Available online at www.sciencedirect.com
SCIENCE DIRECT®
ELSEVIER
Journal of Nuclear Materials 337–339 (2005) 808–815
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Overview of goals and performance of ITER and strategy for plasma–wall interaction investigation

M. Shimada [a,*], A.E. Costley [a], G. Federici [b], K. Ioki [b], A.S. Kukushkin [b], V. Mukhovatov [a], A. Polevoi [a], M. Sugihara [a]

[a] *ITER International Team, 801-1 Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken 311-0193, Japan*
[b] *ITER International Team, Boltzmannstrasse 2, Garching bei Muenchen, Germany*

Abstract

Recent progress in experiments, modeling and theory has increased confidence that ITER can achieve its goal of high Q (>10) in inductive operation. Further, experimental results in the 'improved hybrid' regime suggest a possibility of high Q (>10) operation in a long pulse (>1000 s) with benign ELMs. However, considerable uncertainty still exists in the prediction of several key aspects; for example, tritium retention, disruption, impurity control, ELMs, SOL transport and dust. ITER requires flexibility of operation scenarios at least in the early operation phase to accommodate uncertainties in prediction, to explore wide operational spaces and to incorporate newly developed control schemes. Because of these uncertainties conservative assumptions are adopted in performance predictions, and step-wise implementation of reactor-relevant plasma-facing material, such as tungsten, are planned.

PACS: 52.55; 28.52; 52.25.V; 52.40.H
Keywords: ITER; Disruption; ELM control; Tritium retention; Impurity transport

1. Introduction

ITER [1] is an experimental fusion reactor for the investigation and demonstration of burning plasmas with dominant alpha-particle heating. The design of the major components is completed and procurement packages have been prepared. It is anticipated that the construction of ITER will begin in the near future.

The success of ITER will largely depend on the control of plasma–wall interactions (PWI). These have power fluxes, time scales and divertor particle fluxes one or two orders of magnitude higher than in present devices.

This paper summarises the goals and projection of plasma performance and presents a strategy for plasma–wall investigation in ITER. Section 2 presents the goals of ITER. ITER machine parameters and diagnostics are presented in Section 3. The performance prediction is discussed in Section 4. Section 5 discusses operation phases and a strategy for plasma–wall interactions. A plan of blanket tests is also presented. Section 6 summarises the conclusions.

* Corresponding author. Tel.: +81 29 270 7770; fax: +81 29 270 7460.
E-mail address: shimadm@itergps.naka.jaeri.go.jp (M. Shimada).

0022-3115/$ - see front matter © 2004 Elsevier B.V. All rights reserved.
doi:10.1016/j.jnucmat.2004.09.063

2. The goals of ITER

The primary objective of ITER is the investigation and demonstration of burning plasmas. A burning plasma is characterised by dominant alpha-particle heating (i.e. >2/3 of total heating power) that will enable the plasma to regulate its own profiles. The burning plasma in ITER will be characterised by high alpha particle population, small ratio of ion gyro-radius to plasma radius, low collisionality and long pulse (300–500 s or longer). These features will extend the forefront of the science of plasma, which is rich in complex and non-linear processes (e.g. turbulence), leading to self-organised states and structures [2].

ITER also aims at investigating steady-state plasmas, which require a high fraction (>50%) of self-generated bootstrap current. The local magnetic shear modified by bootstrap current can reduce the turbulence and associated transport, which will further enhance the self-regulation of the plasma. These exciting new regimes of ITER plasmas could exhibit novel and interesting phenomena, that could lead to discoveries of new operation regimes attractive for a reactor.

The performance specifications adopted for ITER by the ITER Council in June 1998 are the following:

(1) to achieve extended burn in inductively driven deuterium–tritium plasma operation with $Q \leqslant 10$ (Q is the ratio of fusion power to auxiliary power injected into the plasma), not precluding ignition, with an inductive burn duration of between 300 and 500 s;
(2) to aim at demonstrating steady state operation using non-inductive current drive with $Q \leqslant 5$;
In terms of engineering performance and testing, the design should
(3) demonstrate availability and integration of essential fusion technologies,
(4) test components for a future reactor, and test tritium breeding module concepts; with a 14 MeV-neutron power load on the first wall $\leqslant$0.5 MW/m^2 and fluence $\leqslant$0.3 MWa/m^2 (neutron power load integrated over years).

3. ITER machine parameters and diagnostics

The magnitude of the main performance parameters of ITER are listed in Table 1. ITER is equipped with multiple control capability, in particular, for the control of plasma shaping to sustain stability and improved confinement. It has high-field-side pellet injectors for efficient fuelling and ELM mitigation, a semi-closed divertor for power and particle control, ECCD and saddle coils for stabilising mhd modes, and impurity gas injection for divertor target heat load control and disruption mitigation. The planned heating and current drive systems are summarized in Table 2. A combination of heating and current drive systems of negative-ion-based neutral beams, electron cyclotron waves, ion cyclotron waves and lower hybrid waves will further enhance the flexibility of operation.

Table 1
Reference performance of ITER and flexibility

	Reference performance	Flexibility
Fusion power	500 MW (∼2000 s)	700 MW (∼300 s)
Burn time (I_p)	∼500 s (15 MA)	3000 s (9 MA)
κ_x/δ_x	1.85/0.49	2.0/0.55 (a = 1.85 m)
Pumping/fuelling	120 Pa m^3/s	240 Pa m^3/s

Table 2
Heating power available for ITER

	Initial	Possible upgrade[a]	
NB (MW)	33	50	33
RF (MW)[b]	40 (EC + IC)	80	100
ECCD for NTM (MW)	20	40	

[a] A total plasma heating power up to 110 MW may be installed in the subsequent operation phases.
[b] The RF power includes ECCD for NTM stabilisation.

An extensive diagnostic system will be installed on ITER to provide the measurements necessary to control, evaluate and optimise the plasma performance and to study burning plasma physics [3,4]. Because of the harsh environment, diagnostic system selection and design has to cope with a range of phenomena not previously encountered in diagnostic implementation. Since 1992 these issues have been tackled in a coordinated programme involving all ITER partners, and a comprehensive diagnostic system that will meet the needs for measurements including measurements of plasma–wall interactions is in preparation.

4. Projection of ITER performance

The projection of ITER plasma performance is based on methodologies documented in the ITER Physics Basis (IPB) [5] and has been further developed in recent years [6]. The projection studies on plasma performance described in the Final Design Report [1] are based on conservative assumption on helium exhaust: $\tau^*_{\rm He}/\tau_{\rm E} = 5$ ($\tau^*_{\rm He}/\tau_{\rm E}$ is a characteristic time for helium pumping normalised by energy confinement time) which corresponds to 4.3% (on axis) and 3.2% (volume average) of helium concentration. The other impurity concentrations assumed are: 2% beryllium and 0.12% of argon, which is required to dissipate 47 MW radiatively inside the separatrix under an approximation of ionisation equilibrium (coronal equilibrium). Z_{eff} is 1.67 and the DT fraction is 0.83. In a similar case with a carbon concentration of 1.2% and beryllium 2%, the radiation power inside the separatrix is 31 MW. With an addition of helium (3.2%) and beryllium (2%), the Z_{eff} is 1.66 and the DT fraction is 0.78. The fusion power projected with these assumptions is 390 MW at a plasma current of 15 MA and $n/n_G = 0.85$ (n/n_G is a volume-averaged density normalised by the Greenwald density n_G, defined by (n_G $(10^{20}\ \mathrm{m}^{-3}) = I_p/(\pi a^2)$, I_p is a plasma current in MA, a is a horizontal minor radius in m). The carbon concentration assumed is roughly consistent with B2/Eirene calculation of carbon concentration at the separatrix, which varies in the range of 0.7–1.8% depending on the sputtering rate at the first wall [7]. The beryllium concentration is estimated conservatively from JET beryllium limiter experiments [8,9].

Early calculations suggested that elastic scattering of helium neutrals by hydrogenic ions could enhance helium atom transport toward the divertor [10,11]. However, spectroscopic measurements have shown that this process mainly heats the helium atoms in the divertor [12], thus increasing their mean-free-path. This results in two competing effects: more helium atoms reach the pumping duct and more reach the core. Inclusion of this process in the divertor modelling shows that the efficiency of helium exhaust can be improved by a factor of 3–5 in ITER [13], suggesting that $\tau^*_{He}/\tau_E \sim 1.2$ may be feasible. This would lead to improvement of core performance, and, for example, to an enhanced fusion power of 500 MW at a plasma current of 15 MA, $n/n_G = 0.85$ and $H_{H98(y,2)} = 1.0$ (Fig. 1) ($H_{H98(y,2)}$ is a confinement quality factor: energy confinement time normalised by the ITER-98(y,2) scaling [5]). Energy confinement at high plasma density $\sim n_G$ has been improved by strong shaping, pellet injection and impurity injection [14,15], which suggest that $H_{H98(y,2)} = 1.0$ could be achieved at $n/n_G = 0.85$ in ITER. Furthermore, this improvement suggests a possibility of achieving high $Q > 50$ within the range of projection uncertainty ($H_{H98(y,2)} = 1.1$) at a plasma current of 15 MA [16]. However, further investigation is required in the particle and heat transport to reduce the uncertainty in the projection of these discharges with density profile peaking to ITER. Further understanding of impurity generation and transport is also required for more reliable projection of performance.

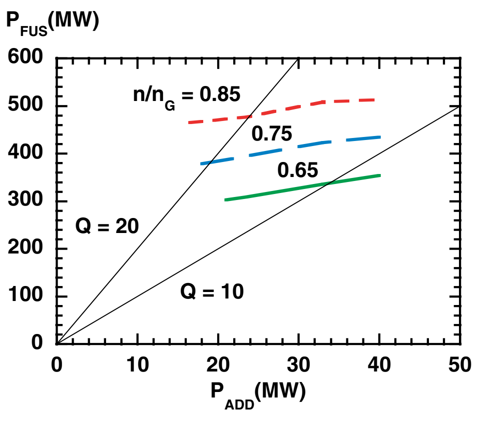

Fig. 1. Fusion power vs. additional heating power with a plasma current of 15 MA.

Theory-based modelling of transport has confirmed the projection based on the empirical confinement scaling, with an assumption that high edge pedestal temperatures (3–4 keV) will be achieved [16]. These pedestal temperatures are within the range of projection with empirical scalings [16]. The target heat load associated with the edge localised modes (ELMs) is a major concern for this inductive operation in Type I ELMy H-mode. The projection of the amplitude of ELM heat load is highly uncertain, because the physical mechanism of ELM is not yet fully understood. Recent experiments suggest that the target heat loads could be suppressed by frequent pellet injection [17] and by edge ergodisation [18]. These experiments also show that confinement deteriorates, albeit modestly, at frequent pellet injection, which suggests that further improvement is necessary. No confinement deterioration is observed with edge ergodisation, but its effect on ELMs should be demonstrated at lower $q_{95} \sim 3$.

The hybrid operating mode, based on a combination of inductive and non-inductive current drive, leading to a long pulse operation (>1000 s) with a significant fusion power (>300 MW, $Q = 5$) at a medium safety factor ($q_{95} = 4$–5) and conservative confinement assumption ($H_{H98(y,2)} = 1$), is a possible operating mode of ITER [1]. Recently, 'improved hybrid' modes have been discovered that are under investigation in many tokamaks [19–22], suggesting that improved confinement and high beta can be achieved with tailoring of the current profile at a medium safety factor ($q_{95} = 4$–5) and $q_{min} = 1$. For example, in ASDEX-Upgrade, $H_{H98(y,2)} = 1.2$ is achieved at $n/n_G = 0.85$ [19]. If ITER operation in this mode could result in achieving these normalised parameters, fusion powers of ~350 MW, $Q > 10$ would be achieved at $\beta_N \leqslant 2.2$ (Fig. 2). β_N is a beta value (plasma pressure normalised by magnetic pressure) in % normalised by $I_p/(aB_T)$, in the unit of MA, m and T). The required β_N is well below the no-wall ideal MHD limit for resistive wall mode. A burn time longer than 1000 s would be expected (Fig. 2). This operation scenario is a potential candidate for an operation mode with high Q, long pulse and benign ELMs. In this regime, neoclassical tearing modes could become unstable, which need to be stabilised by using ECCD.

In the steady state (SS) operation, the total plasma current at the current flat-top phase is generated

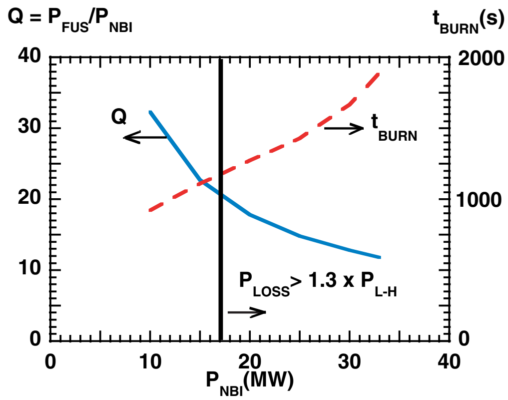

Fig. 2. Fusion gain and burn time vs. NBI power with a plasma current of 12 MA at HH = 1.2 and density of 85% of Greenwald density.

non-inductively by the bootstrap effect, neutral beam injection and RF waves. To provide SS operation in ITER with $Q > 5$ with the available additional power, it would be necessary to decrease the plasma current to 9–12 MA. In this case the bootstrap current fraction will increase due to an increase of the poloidal beta. To achieve SS operation with $Q > 5$, an improved confinement with $H_{H98(y,2)} \sim 1.5$–1.6 is necessary. Example scenarios for the SS operation are summarised as following:

(1) 9–10 MA, NB + LH, RS, $H_H \sim 1.3$–1.5, $\beta_N \sim 2.5$, $Q \sim 5$ [23].
(2) 12 MA, NB + LH, RS, $H_H \sim 1.5$, $\beta_N \sim 3.6$, $Q \sim 8$ ($P_{fusion} = 0.7$ GW) [23].
(3) 9 MA, NB + EC, WS, $H_H \sim 1.5$–1.7, $\beta_N \sim 2.7$, $Q \sim 5$.

The requirements on confinement and beta are within the range achieved experimentally. Furthermore, numerical studies suggest that stabilisation of resistive wall mode is possible at these values of normalized beta with external coils to be implemented in ITER [23].

To summarise, projection studies based on recent results of experiments, theory and modelling suggest improved prospects of achieving the goals of ITER.

5. Operation plan and strategy for plasma–wall interaction investigation

Fig. 3 shows an initial operation plan of ITER. Since ITER will be the first experimental fusion reactor, flexibility in the available operating scenarios is very important especially during the first 10 years of operation to accommodate uncertainties in projection, to explore a wide range of parameter space and to incorporate new control schemes. Various operating modes (inductive, hybrid and steady state) are being prepared with a range of plasma current for potential different confinement modes. In addition, the capability to operate with H, D and DT (and He) is being prepared along with different fuelling and particle control methods for the initial phase of ITER operation. Flexible plasma control will allow operation of 'advanced' scenarios based on active control of plasma profiles by non-inductive current drive, auxiliary heating and fuelling. Further, ITER has the capability to change the material of the first wall and divertor target if necessary. In the subsequent 10 years of operation, more emphasis will be placed on engineering tests, e.g. blanket development, requiring reliable long-pulse operation.

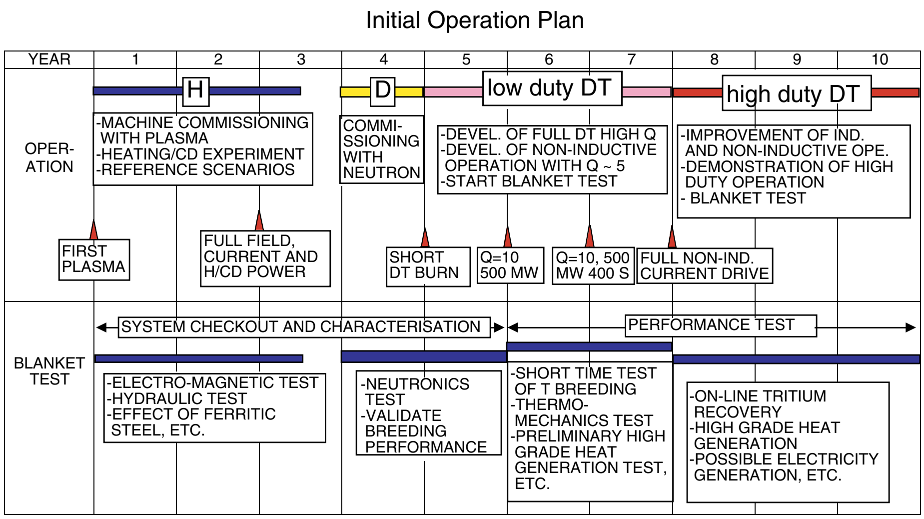

Fig. 3. Initial operation plan and blanket test plan of ITER.

5.1. Goal of plasma–wall interaction investigation

The goal of plasma–wall interaction (PWI) investigation in ITER is to develop the understanding of PWI processes and to establish methodologies for prediction and control of PWI in DEMO and ultimately fusion reactors. The success of ITER largely depends on the control of PWI, which will be challenging because the power, particle fluxes to the divertor and time scales are one or two orders of magnitude higher than in present devices. Significant progress has been made in the understanding of PWI processes (e.g. divertor codes) and possible control methods have been demonstrated (e.g. ELM mitigation by frequent pellet injection, benign ELM and edge ergodisation). However, considerable uncertainty exists in some key areas including T retention/removal, heat load due to disruption and ELMs, impurity transport, SOL transport and dust. This indicates the need of further pursuit of physics understanding and continued development of control methods.

5.2. Selection of initial set of plasma facing materials

Fig. 4 shows a schematic of plasma and plasma facing components of ITER. For the initial operation, the first wall and limiter are planned to be covered by beryllium, which is low Z and has high oxygen gettering capability. The divertor baffle and dome will be covered by tungsten. Tungsten has a low sputtering rate, thus low erosion and long lifetime are expected. The divertor target will be covered by graphite carbon–fibre-composite (CFC), which will not melt under transient power loads such as ELMs and disruptions. CFC is compatible with a wide range of plasma regimes, due to its capability to withstand impulsive heat loads, and carbon is a very good radiator in the divertor.

High-Z metals, such as tungsten, are promising as the plasma-facing material for fusion reactors, because of their resistance to erosion. However, if they were used for divertor targets, the heat load of disruption would melt the target, creating irregular surfaces, which would melt during the normal operation phase and deteriorate the performance of the core plasma. In a number of tokamak experiments, the first wall materials are observed to accumulate at the plasma center during enhanced confinement with internal transport barrier (ITB) [24,25] and H-mode without ELMs [26]. If they were used for the first wall, an unacceptable amount of high Z impurities could accumulate at the center of the core plasma in enhanced confinement modes. With central heating, such an accumulation can be controlled [24,25], but the critical power for the impurity accumulation control cannot be predicted reliably. During the initial phase of the discharge lasting for $\sim$30 s, the discharge leans against limiters. If high Z metals are used for the limiter material, high Z ions could accumulate at the center. In TEXTOR, the limiter material is observed to accumulate at the center, in agreement with the prediction of neoclassical transport theory [27]. Therefore carbon–fiber-composite (CFC) will be used for the divertor target and beryllium will be used for the first wall during the first phase of operation, to allow flexibility in operation. These arguments are supported in [28,29]. As the experiments progress, the PFMs can be replaced with materials with higher erosion resistance, e.g. tungsten. Establishment of control schemes of disruption and impurity transport is a prerequisite before this replacement of plasma facing materials.

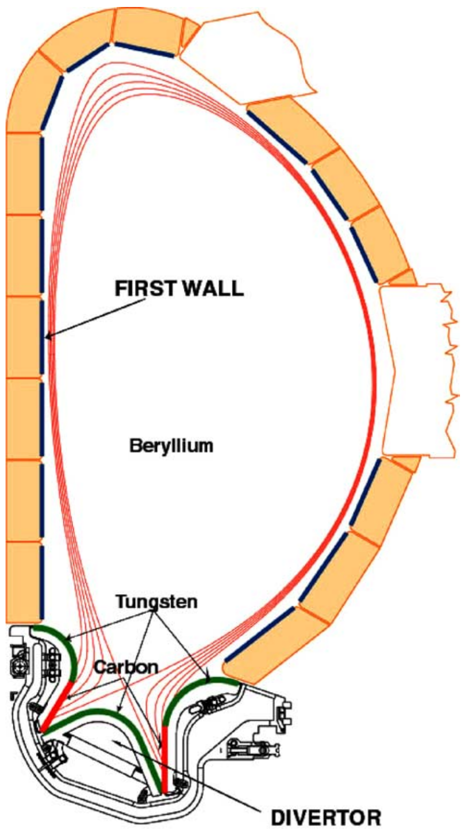

Fig. 4. Cross-section of ITER with plasma-facing components.

ITER has a built-in flexibility to allow replacement of the plasma facing components, which constitutes an essential part of the flexibility of ITER. Four change-overs of divertor modules are possible during the whole ITER operation period, enabling change of divertor material if necessary. For replacement of the plasma facing components, it would take 2 months for the limiter and 6 months for the divertor. For a full replacement of blanket modules which the first wall is attached to, 2 years could be necessary if two vehicles/manipulators are used. Further work is required to carry out detailed assessment of the change-over time of the first wall.

Acceleration of the schedule to e.g. one year would be a possibility if the replacement is carried out for a part of the first wall (e.g. half of the total area) and if two additional vehicles/manipulators are implemented.

5.3. T removal techniques

To assure environmental safety, it is important to control the tritium inventory in the vacuum vessel. The safety assessment is carried out with an assumption of 1 kg tritium in the vacuum vessel, and a goal of maximum tritium inventory in the vessel is set at 350 g [1]. Transport calculations suggest that an inductive operation requires 50 Pa m^3/s of T fuelling during a DT burn, which corresponds to 54 g of tritium for a discharge with 400 s burn. If we assume that 30% of T is retained in the vessel [30], 22 DT shots can run before the T retention reaches 350 g. If the uncertainty in the T retention measurement is 20%, a goal of maximum tritium inventory in the vessel can be set at 800 g, enabling 49 discharges of DT burn. A model calculation suggests that the tritium retention increases at a rate of 2–5 g/discharge in ITER [31], which will enable operation of 70–175 DT discharges and 160–400 DT discharges before the T retention reaches 350 g and 800 g, respectively. The build-up of tritium retention could be significantly reduced by the coverage of carbon surface by beryllium [32]. Since the most part of discharge scenario optimisation would be made in DD and initial experiments would be performed with low performance requiring low T throughput, these conditions would be acceptable at least for initial experiments.

Tritium is expected to reside in the vacuum vessel mostly in codeposited layers. A number of T removal techniques has been tested, including ICRF, ECRF, D tokamak discharges, He GDC, D GDC, D gas soak, outgas, He/O GDC, N_2 vent, disruption, PDC, boronization, baking, venting, oxidation and ablation by flashlamp or lasers [33]. A new technique for tritium removal by radiative plasma termination has been proposed [34]. An initial estimation suggests that ~10 g of tritium can be removed with one radiative plasma termination in ITER. A challenge is the removal of tritium deposited on the side of the tiles and shadowed areas, where access is difficult. An accelerated effort is required in this area to understand the mechanism of T retention, to optimize schemes for tritium removal and to extrapolate to ITER.

5.4. Operation plan and strategy for PWI investigation

5.4.1. Construction phase

A strategy for PWI investigation is shown in Fig. 5. During the ITER construction phase, research and development efforts should continue to be focused on key areas including T retention/removal, heat load due to disruption and ELMs, impurity transport, SOL transport and dust. Physics models of these processes should be developed and validated against experiments. In addition, it is necessary to test control methods of tritium retention, disruption, ELM and impurity in the present devices and make assessment of their performance in ITER. Further, it is important to make assessment of reference and alternative PFMs.

5.4.2. Operation plan and strategy during the early operation phase

During the hydrogen and early DT phases, the flexibility of the machine will be used to explore wide

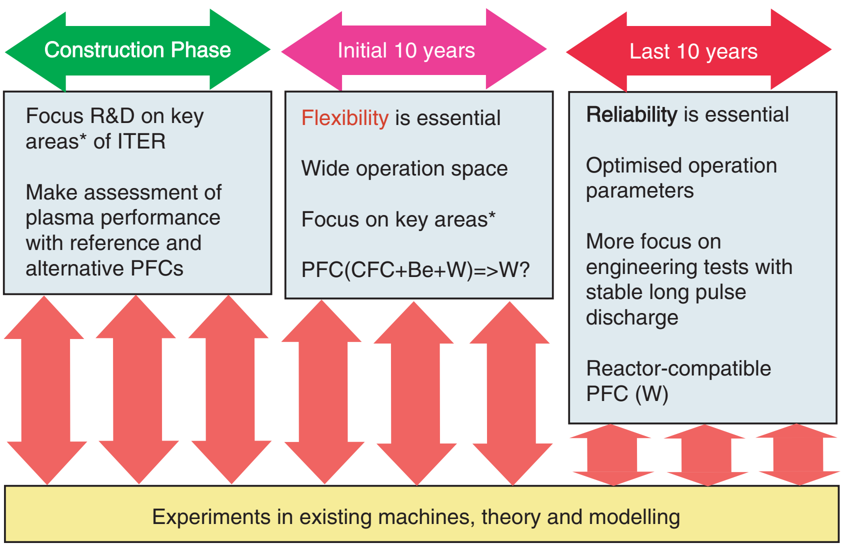

Fig. 5. Strategy for plasma–wall interaction investigation in ITER.

parameter spaces to reduce the remaining uncertainties in the key areas. The operation plan of the first 10 years includes an initial operation phase which will be conducted with hydrogen and/or helium plasmas (Fig. 3). During this phase, commissioning of various tokamak systems will be completed, and elements of reference operation scenarios will be developed. During the hydrogen phase (Table 3), most of the plasma parameters (I_p, B_t, n_e, heating power), except the neutron levels, reach the full or close to the maximum values, thus enabling experiments to be performed that can reduce the uncertainties for projection to the DT phase.

During the hydrogen phase, adequacy of heating systems for L–H transition and divertor function will be tested. Extrapolation of performance during the DT phase can then be made based on the confinement characteristics during the H-phase. The characteristics of transient phenomena, such as disruption, vertical displacement events and ELMs will be investigated, and control measures – for example, impurity gas injection for disruption mitigation [35] and pellet injection for ELM mitigation [17] – will be tested. A neural network will be trained [36,37] and tested for disruption prediction. Schemes for impurity transport control will be tested. Erosion and re-deposition of first wall and divertor materials will be investigated. Wall conditioning procedures will be developed especially for tritium removal. The quantities, composition, size and location of dust will be investigated and dust removal techniques will be verified. The investigation in these areas will be continued in the DD and DT phases.

These control methods will have been developed in the present devices, but their projection to ITER will be uncertain. These methods will be tested in ITER and further development will be made in ITER if it is necessary. Assessment will be made of the impurity control and plasma performance for the case of tungsten plasma facing components, and to establish reliable scenarios for long pulse/steady state operation. The L–H power threshold in helium-4 discharges in JET has been found 42% higher than that in deuterium plasma and approximately 40% lower than that in hydrogen plasma at the same electron density and magnetic field [38]. This makes helium discharges of interest for initial (‘hydrogen’) phase of ITER operation.

During the DT phase, a reference DT scenario will be developed by optimizing DT fuelling, fusion power, auxiliary heating power and burn pulse length. Exploration will be made in wide operation regimes to investigate burning plasmas in the inductive, hybrid and steady state regimes. Reliable scenarios will be developed for long pulse engineering tests without severe disruptions, vertical displacement events and giant ELMs.

Table 3
Capability during hydrogen and DT operation phases

	H/He*	DT
B_t/I_p	5.3 T/15 MA	←
Paux	~70 MW	73 MW → 110 MW
Pheat	~70 MW	80–120 MW
Confinement mode	(H), low shear, RS	H, low shear, RS
Power/particle control	Divertor/SOL transport, radiative cooling, He exhaust Impurity transport	
Lifetime/retention	Erosion, codeposition, H isotope removal	
Disruption/VDE	EM/heat load, run. ele., neural network, mitigation Disruption-free operation	
ELM	Heat load, Mitigation, type-II	
Dust	Measurement, removal	

5.4.3. Strategy during the later operation phase

During the later operation phase, more focus will be placed on engineering tests. However, depending on the progress made during the early operation phase, further development of the scenarios for long pulse/SS operation, and means of disruption control and impurity control may be carried out in order to reduce uncertainties in projections to DEMO.

5.5. Blanket tests

Blanket tests during the initial 10 years of operation will be included (Fig. 3). During the hydrogen phase, electromagnetic and hydraulic tests will be carried out and the effect of ferritic steel will be investigated. During the low duty DT phase, non-inductive operation scenarios will start to be developed, enabling short time tests of tritium breeding, thermomechanical tests and preliminary high-grade heat generation tests. During the high duty DT phase, non-inductive operation scenarios will be developed, which facilitates tests of on-line tritium recovery, high-grade heat generation and possible electricity generation.

6. Summary

The principal conclusions of this paper are:

(1) The primary objective of ITER is the investigation and demonstration of burning plasmas in a long pulse. The ITER operation also aims at steady state operation. ITER is an essential step toward DEMO.
(2) Projection studies based on recent results of experiments, theory and modelling have improved the prospects of achieving the goals of ITER and have opened the possibility of achieving enhanced performance.

(3) The success of ITER largely depends on the control of plasma–wall interaction.
(4) As ITER is an experimental device, flexibility of operation is important at least in the first 10 years of operation. In the later 10 years of operation, more emphasis will be put on engineering tests, which require reliable long pulse operation with a high fusion power.
(5) The strategy for the control of PWIs includes the semi-closed divertor, strong fuelling and pumping, disruption and ELM control, replaceable plasma-facing materials and stepwise operation.
(6) During the ITER construction phase, R&D should continue in several key areas including tritium retention, impurity transport, disruption, erosion, ELMs, SOL transport and dust.
(7) The ultimate plasma facing material for ITER and DEMO will probably be high Z material such as tungsten. Full implementation of a high Z material will require establishment of disruption control and impurity transport control methods.

Acknowledgment

Fruitful discussion with Dr Y. Shimomura, Dr T. Honda and Dr T. Mizoguchi of ITER International Team and Professor H. Bolt is gratefully acknowledged.

This report was prepared as an account of work undertaken within the framework of ITER Transitional Arrangements (ITA). These are conducted by the participants to the ITA: China, the European Atomic Energy Community, Japan, Korea, the Russian Federation, and the United States of America, under the auspices of the International Atomic Energy Agency. The views and opinions expressed herein do not necessarily reflect those of the Participants to the ITA, the IAEA or any agency thereof. Dissemination of the information in this paper is governed by the applicable terms of the former ITER EDA Agreement.

References

[1] ITER Technical Basis, ITER EDA Documentation Series No. 24, IAEA, Vienna, 2002.
[2] K. Itoh et al., Transport and Structural Formation in Plasmas, IOP, 1999.
[3] A.E. Costley et al., in: Proceedings of 19th IAEA Fusion Energy Conference, Lyon, 2002, CD-ROM file CT-5 and Available from: <http://www-pub.iaea.org/MTCD/publications/PDF/csp_019c/html/node54.htm#13254>.
[4] A.E. Costley et al., Fusion Eng. Design 55 (2001) 331.
[5] ITER Physics Basis, Nucl. Fusion 39 (1999) 2175.
[6] V. Mukhovatov et al., Plasma Phys. Control. Fusion 45 (2003) A235.
[7] A. Kukushkin et al., these Proceedings, doi:10.1016/j.jnucmat.2004.10.006.
[8] M. Bures et al., J. Nucl. Mater. 176&177 (1990) 387.
[9] P.R. Thomas, J. Nucl. Mater. 176&177 (1990) 3.
[10] H.H. Abou-Gabal, G.A. Emmert, Nucl. Fusion 31 (1991) 407.
[11] P. Bachmann, D. Reiter, A.K. Prinja, J. Nucl. Mater. 196–198 (1992) 865.
[12] H. Kubo et al., Plasma Phys. Control. Fusion 41 (1999) 747.
[13] A. Kukushkin et al., Plasma Phys. Control. Fusion 44 (2002) 931.
[14] G. Saibene et al., Plasma Phys. Control. Fusion 44 (2002) 1796.
[15] J. Ongena et al., Nucl. Fusion 44 (2004) 124.
[16] V. Mukhovatov et al., Nucl. Fusion 43 (2003) 942.
[17] P.T. Lang et al., Plasma Phys. Control. Fusion 44 (2002) 1919.
[18] T.E. Evans et al., these Proceedings, doi:10.1016/j.jnucmat.2004.10.062.
[19] A.C.C. Sips et al., Plasma Phys. Control. Fusion 44 (2002) B69.
[20] T.C. Luce et al., Nucl. Fusion 43 (2003) 321.
[21] M.R. Wade et al., Phys. Plasmas 8 (2001) 2208.
[22] T. Fujita et al., Nucl. Fusion 43 (2003) 1527.
[23] A.R. Polevoi et al., in: Proceedings of 19th IAEA Fusion Energy Conference, Lyon, 2002, CD-ROM file CT/P-08 and Available from: <http://www-pub.iaea.org/MTCD/publications/PDF/csp_019c/html/node170.htm#39518>.
[24] R. Dux et al., Nucl. Fusion 44 (2002) 260.
[25] H. Takenaga et al., Nucl. Fusion 43 (2003) 1235.
[26] ASDEX Team, Nucl. Fusion 29 (1989) 1959.
[27] T. Tanabe et al., J. Nucl. Mater. 283–287 (2000) 1128.
[28] H. Bolt et al., J. Nucl. Mater. 307–311 (2002) 43.
[29] K. Lackner et al., J. Nucl. Mater. 307–311 (2002) 10.
[30] P. Andrew et al., Fus. Eng. Design 47 (1999) 233.
[31] G. Federici et al., 19th Symposium on Fusion Engineering, Atlantic City, NJ, USA, 2002.
[32] K. Schmid et al., Nucl. Fusion 44 (2004) 815, and these Proceedings, doi:10.1016/j.jnucmat.2004.08.018.
[33] G. Federici et al., Nucl. Fusion 41 (2001) 1967.
[34] D.G. Whyte et al., these Proceedings, doi:10.1016/j.jnucmat. 2004.10.134.
[35] D.G. Whyte et al., Phys. Rev. Lett. 89 (2002) 055001.
[36] R. Yoshino, Nucl. Fusion 43 (2003) 1771.
[37] G. Pautasso et al., J. Nucl. Mater. 290–293 (2001) 1045.
[38] D.C. McDonald et al., Plasma Phys. Control. Fusion 48 (2004) 519.