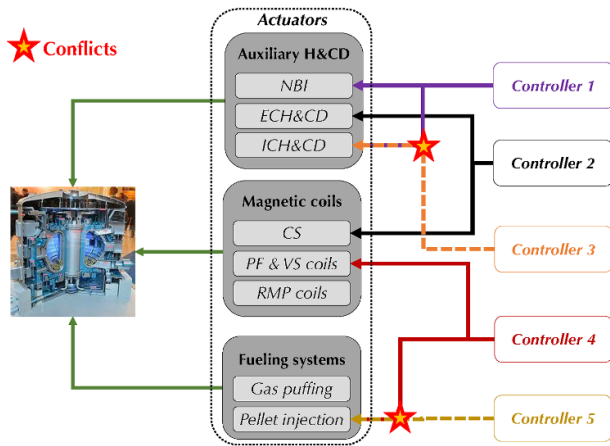


Figure 48. An illustration of a generic actuator-sharing problem for different types of actuators. The stars represent conflicts in which more than one controller tries to command a single actuator.

plasmas. Further work in the community would be beneficial to determine optimal ramp-down strategies tuned for particular emergency situations.

8.5. Scenario control integration

8.5.1. Possible approaches for actuator-sharing. A novel control problem in ITER is the share of actuators required to successfully perform all control functions needed during a plasma discharge [621]. This is illustrated within figure 48. An illustrative example is the sharing of the ECH&CD system for plasma breakdown, core kinetic (burn) control, and advanced control functions such as profile control and NTM control. Those control functions can be executed simultaneously or asynchronously as dictated by the sequence of discharge segments in the pulse schedule. Some general remarks can be made.

A first general remark is that such sequencing of segments, in turn, specifies how the actuators must be shared and ultimately characterizes the actuator-sharing algorithm functions. For instance, plasma breakdown and burn control will necessarily be carried out asynchronously, so they require a switch in the ECH&CD usage with the transition in discharge segment. On the other hand, burn control and profile control may be carried out during the same segment, so an algorithm to continuously share the ECH&CD system is needed.

A second general remark is that the number and types of actuators available within a specific device design, as well as the physics coupling between control functions, determine the nature of the actuator-sharing problem and delimit the control architecture that supports its solution. In ITER's PFPO and FPO phases, simultaneous burn control and NTM control will be enabled by the versatility of its ECH&CD system with dedicated sets of launchers from the upper and equatorial ports [650]. Due to the close relationship between burn condition and magnetic-island dynamics, a significant level of physics coupling exists between these control functions that modifies the corresponding actuator-sharing requirements.

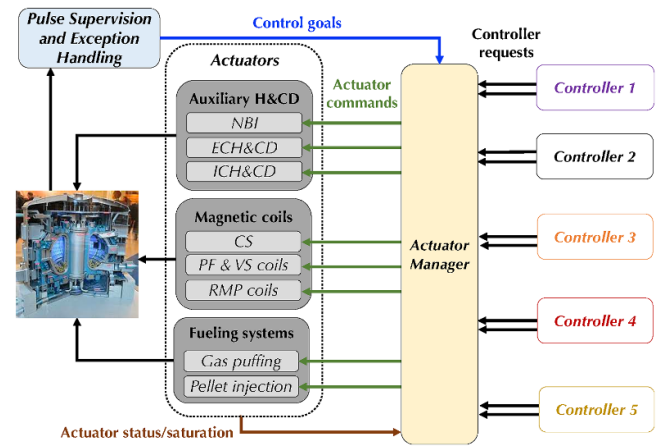


Figure 49. An illustration of how the actuator-sharing problem is solved by the use of an integrated control architecture with an actuator manager. Supervisory and exception handling systems also play a role within such a control scheme.

Broadly speaking, two approaches have been considered to solve the actuator-sharing problem. On the one hand, multi-input, multi-output controllers have been designed that take into account the coupling between different aspects of the plasma dynamics and shared actuators in order to execute several control functions. Examples of such designs can be found in [686, 719, 837–840] for shape control, NTM control, current profile control, beta control, and/or volume-average toroidal rotation. This approach has the advantage of producing relatively compact controllers specifically designed to share a subset of the tokamak actuators, but its extension to the whole ITER system may be challenging.

A second approach is the design of individual controllers for single control functions which are then interconnected within an integrated-control architecture by components such as monitors, supervisors, and exception handlers [841–844], together with an actuator manager that commands the actuators (see figure 49). Along such lines, several algorithms and control architectures have been proposed and tested in present devices, like for example in [687, 845–849]. In [687], some functions for actuator management are developed in simulations for safety factor profile control as well as tearing mode suppression. In [845], an ECH&CD actuator manager is shown to carry out beta and tearing mode control simultaneously in AUG. In [846], various architectural designs are proposed and analyzed for actuator management, and a mixed-integer programming algorithm is presented. In [847, 848], a generic integrated-control architecture with a task-based actuator management scheme is presented and tested in TCV, and in ITER simulations. In [849], an integrated-control architecture is developed and experimentally tested in DIII-D including an optimization-based actuator management algorithm.

The second approach to actuator management facilitates the independent synthesis and design of controllers for specific control functions by independent research teams, while it also entails a clear path to later integrate them into the overall ITER PCS architecture. However, a more complex architectural design must be carried out beforehand [850] in order to

ensure that all the required control functions can be properly supported [651, 851].

It must be noted that the two approaches introduced above are not mutually exclusive. A combination of multi-input, multi-output controllers for several functions, individual controllers for single functions, and an actuator manager can compose an effective path to fulfill the actuator-sharing requirements in ITER. The final solution adopted will depend on the evolution of the ITER experimental plan as well as the experience gained on the critical control needs during its operation [651, 851].

8.5.2. PCS architectures for scenario control integration.

The complexity of the ITER PCS calls for new approaches in organizing and structuring the PCS. The overall design of this structure is illustrated in [140], which details the different components and architectural layers of the proposed PCS design.

- Diagnostic interface layers, implemented as input support functions in the scheme of [140], deals with details of diagnostics, translating their signals (using models e.g. state observer schemes) into generic plasma state representations that do not depend on specific diagnostic signals. This is because diagnostics may fail or have time-varying availability, while control variables can be formulated based on physical quantities that can be estimated by combining multiple diagnostics. Also plasma events are detected at this level.
- Control layers where control tasks are carried out based on the general plasma state descriptions. This layer includes various plasma control functions such as feedback controllers for specific plasma quantities, but also support functions such as plasma health monitoring, real-time forecasting. A supervisory system monitors off-normal events and status of the controllers and takes action in real-time to enable/disable and prioritize different control functions. In particular this provides important information for actuator management, so a decision can be made on which actuators to assign to what control task, in case several control tasks require the same actuator simultaneously.
- Actuator interfaces which serve as a layer of abstraction between the actuator commands sent by the actuator manager to a set of actuators and the specific way in which actuators will implement these requests. For example, this layer might receive a request for EC power deposition on a given radial location in the plasma. This requires an inverse ray tracing calculation to determine the injection angle required for one or multiple EC launchers to reach this plasma location [852].

8.6. Exception handling

ITER is an experimental tokamak device, and as such it will explore different operational scenarios, where the variety of unexpected physical perturbations and technical faults that could interrupt the development of the intended plasma

discharge can be very large. This is more critical for ITER than for present tokamaks, because of the large potential damage that an unexpected event causing a disruption can deal to the machine.

A warning of an off-normal condition that is about to occur, or that is already in progress, is likely to require a response by the PCS in order to prevent a disruption, with the response being determined by the nature of the off-normal condition. If feasible, continuing the discharge is preferable to ending it prematurely, and a controlled shutdown is preferable to a rapid termination. For example, the onset of a locked tearing mode is often a precursor to a disruption, but may allow time for other actions aimed at removing the instability and recovering normal operation, or limiting its growth during a controlled shutdown.

For this reason, it is important that the PCS takes the most appropriate response (exception handling) to any off-normal event, also called exception, depending on the plasma state and the state of the tokamak diagnostics and actuators. In order to do so, in [853, 854] different ways of dealing with an exception have been categorized on the basis of their locality in the AUG tokamak; some exceptions are treated efficiently at the level of a diagnostic system or actuator, as an example we can take the failure of a magnetic sensor that can be replaced by the average of the signals from the neighboring ones. In this case, the method is called basic exception handling. In other cases, the exception could affect systems that are outside the system in which the exception is generated, for instance when data from a diagnostic is used in real-time in complex monitors, or in the processing of other diagnostic data. In this case the failure of one diagnostic system thus can be mitigated by data replacement strategies or Bayesian filtering; while in case of actuators failure, advanced command conditioners can select different actuating systems (e.g. ICRH instead of NBI for central heating).

Finally, in other cases the nature of the exception is such that it cannot be dealt locally, nor limited to a set of systems, and central exception handling is necessary to conduct a coordinated response across the multitude of involved control functions. A specific example of this type of exceptions could be the impurity accumulation in the center of the plasma that could lead to MHD mode locking, or the need to control NTMs using ECCD chapter 4. The action coordination can comprise turning specific functions on and off, changing the settings of functions and changing reference values for feedback controllers and command output functions. While the essence of local exception handling is to sustain the nominal functionality, central exception handling can be applied to achieve more versatile targets that affect the whole tokamak discharge.

Generally, the ability of dealing properly with global exceptions comes with the capability of diagnosing in higher level of detail the state of the tokamak plasma and plant. For this reason, with time more and more sophisticated observers have been developed, some of which can be physics-based observers, like observer of the frequency and radial position of NTMs, but often lately some observers are based on ML tools and detectors, which have been mainly focused on prediction

and avoidance of disruptions using random forest [855–857] algorithms, and GTM [858].

At the same time, more and more complex detection algorithms have prompted the creation of dedicated systems to perform plasma state monitoring systems, like PETRA in [859], which is focused on running complex disruption prediction algorithms in JET, and tokamak plant monitoring systems like VTM [860] which is dedicated to monitoring the heat loads on tokamak FW through an extensive system of infrared cameras.

Effective global exception handling policy shall allow remedial actions that involve the whole tokamak plant, and has the authority of changing the course of plasma discharge in order to recover from the dangerous state, or switch to an emergency discharge termination, or in the eventuality of irrecoverable plasma discharge, triggering DMS. In order to fulfill this need, both JET and AUG have developed independently plasma supervisor systems based on fully programmable discharge ‘segments’, which are an ensemble of control reference waveforms for all tokamak actuators, that if programmed sequentially can make a tokamak discharge. In JET RTPS system [861] each segment can be connected to a series of alarms, that are triggered on specific Boolean conditions (for example amplitude of mode lock signal larger than a specific threshold), or by more sophisticated systems like PETRA. In AUG, the DCS system can be programmed in detail each segment with predefined rules to allow the transition to the next segment of the discharge. Discharge segments can be activated for a limited amount of time, for instance discharge recovery segments, and the DCS can revert back to the standard discharge programming when pre-programmed conditions are met [862].

A further evolution of this concept is considering the tokamak as a FSM, where the condition of the tokamak can be described as an ensemble of a finite number of discrete states. The state of the tokamak is comprised of the smallest number of physics quantities that describe the system for the purpose of control [843]. The state machine can be implemented in various ways, for instance in the DIII-D PCS, the role of plasma state supervisor is covered by the ONFR algorithm [842, 863], which describes the plasma state as a series of nested states, which at the higher level has a ‘controlled state’ and (disruption) ‘mitigation’, while within the ‘controlled state’ medium level states can be found, which describe the recoverable and emergency termination plasma states. Finally, a further level of nested states is present within the recoverable state, which describes the ‘Baseline’ state of the plasma, and finite recovery states that require action by the PCS in order to return to the baseline state (i.e. locked mode, high density disruption risk, etc). A similar description can be found in the KSTAR PCS, which makes use of the same ONFR algorithms, but is described by different states, due to different plasma actuators and superconductive coils systems.

The finite states of a FSM are usually not mutually exclusive, also reflecting the fact that multiple physics events/technical failures can happen at the same time in a complex machine like a tokamak. This means that different control

actions, triggered by the different states, might happen at the same time, and therefore may compete for the same actuator resources that are used in the basic unperturbed plasma discharge. For this reason, tokamak FSM must include a higher decision logic, which must decide which state takes control of the different control actuators when multiple states happen at the same time. This is done in the ONFR through the CRM, which is a decision matrix which assigns an integer priority value to each actuator and each finite state.

Although clear, this logic might be considered coarse, and in other tokamaks (TCV, AUG) actuator-sharing algorithms (section 8.5.1) have been developed that decide the assignment of actuators between different states dynamically, for instance assigning different gyrotrons to different control tasks, like kinetic profile control and NTM suppression at the same time [843, 845, 848].

The design and implementation of sophisticated decision logics in the PCS of a new or existing tokamak can be a very lengthy and complex process, which also includes the commissioning of all the functionalities of the different controllers and the test of the PCS decision chain when different events are triggered at the same time. For this reason, the ability to properly model and simulate the response of the PCS is of the highest importance and can save precious experimental time during the commissioning. Examples of such modeling are listed in [842, 843] where the intrinsic Stateflow[®] finite-state extension to the MATLAB[®] Simulink[®] coding environment are used in order to simulate the PCS response to different off-normal event. Similar modeling has also been carried out for EAST tokamak, for which a FSM based event handling algorithm EDH has been developed, and its functions have been simulated in anticipation to the installation in the tokamak [856]. A very extensive FSM-based algorithm has been designed for ITER PCS anomalous event handling [140, 841], including its interaction with ITER SDN and CIN, which is the system that is in charge of ITER MP. PCS exceptions are handled in a pulse supervision layer operating on top of the PCC feedback loops. It does monitor events, decide on which exceptions to respond, and compute new control references to modify PCC behavior [140].

As mentioned in section 8.1, a simulation environment called PCSSP, has been developed in order to simulate continuous control in ITER PCS, and it has been specifically upgraded in order to simulate ITER PCS response to unexpected events [864]. PCSSP supports the analysis and development of methods to be used by the ITER PCS for handling exceptions to optimize pulses and assists in MP. PCSSP permits to investigate physical and technical events, such as component failures, control degradation, operation domain excess, plasma state bifurcation or instabilities, and interlock activity. Configured by an event schedule, an event generator orchestrates the activation and manipulates the characteristics of off-normal computation. The simulated PCS exceptions are handled in a pulse supervision layer operating on top of the PCC feedback loops. It does monitor events, decide on which exceptions to respond, and compute new control references to modify PCC behavior.

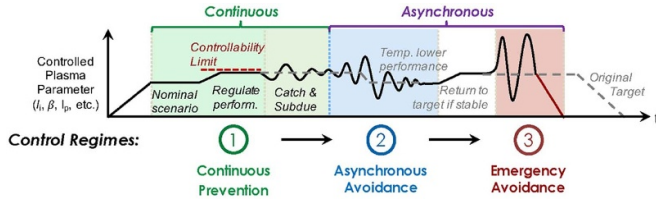


Figure 50. An illustration of the stages of control intervention for robust disruption avoidance. Reprinted from [871], Copyright (2021), with permission from Elsevier.

8.7. Disruption prevention and avoidance

There is a wide variety of plasma and control instabilities which, if not addressed, can escalate to the point of disruption and potentially substantial damage. Systematically identifying and developing control solutions for these problems has been an international focus of the fusion community [836], and continues to be a direct research request from ITER [865] to prepare for its operation in time for commissioning in ITER's first PFPO-1 phase [10] and reach sufficiently reliability to meet ITER's disruption rate tolerances at various operational phases [836]. In depth discussion of the latest advances in handling techniques for a wide variety of physics instabilities is discussed in chapter 4.

Several studies on JET have systematically diagnosed their most common disruption triggers [866–868] and the change in the disruption distribution with the introduction of the ILW [869]. In addition to this, the surveys found that disruptions are more often best characterized by chains of events which can complicate and escalate problems until the point of disruption. The DECAF code has been developed for applying this type of systematic analysis to physics problems on multiple devices for cross-device and cross-scenario comparison [870]. Given the potential interaction of multiple physics, control, and human limitations, robust disruption avoidance needs to incorporate tools for stopping disruptions at multiple points in these chains to prevent problem escalation. Balancing these challenges with exception handling (discussed in section 8.6) and avoidance tools will require control coordination through supervisory control layers as well (discussed in section 8.6 and more in section 8.8).

In general, the methods for controlling the risk due to these potentially disruptive problems can span a variety of control regimes, as illustrated in figure 50. Ideally operations would begin with nominally safe scenarios reasonably far from any instability boundaries, but, as performance goals demand, it may be unavoidable to approach instability limits and controllability limits. This is most certainly the case for IBL Scenario, which is expected to require continuous management of at least tearing modes [221] chapter 4. Should instabilities grow beyond the scope of the standard control algorithms, such events would require asynchronous avoidance techniques to subdue the growing instabilities. Should these interventions provide insufficient remediation, ITER and large future devices will need to escalate further and resort to emergency avoidance techniques like rapid, early controlled shutdowns (section 8.4.5) or even disruption mitigation chapter 4.

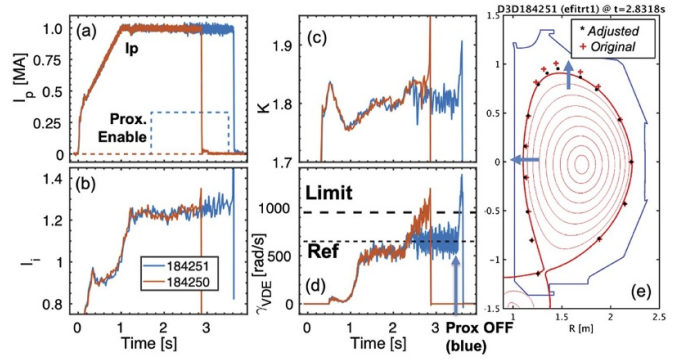


Figure 51. (a) The I_p (solid) and proximity controller enable state (dashed), (b) ℓ_i , (c) elongation κ , and (d) full-linear real-time calculated VDE growth rate (γ) of two discharges: 184 250 in red without the proximity controller, and 184251 in blue with the proximity controller between 1.75–3.5 s. In (e), the real-time rt-EFIT reconstruction of 184251 is shown at 2.832 s along with the separatrix control target locations: the original, pre-programmed control targets as red pluses and the real-time adjusted targets as black stars. Blue arrows indicate the directions of adjustment of inner and top control points. Reproduced from [828]. © 2021 IAEA, Vienna. All rights reserved.

Techniques and methods for real-time monitoring and regulation of the proximity to stability and controllability limits, here referred to as proximity control [621], have been developed to leverage early monitoring and predictive capabilities to make real-time adjustments to the discharge control targets to enforce a minimum distance from instability limits, and potentially multiple stability problems. This methodology fundamentally relies on fast assessment of the current proximity to controllability limits and the architectural capability for making real-time adjustments to pre-shot determined control targets and constraints (i.e. a reference governor). This technique has been applied to several physics problems, including VDEs [828, 871] as well as density limits [872, 873] and unintended H–L back-transitions [874].

8.7.1. Early intervention by proximity regulation and asynchronous control. In the case of proximity control for VDE prevention, real-time calculation of the linear VDE growth rate was monitored relative to known VS control limits on DIII-D [828, 871, 875], and was used a feedback input for adjustment of the plasma shaping targets in real-time. Figure 51 illustrates the reduction in elongation and inner gap by the controller to regulate the VDE growth rate once it reached levels deemed close enough to control limits to warrant intervention. In the case of density limit avoidance, AUG implemented a system monitoring $H_{98/2}$ and the density in real-time to assess nearness to likely H–L back-transition and eventually disruption [872, 873]. After this, combinations of external heating and fueling changes were applied to intercede and prevent further escalation to the level of disruption. A similar scheme was implemented on DIII-D, where a ML prediction for on-coming H–L back-transitions was used to dynamically adjust the minimum NBI heating power constraint during β_N feedback control to prevent unintended H–L back-transitions.

Architectures for asynchronous changes in control algorithms for handling disruption problems as they progress have been developed and implemented on multiple devices (asynchronous avoidance techniques), with especially flexible systems on TCV [844], DIII-D [842], KSTAR [842], and AUG [862]. These systems each have been applied to a variety of disruption causes, including tearing mode and density limits. The AUG system [862] has proven especially robust for density limit avoidance, and enabled detailed diagnosis of this limit. The TCV [844] and DIII-D [842] systems have been developed to manage multiple physics problems in parallel with a state monitoring.

The combination of both Proximity Control and asynchronous avoidance schemes are expected to manage instabilities and reduce the disruption rate to a suitable level to meet ITER's tight disruption budget [836]. The continued development and assessment (including needed accuracy and robustness) of the various disruption avoidance and prevention methods is an ongoing research topic which ITER has identified as a priority for existing devices [865].

8.7.2. ML tools. At its base, early interventions depend on the ability to quickly (ideally faster-than-real-time) assess and predict stability limits. Physics-based control models have been extensively applied, as highlighted in many of the examples above. Additionally, ML has provided a new tool for real-time stability assessment to multiple physics problems in cases where sufficiently fast or accurate physics-based models are not readily available [876]. General ML models for disruption prediction are common on most tokamaks today, often used to trigger early shutdown or mitigation, of which there are many examples [858, 877–881]. While highly valuable for early shutdown and mitigation triggers, models at predicting particular physics-problems provide more actionable information which can be used for early intervention and potentially prevent the need for mitigation. Examples of such tools include real-time ML-based predictors for density limits [882] as well as assessing current confinement mode [874, 883]. Other interpretable ML techniques such as the DPRF [884] provide actionable information by reporting how much individual model inputs are currently contribute (destructively or constructively) to the current stability assessment. By leveraging other technologies in parallel, like GPUs and genetic algorithms in the case of the SORI code [885], these models can be evaluated many times to quickly model the operating space near to the current operating point for optimized control.

Commissioning of disruption prediction systems for the ITER DMS is planned for the first PFPO-1 phase [10]. ML-based disruption prediction tools would ideally be sufficiently trained with few actual disruptions on ITER, and in lower risk scenarios (including at lower plasma current). The applicability of models trained on current devices to ITER is an area of active research including cross-device comparisons of ML-based disruption predictors [855, 882], as is how quickly models could be trained with a minimal number of discharges and disruptions.

8.8. Control issues for long-pulse operation

It is important to realize that there is a strong relationship between the plasma dynamics in the core (inside the last closed magnetic surface) and the plasma dynamics in the SOL and divertor regions. Heat and particle fluxes into the SOL determine its temperature and density, which in turn affect impurity production. Moreover, the impurity flux from the wall and divertor into the core influences the power balance and therefore the burn condition, which in turn affects heat and particle fluxes into the SOL. Therefore, the SOL/wall/divertor and core plasma dynamics are strongly coupled and any practical solution to the problems of burn control and wall/divertor heat/particle-load control demands an integrated approach as described in section 1 and figure 1. Beyond attaining high performance plasma conditions, the ultimate goal of the models and model-based control components (i.e. controllers, estimators, real-time optimizers) developed for real-time burning plasma scenario control (i.e. current, pressure, rotation profile/scalar control) is secure machine operation. This implies that plasma-performance goals in the core may need to be relaxed to be able to guarantee secure divertor operation, particularly in long-pulse operation.

As a complement to local controllers designed for the regulation of the particle/heat flux and temperature on the divertor chapter 5, active control of the core plasma conditions in terms of the internal plasma profiles, their spatially integrated values, and the burn condition is being assessed as a potential mechanism to guarantee that the particle/heat flux and temperature limits for the divertor are not exceeded. Real-time measurements or estimations of the divertor conditions can be fed into a well-designed governor, which will decide on the references (i.e. desired targets) and/or objectives for the different controllers regulating the core conditions either to assist or to supplement the efforts by the local divertor controllers. Assisting the divertor controllers would imply keeping the conditions at the separatrix in terms of density, temperature, and fluxes within a nominal range that guarantees divertor protection by the local controllers during normal operation. On the other hand, supplementing the divertor controllers would imply a response in a much faster time scale by the core controllers in order to move the separatrix conditions away from the nominal range with the ultimate goal of rejecting heat flux/temperature damaging transients that the local divertor controllers cannot reject on their own if the separatrix conditions are kept within the nominal range. Due to their supervisory role, monitoring plasma/machine conditions and modifying control objectives, these governors could become key components of a supervisory system in ITER.

To study the interplay between core and divertor conditions from a plasma control perspective, reduced models of the SOL and divertor region can be coupled with plasma response models for the core. A substantial theoretical and experimental physics effort has been going on for several years to develop predictive models for the SOL/divertor dynamics. As an alternative to 3D turbulent transport codes for the SOL/divertor dynamics, which are still under development and are very computationally expensive, simplified codes such as

SOLPS [886] have been developed by describing perpendicular turbulent transport as a diffusive process with diffusion coefficients empirically fitted to experimental data. Simulation results from SOLPS have been parameterized in terms of core conditions and external inputs, and these scalings have been applied as boundary conditions for core simulations using different predictive codes [601, 887–890]. Further simplifications of these transport models include the so-called two-point models [891–899], which provide relationships between the plasma parameters at the target (e.g. the divertor) and an upstream location far from the target and in contact with the core. Both two-point models [900] and SOLPS-parameterized models [608] can be integrated with core plasma control-oriented models to obtain fast, approximate, self-consistent estimates of the SOL/divertor conditions and to enable the design of scenario control strategies while safeguarding the divertor from dangerous heat and temperature loads.

8.9. Outstanding issues

Outstanding progress has been made in the last couple of decades towards the development of reliable control solutions for a huge variety of problems arising in tokamak operation. While the solutions for this collection of individual control problems still need further development in many cases, the plasma control community has reached a level of maturity where the main challenge associated with reactor-grade tokamak operation arises from the need of operating all these individual control solutions, i.e. controllers, simultaneously in a relatively unknown and more challenging burning plasma environment. For instance, the non-linear dynamics arising from the fusion reaction in long-pulse operation will increase the coupling between control objectives, e.g. profile and burn control, and between plasma regions, e.g. core and divertor control. This integration of individual controllers presents a significant challenge, which is further compounded by the presence of multiple time scales inherent in the plasma dynamics and actuator systems. As devices grow larger and hotter, the disparity between time scales becomes more pronounced. This necessitates the development of advanced control strategies capable of effectively managing dynamics across a wide range of time scales. Moreover, reactor design (less room for diagnostics) and operation (more neutronic noise) will demand control solutions with reduced diagnostic capabilities. Disruption-free operation will demand an overall control system with an unprecedented level of robustness, adaptiveness, and integration. Supervisory and actuator-allocation schemes will be needed to administer in real-time limited resources in the form of actuators as the control priorities evolve during the discharge in response to both normal phase transitions and off-normal events. Estimation in real-time of both the plasma state and the boundaries of stability, controllability and safety will be a critical input to any supervisory system. To add to the challenge, the control solutions will need to be ready before reactor operation. In spite of recent progress, there is still lot of work ahead in order to reach this level of readiness. It is very important to emphasize that much of the needed work is in the form of control-oriented modeling. Control-oriented

models, with levels of complexity tailored to their applications, will be the enablers of many of the needed control solutions. Their applications include synthesis of controllers and observers, design of control integration and actuator allocation schemes, synthesis of estimators of stability, controllability and safety boundaries, generation of synthetic data for training of ML-based controllers and observers, and assessment of control performance in simulations before implementation. It is anticipated that the integration of physics-driven and data-driven modeling techniques, as well as of model-based and data-based control design approaches, will be a key component of this effort in coming years.

8.10. Conclusions

The IOS has demonstrated the transformative power of a model-based approach in advancing control solutions for fusion plasmas. This paradigm shift, marked by the development of physics-based predictive codes and control-oriented models, has enabled significant progress in kinetic control, profile control, and actuator management, as evidenced by successful experimental validations on leading tokamaks.

Looking ahead, the foundation laid by these advancements is crucial for addressing the complex control challenges inherent in reactor scale devices. The ability to extrapolate control solutions, design sophisticated architectures, and validate algorithms through simulations will be pivotal for ensuring robust scenario realization and sustainment in ITER and beyond.

As we transition towards long-pulse, burning plasma operations, the integration of advanced control strategies, coupled with real-time estimation and supervisory control, will be essential for maintaining stable and high performance plasmas. Furthermore, the continued refinement of model-based approaches and the integration of data-driven techniques will be key to unlocking the full potential of fusion energy.

9. Summary

Coordinated by Yong-Su Na

Since the publication of the IPB in 2007, substantial progress has been made in IOS research, both experimentally and through modeling and simulations.

Although current devices cannot fully replicate the plasma conditions anticipated in ITER and future reactors, significant efforts have been made to approximate ITER-relevant or ITER-like conditions in experiments to support performance prediction. Given the inherent uncertainties, such as the transferability of present-day experimental results, it is valuable to explore local optimizations specifically for ITER to help overcome the limitations of the physics basis. A notable example is the reoptimization of plasmas in JET after the installation of the Be/W wall, which led to promising outcomes such as the DTE2 results, even if the process required time. Despite the uncertainties in various models and assumptions, especially in the context of metallic wall environments, modeling and simulations have continued to provide crucial insights, enabling

extrapolation to ITER and beyond and supporting the development and refinement of burning plasma scenarios.

This section summarizes the state-of-the-art in experiments, modeling, and simulations related to IOS research, highlighting key optimizations and discoveries achieved so far. This study goes beyond a simple update of the IPB. It offers valuable insights and guidance for the broader field of burning plasma research, with profound implications for the next generation of fusion experiments and devices.

Experiment. Plasma start-up experiments in ITER-relevant conditions have demonstrated successful EC injection schemes, such as the X2 and O1 waves, alongside improved PF configurations for ECH-assisted initiation, such as TPC. The IBL scenario has been explored through experiments on various devices, highlighting the importance of wall materials in determining plasma performance. Hybrid scenarios have been extensively investigated across various operational windows, achieving high performance with $H_{98y2} \geq 1$ and $\beta_N \geq 2.4$, surpassing the IBL scenario with stationary conditions lasting over $5\tau_E$. For steady-state scenario, two main pathways have been pursued to achieve ITER's goals of fully non-inductive CD with $Q \sim 5$; scenarios with strong reversed shear and high bootstrap current (f_{BS}) and scenarios with weak reversed or low magnetic shear (high auxiliary power fraction, f_{Aux}). Low activation scenarios with hydrogen and helium have also been explored, with systematic studies conducted on isotope effects.

Highlights include the development of ITER demonstration discharges by matching various dimensionless parameters and integrated scenario development. Under ITER-relevant conditions, core-edge integrated scenarios with ELM suppression via RMP and radiative divertor operations have been explored. Additionally, scenarios have been established in an ITER-like tungsten environment at AUG and JET, including T and DT operations at JET. Long-pulse operations have been successfully executed in several devices, particularly in superconducting tokamaks such as EAST and KSTAR with similar coil constraints to ITER.

Modeling and simulations. Modeling and simulations have advanced substantially as well. Benchmark studies for plasma burn-through, energy and particle transport, and various H&CD schemes have been conducted, laying the foundation for predictive simulations for ITER. More advanced anomalous transport models, such as TGLF and QuaLiKiz, as well as a pedestal model like EPED1, have been developed and employed for integrated modeling of ITER scenarios. Full discharge simulations, from plasma initiation to termination, have been carried out to evaluate fusion performance and plasma dynamics. These simulations include free-boundary codes to assess the controllability of ITER plasmas within the constraints imposed by the voltage capabilities of the power supply systems and FW geometry.

Outstanding progress has also been made in developing control solutions through a model-based approach, which facilitates the extrapolation of control strategies from current devices to future reactors like ITER. This approach enables the

design of control architectures and actuator-sharing strategies that incorporate the physical dynamics of the system.

Despite these significant advancements, several outstanding issues remain.

Plasma initiation. Some experiments indicate that ECH may affect the electron distribution function, raising the possibility of runaway electron generation in ITER. Dedicated experiments are needed to investigate this issue and to determine the operational conditions required to avoid it. To optimize control parameters in ITER and ensure reliable plasma initiation at high pre-fill gas pressures without runaway electron generation, it is necessary to improve the predictive model including runaway electron generation and validate it with existing devices. Additionally, appropriate methods for real-time monitoring and control of runaway electrons during the plasma initiation phase should be established and tested in current devices.

Low activation phase scenario. Uncertainties remain as empirical extrapolation to ITER can be unreliable, and predictive plasma models lack sufficient validation for accurate ITER predictions. Key challenges include impurity transport in H-mode plasmas with ITER-like low collisionality pedestals and the access to and confinement of H-mode in deuterium plasmas under electron heating and low density conditions, where low edge ion heat flux may play a role. Validating pedestal models for ITER low activation scenarios is crucial, particularly regarding isotope effects on transport, stability, and electron versus ion temperature pedestal gradients in electron-heated plasmas. Lastly, integrated core-edge-SOL modeling must be validated against experimental data to ensure reliable scenario predictions.

IBL scenario. Challenges in achieving stable conditions at the H-mode threshold have been identified, particularly with tungsten walls, where impurity accumulation occurred due to reduced ELM frequency. Adjustments such as gas puffing and boronization have impacted H-factors and ELM behavior, while material changes have led to deviations from established empirical scaling laws. A major challenge in predicting reliable scenarios for the IBL is the lack of a complete set of validated physics models for ITER's operational space, including models for the pedestal during transient ramp phases and pedestal density. Energy and particle transport models are not fully validated in ITER-relevant regimes, and modeling of the ramp-up and ramp-down phases has notable uncertainties due to the lack of self-consistent evolution of density, H&CD, MHD, and actuator control. The methodology for controlling impurity accumulation to avoid core radiation has not been fully identified, necessitating further experiments in current devices. Particle balance will be crucial for sustaining the IBL discharge, especially in long-pulse operations. Unfortunately, this balance heavily depends on hardware specifications, such as the vacuum vessel's pumping capability and the fueling rate of the NB sources. Data collection from long-pulse machines may be required.

Hybrid scenario. The physics mechanisms for clamping the q -profile against current diffusion need to be further identified,

as these are closely tied to the confinement and stability properties of hybrid scenarios. Candidate mechanisms, such as flux pumping, must be evaluated across various devices before being projected to ITER. Even if a mechanism is identified, it remains uncertain whether it will operate similarly in ITER due to differences in plasma parameters including the dimensionless parameters. The mechanism behind confinement enhancement requires further clarification, supported by additional experimental data, including high Greenwald density fractions, high electron heating fractions, and low input torque. Additionally, the dependency on wall material and isotope effects needs to be explored more systematically to assess the evolution of the q -profile, L–H transition, and pedestal dynamics within the scenario, thereby enhancing understanding of these effects in DT plasmas. In addition to the issues that arose in IBL scenario modeling, predictive integrated modeling of hybrid scenarios must be further advanced by validating against ITER-relevant plasmas, such as low torque and predominantly electron-heated plasmas, while also incorporating various MHD activities. Furthermore, tungsten transport must be addressed in core-pedestal-edge-SOL-divertor/FW integrated simulations.

Steady-state scenario. Despite significant progress in achieving high normalized fusion performance, the challenge remains to simultaneously attain both high performance and steady-state conditions. Although high global confinement is required, it is unclear whether the high confinement quality achieved in current experiments can be extrapolated to ITER conditions. Experimental demonstrations face the key difficulty of providing sufficient external off-axis CD for both high f_{Aux} scenarios and high f_{BS} scenarios in present tokamaks. Further exploration of scenarios is needed to identify conditions that can achieve these goals simultaneously. Predictive modeling indicates that some form of upgrade to ITER's FPO phase H&CD capabilities would be necessary. Both passive stabilization and active control of potential RWMs could be important physics issues during plasma operation with high β_N . Fast particle confinement and the control of associated MHD instabilities also present challenges, thus, optimizations of scenario parameters or control approaches are required to minimize their impact on fusion performance. Lastly, core-edge integration remains a serious challenge for achieving $Q = 5$ steady-state operation, particularly regarding ELM control, detachment, and impurity control.

Actuators for integrated scenario. The validation of the predictability of integrated control, considering the interplay of different actuators and operational boundaries, remains a critical issue requiring further investigation. This includes examining the compatibility of density control with ELM mitigation and suppression. The anomalous responses caused by the application of actuators, such as helical structures, necessitate theoretical explanations and further validation of predictability. Extrapolating the phenomenology related to control via H&CD and PLCS actuators to reactor scale

plasmas requires a better theoretical understanding and the extension of experiments on existing machines that closely mimic reactor scale parameters, along with comprehensive integrated simulations. For instance, the multi-step processes involved in beam ionization and plasma fueling of ITER's NBI, the roles of two main ion fueling methods, gas puffing, and pellet fueling exemplify this need. Additionally, operating beyond the boundaries of the controllable operational space presents another challenge, as regular control by H&CD and PLCS actuators may become impossible for reasons unrelated to the actuator design.

Scenario control. While solutions for individual control problems still require further development, the main challenge associated with reactor grade tokamak operation arises from the need to implement all these individual control solutions in a relatively unknown and more challenging burning plasma environment. The coupling between control objectives will increase during long-pulse operation, and reduced diagnostic capabilities under reactor conditions will further complicate control efforts. Disruption-free operation will be critical in fusion reactors, necessitating an unprecedented level of robustness, adaptability, and integration within the overall control system. Additionally, the required control solutions must be ready before reactor operation begins. This necessitates advancements in control-oriented modeling that integrate both physics-driven and data-driven modeling techniques, as well as model-based and data-based control design approaches.

Controlling strongly non-linear, self-organizing plasmas presents a significant challenge, particularly for hybrid and steady-state scenarios, which may be essential for achieving higher Q when fusion power is utilized. The dependence of the q -profile during the flat-top phase on the specific path taken to reach it raises the question of whether transitions to other q -profile shapes are hindered by non-linear properties of the plasma, or whether active control is required to prevent the plasma from evolving toward less desirable profiles. This highlights the need for more systematic studies to tailor q -profiles starting from the current ramp-up phase in hybrid and steady-state scenarios. Additionally, developing scenarios for high-power DT plasmas in a tritiated vessel is particularly complex due to the severe consequences of failure, whether in the form of low performance or high-energy disruptions. These plasmas differ greatly from those in the low activation phase, where scenario options are more limited. A key issue is ensuring sufficient simulation fidelity; determining what level of accuracy is necessary remains an open question. Furthermore, a critical challenge lies in leveraging results from today's experiments, as these plasmas often differ significantly from those expected in ITER. The feasibility and concept of model validation using current experiments is highly complex and remain intricate.

Additionally, the impact of changing the wall material from Be to W in ITER under the new IRP must be systematically evaluated for scenario development and integrated modeling. Key aspects include confinement, impurity transport, L–H

transition, pedestal behavior, and core-edge coupling. This wall change introduces uncertainties, particularly in confinement time scaling. Notably, H_{98y2} values below 1 have often been observed in W-wall experiments at AUG and JET. To accurately assess the wall material's impact, it is necessary to analyze actual confinement time rather than relying solely on the H-factor, with support from integrated modeling.

The IOS group will continue to address the outstanding issues mentioned above through JEXs and JAs. The development of IBL, hybrid, and steady-state scenarios will proceed under more ITER-relevant conditions, including low torque, electron-dominated heating, and high density. Particular focus will be placed on developing IOSs by incorporating ELM mitigation/suppression, radiative divertor strategies, long-pulse operation, and more into the aforementioned scenarios. Integrated modeling will be further pursued for higher fidelity in core-pedestal-edge-SOL-divertor/FW simulations within the context of full discharge modeling from plasma initiation to termination, including momentum and impurity transport. Integrated control with actuator-sharing will also be addressed through model-based control methods and ML. In particular, as ML/AI techniques continue to rapidly evolve, their application to scenario studies will be highly valuable. ML utilizing databases from various devices could help explore new plasma regimes beyond those currently known. Once a new regime is identified, efforts can be made to project it onto ITER or beyond. Additionally, ML based on theory-driven models or physics-informed approaches can be applied for projections to next-step machines.

All these activities will need to be re-assessed in burning plasma environments, particularly given the lack of diagnostics due to the high neutron fluence and the installation of blankets.

Acknowledgments

The authors would like to express our deep gratitude to George Sips, Shunsuke Ide, Tim Luce, and Joseph Snipes for their exceptional leadership in guiding IOS activities as chairs. We also extend our heartfelt thanks to the Guest Editors responsible for this chapter, Charles Greenfield and Francesca Poli, as well as the entire team of guest editors for this Nuclear Fusion Special Issue. Additionally, we are grateful to all the experts in the ITPA IOS topical group who participated in discussions about this manuscript at IOS and other meetings over the past four years, providing valuable comments and suggestions. Special thanks to Paul Bonoli for his helpful feedback. Finally, we would like to sincerely thank Mr Min Ki Jung (SNU) for his thorough editing of the entire chapter. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

This work was supported by Korean Ministry of Science and ICT under the Korea Institute of Fusion Energy (KFE) R&D Programs of 'High Performance Tokamak Plasma Research & Development (Code No. EN2501-16)', the National R&D Program through the National Research Foundation of Korea (NRF) funded by the Korea government (Ministry of Science and ICT) NRF-2021M1A7A4091135 and RS-2024-00358933. The authors also gratefully acknowledge The Research Institute of Energy and Resources, The Institute of Engineering Research, and the SNU Energy Initiative at Seoul National University. Eugenio Schuster was supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, under Award DE-SC0010661. This work is also supported in part by the US Department of Energy (Grant Nos. DE-SC0010685, DE-FC02-04ER54698).

Appendix

Table A.1. Joint experiments in IOS TG.

IOS-1: Baseline scenario	
IOS-1.1	ITER baseline, at $q_{95} = 3$, $\beta_N = 1.8$, $n_e \leq 0.85n_{GW}$ (2009–2016)
IOS-1.2	Study seeding effects on ITER baseline discharges (2009) Divertor heat flux reduction in ITER baseline scenario (2013–)
IOS-1.3	Operation near P_{LH} (2013–)
IOS-1.4	ITER baseline with dominant electron heating/low torque (2017–)
IOS-2: Plasma start-up, ramp-up/ramp-down, non-activation scenario	
IOS-2.1	ECRH breakdown assist at 20° toroidal angle (2008) Compare helium H-modes in different devices (2015–2019)
IOS-2.2	Ramp-down from $q_{95} = 3$ (2009)
IOS-2.3	X2-breakdown assist after application of U_{loop} (2011) Development of integrated H-mode termination scenarios
IOS-2.4	Experimental comparison of RF pre-ionization plasma development
IOS-2.5	Start-up on W limiter surfaces
IOS-3: Steady-state scenarios	
IOS-3.1	Beta limit for steady-state with ITER recommended q -profile (2008) Evaluation of steady-state scenarios with $\min(q) \sim 1$ for meeting the ITER physics objective (2017–)
IOS-3.2	Define access conditions to get to steady-state scenario (2009–2014)
IOS-3.3	Core confinement for $q(0) = 2$ Fast ion and thermal energy transport with elevated q_{min} (2014–2019)
IOS-4: Hybrid scenarios	
IOS-4.1	Access conditions for hybrid with ITER-relevant restrictions (2009–2014) Origin of improved confinement in hybrid Advanced Inductive scenario
IOS-4.2	ρ^* dependence on transport and stability in hybrid scenarios (2010–2017)
IOS-4.3	Collisionality scaling of confinement in advanced inductive plasmas (2011–2017)
IOS-5: Heating and current drive actuators	
IOS-5.1	Ability to obtain and predict off-axis NBCD (2008) ICRF Impurity Generation (2016–)
IOS-5.2	Maintaining ICRH Coupling in expected ITER Regime (2010–2015)
IOS-5.3	Assessment of LHCD at high density for extrapolation to ITER advanced scenarios (2010–2016)
IOS-6: Control of experimentally simulated burning state	
IOS-6.1	Modulation of actuators to qualify real-time profile control methods for hybrid and steady-state scenarios (2008–2011) Density and fuelling control
IOS-6.2	l_i controller (I_p ramp) with primary voltage/additional Heating (–2010)
IOS-6.3	Control of experimentally simulated burning state (2011)
IOS-6.4	Integrated magnetic and kinetic profile control (2012)

Table A.2. Joint experiments with other TGs.

PEP-IOS-1	Dependence of H-mode pedestal and integrated performance on H isotope and ion species mass
MDC-IOS-1	Control for Disruption-free Operation

Table A.3. Joint activities in IOS TG.

IOS-JA1	ITER demonstration discharge modeling
IOS-JA2	ITER ramp-down simulations
IOS-JA3	Hybrid benchmark
IOS-JA4	Steady-state benchmark
IOS-JA5	Ramp-up of ITER hybrid and steady-state scenarios
IOS-JA6	IC Benchmarking
IOS-JA7	Steady-state exploration
IOS-JA8	Burn control simulation
IOS-JA9	Optimization of operational space for long-pulse scenarios
IOS-JA10	Scenario modeling for low- and non-activation early operation of ITER
IOS-JA11	Particle transport benchmark
IOS-JA12	Pedestal modeling
IOS-JA13	Pellet source modeling
IOS-JA14	Plasma burn-through modeling benchmark and validation
IOS-JA15	Development of fast electron discharges at plasma initiation

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