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Journal of Nuclear Materials 363–365 (2007) 467–475
journal of nuclear materials
www.elsevier.com/locate/jnucmat

ITER and plasma surface interaction issues in a fusion reactor

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Abstract

The integration of burning plasma physics with fusion technologies in ITER is an essential step on the strategic path towards establishing the fusion energy option. Once ITER plasmas are established, the major physics parameters of the core plasma will be in the same range as those expected in the demonstration fusion reactor. However, there is a significant difference between ITER and the next generation reactor with regard to plasma surface interaction. The major differences are in requirements of fusion power, lifetime of plasma-facing components and reliability of operation. Therefore, it is essential to develop an integrated model that can accurately account for plasma surface interactions with a core plasma and can be used to make predictions for a reactor. For this development, research programs in various magnetic fusion devices, especially in ITER, will have to be integrated as well as theoretical and basic research programs executed in this area.

Keyword: ITER

1. Introduction

A treaty for the ITER Project covering construction, operation, exploitation and de-activation with a period of more than 30 years will be initialed on May 24th, 2006 by representatives of Europe, China, India, Japan, Korea, Russia and the United States of America. The ITER construction will start soon thereafter. It will take about 10 years including licensing process and the integrated commissioning of the facility [1]. The first plasma is expected around 2016. The operation will start with hydrogen plasmas. After confirming the full performance of the machine with hydrogen plasma, deuterium and then tritium will be introduced. By the end of 2020, ITER aims to demonstrate the extended burn of deuterium–tritium (D–T) plasma at a few hundred MW of fusion power where the majority of the plasma heating power will be supplied by fusion alphas. This goal will be achieved in an inductively driven plasma or in a hybrid mode plasma. The predicted fusion energy gain Q is larger than 10. It aims also to demonstrate steady-state plasma with a non-inductively driven plasma current in ITER.

The topics to be addressed in ITER include (a) burning plasma physics, (b) reactor scale plasma physics. (c) issues of plasma surface interactions in a fusion reactor and (d) reliable operation of a reactor level plasma. Once ITER plasmas are established, the physics parameters of the main core in ITER will be in the same range as those expected in a demonstration fusion reactor aiming at electricity

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0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2007.01.215

production – usually called DEMO – which is the logical next step on the development path towards commercial power production. However, there is a significant difference between ITER and DEMO. The reason for this is the difference in requirements of fusion power, lifetime of plasma-facing components (PFC) and reliability of operation.

ITER will offer opportunities for the study of burning plasmas in a wide range of parameters [2]. The major issues and questions of plasma surface interactions that will be investigated include: (1) how to accommodate flexible plasma operations in the plasma research phase of the ITER operation, (2) how to realize reliable repetitive operations in the high neutron fluence engineering test phase, and (3) how to establish methodologies for the prediction and control of plasma surface interactions in DEMO. The material choice of PFC and details of the edge plasma control might be different for (1) and (2) [3]. The initial set of plasma-facing components, i.e. beryllium (Be) for the first wall, graphite carbon–fiber-composite (CFC) for the divertor target and tungsten (W) for other areas of the divertor, has been chosen for (1). This set will avoid heavy impurity contamination as well as keep oxygen and carbon contamination at a sufficiently low level and various plasma operation modes with wide ranging parameters will be accommodated. This set will minimize the effect of disruptions and other large transitional events. The mixture of three plasma-facing materials could become a critical issue after long operation. The lifetime of the CFC target is limited and tritium retention might become a serious problem for a long operation. In the first plasma research phase, various types of operations could be studied with burn of a relatively small amount of tritium, typically 1–2 kg. However, there are still large uncertainties in relation to tritium retention and removal and these will have to be intensely investigated by using hydrogen isotopes, in parallel with ITER construction.

ITER will be used for testing nuclear components such as tritium breeding blankets and high heat flux components for reactors of the next generation. In the high neutron fluence engineering test phase, the initial set of plasma-facing components would not be appropriate. In this phase, the number of modes of plasma operations will be limited and different material(s), or a simpler set of material(s), could be chosen such as Be and W without CFC. For the topic (3), a comprehensive research program will have to be developed. The ITER Project is the most important element of this program but will be able to contribute to only parts of it and so supporting activities in laboratories will be needed.

2. ITER device and plasma-facing components [1]

ITER is designed to

- achieve extended burning inductively-driven plasmas at $Q > 10$ whilst not precluding the possibility of controlled ignition;
- aim at demonstrating steady-state operation through current drive at $Q \geqslant 5$;
- demonstrate availability and integration of essential fusion technologies;
- test components for a future reactor;
- test tritium breeding module concepts, with the 14 MeV neutron power load on the first wall $\geqslant 0.5$ MW/m^2 and fluence $\geqslant 0.3$ MW a/m^2.

The major parameters of ITER are summarized in Table 1. It is impossible to have human access once deuterium plasma operation has begun because of activation and flexibility of the machine has been incorporated in the design as much as possible. For example, all in-vessel components consist of modules which are replaceable by remote handling techniques. The first wall is attached to a shield blanket module body and the divertor target to a divertor cassette body. Therefore, all plasma-facing components can be replaced in the hot cell.

Table 1
Main parameters of ITER

Total fusion power	500 MW (700 MW)$^{(1)}$
Average 14 MeV neutron wall loading	0.57 MW/m^2 (0.8 MW/m^2)
Plasma inductive burn time at 15 MA	>400 s
Non-inductive burn time at 500 MW	>3000 s
Plasma major radius (R)/minor radius (a)	6.2/2.0 m
Plasma current (I_P)	15 MA (17 MA)$^{(2)}$
Vertical elongation (κ_{95}) @ 95% flux surface/separatrix	1.70/1.85 (1.85/2.0)$^{(3)}$
Triangularity (δ_{95}) @ 95% flux surface/separatrix	0.33/0.48 (0.45/0.55)$^{(3)}$
Toroidal field @ 6.2 m radius (10^{-5} B_T)	5.3 T
Plasma volume	831 m^3

(1) and (2) The pulse length is limited to about 200 s. (3) The plasma shifted to the outboard and has a minor radius of 1.85 m.

Modifications of divertor cassettes are also possible if necessary. Fig. 1 shows schematically an initial set of plasma-facing components.

2.1. The first wall

The first wall is covered by Be, which has low-Z and a high oxygen gettering capability. Therefore, it is the best material for obtaining a high performance of core plasma. Beryllium deposited on carbon has a favourable effect to reduce tritium retention compared with pure carbon. The design value for normal operation is $0.5\,\mathrm{MW/m^2}$ for the heat flux and 136 MW for the total power. With careful alignments and elimination of leading edges, the expected heat load is much lower than the design value in a normal operation except hydrogen neutral beam shine through in a low hydrogen plasma ($<5\times10^{19}\,\mathrm{m^{-3}}$) operation which may not be important in ITER. A typical surface temperature during steady-state burn is 170–270 °C.

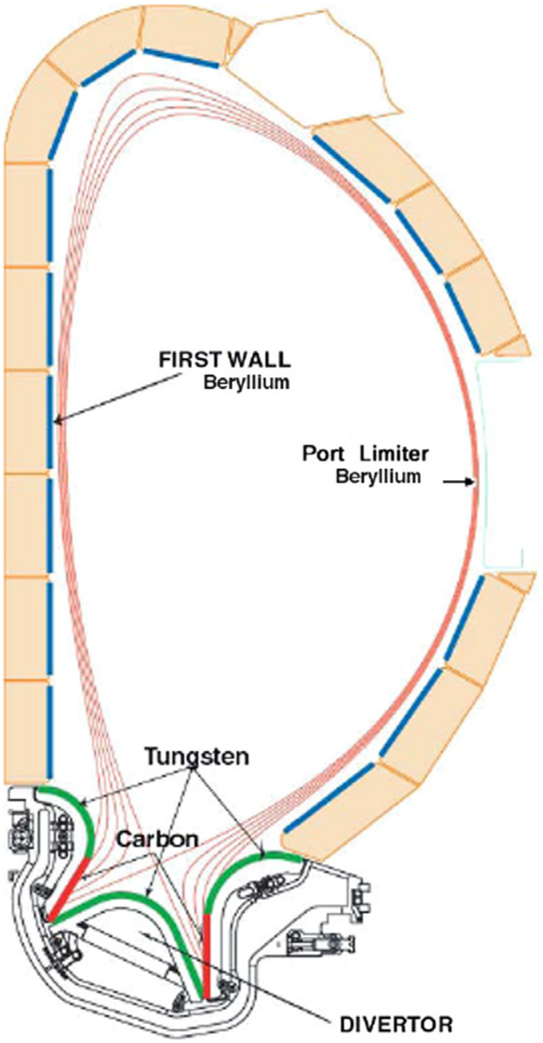

Fig. 1. Cross-section of ITER with plasma-facing components.

Transient heat load due to disruptions and/or vertical displacement events could be much higher than the critical heat load inducing melting of Be ($20\ \mathrm{MJ\ m^{-2}\ s^{-0.5}}$). These events will limit the lifetime of the first wall. The plasma-facing surface of Be can rise up to only 1350 °C and the structure of the heat sink is well protected. This critical heat load for melting of W and sublimation of CFC is about 55 and 40 $\mathrm{MJ\ m^{-2}\,s^{-0.5}}$, respectively, which is also not sufficiently high but the lifetime is longer for very short heat pulses. In a case of a large heat pulse with a duration of order of 1 s due to a slow disruption or a vertical displacement, the surface temperature could become very high (∼3400 °C) and the backside temperature of these armour attached to the panel structure could become also high. This may lead to more serious damage of the structure of the first wall panel than on the plasma-facing surface. In this sense, Be is a good sacrificial material to prevent a possible damage to the structures as well as a good oxygen getter and low-Z. As the sacrificial material, the Be first wall will be relatively easily damaged due to a large transient heat load. However, JET results let suggest that the consequences of the damage are relatively minor for plasma operation and performance [4]. In order to keep a reasonable life time of the first wall, it is essential to reduce frequencies of disruptions and to mitigate their effects.

2.2. Start-up limiters

Two start-up limiters covered by Be are installed in equatorial ports. The heat flux must be limited to around $5\,\mathrm{MW/m^2}$. The edge of the limiter is not strong enough to withstand large ELM heat loads as well as disruption heat loads during high power operation. Therefore, the port limiter is retracted to the first wall position after the formation of the divertor configuration within several seconds. The limiters are reinserted for use during the current ramp-down phase.

2.3. Divertor baffle and dome

The divertor baffle and dome are covered by W. These areas have intensive interactions with neutral hydrogen particles. Tungsten has a low sputtering rate, which means low erosion and long lifetime. Impurity contamination from these areas is not expected to be serious. The design value is 5 MW/m^2 for steady-state heat load and 20 MW/m^2 for 10 s duration. A typical surface temperature during steady-state burn is 150–250 °C .

2.4. Divertor target

The divertor target which is in contact with the divertor plasma is made of CFC because of the following reasons: (1) good thermo-mechanical properties and the lack of melting under transient power fluxes and (2) core radiation loss is low, while divertor radiation loss is effective in maintaining semi-detached operation.

However, there are serious issues with the use of graphite, especially tritium retention and lifetime restrictions due to the large erosion yield of carbon. The lifetime of the divertor target is expected to be a few years in the initial phase of operation, or a few thousands reference shots with 400 MW of fusion power and 400 s of the pulse length, and limited by erosion. Very frequent pulse heat loads due to large ELMs may shorten this lifetime. In order to keep this lifetime, the heat load per ELM, assuming ELM duration of –0.5 ms, must be limited to 1.2 MJ/m^2 [5]. This heat load corresponds to an energy loss per ELM of 12 MJ with the profile shown by the green broken line in Fig. 2 [6]. The disruption heat load might be much higher than this value but due to the much reduced numbers of events, the erosion by disruptions should be acceptable. The estimated allowable number of disruptions is 100–1000 depending on assumptions [5].

Tritium retention due to co-deposition with eroded carbon is one of the most critical issues and subject to large uncertainties. This could be solved by using W which might, however, limit plasma operational flexibility because of potential problems of W contamination of the core plasma, crack penetration deep into the bulk along grain boundaries after the grain grows owing to heat cycles such as ELM heat cycles, and/or enhanced melting or evaporation due to irregular surface leading to melting during disruptions. For this reason, the W target is not suitable for the investigation of all the operational scenarios and windows which is fundamental in the early phase of the ITER operation. After establishing reliable operation modes, it would be better to install metallic targets such as W in order to reduce tritium retention and to ensure a long lifetime of the divertor target as well as to demonstrate the target suitable for DEMO.

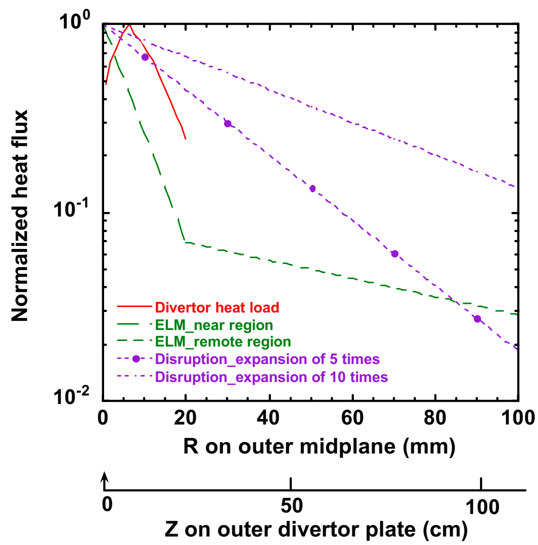

Fig. 2. Expected heat flux profile on the divertor target for a normal phase (solid line), ELM (broken line) and disruption (dotted line) [6]. '0' represents the separatrix position.

The design value for steady-state heat load is 10 MW/m^2 which in principle corresponds to a total heat load of 100 MW flowing to the divertor plasma to the target, but due to special variations, the maximum value is probably lower, and 60 MW is taken as a reasonably conservative value. For 10 s pulse length, 20 MW/m^2 is allowed. The cooling capability for the total surface heat load to the divertor plasma-facing components is 136 MW. A typical surface temperature is about 1000 °C at the striking point with a partially detached plasma during a steady-state burn, and about 200 °C or higher away from the striking point.

3. Plasma surface interaction issues in ITER operation

3.1. Limiter phase

It takes about 20 s before achieving the divertor configuration. The maximum plasma current with the limiter configuration is around 4–7 MA depending on the scenario. The total heat load to the scrape-off layer is estimated to be about 3–4 MW due to ohmic heating. One limiter cannot supply a large enough surface and therefore, two limiters will be installed. The alignment of these limiters is a critical factor. Heat flux profiles on two limiters are calculated by using a three dimensional transport code [7]. The result shows a few mm of relative limiter misalignment is acceptable. The maximum heat load to limiters is less than about 5 MW/m^2 which is consistent with the current design.

3.2. Hydrogen/helium and deuterium operation phase

During the first few years, ITER will be operated with hydrogen and/or helium. During this period, all systems will be commissioned up to their maximum performances except systems directly relating to D–T burn. It is desirable to reduce uncertainties of plasma surface interactions as much as possible in this phase because human access into the vacuum vessel is only possible in this phase. However, H-plasma behavior in H-mode operation is significantly different from that of D-plasma or DT-plasma. Therefore, uncertainties will remain. Then deuterium will be introduced. The edge plasma behavior of D-plasmas with a high heating power, e.g. 60 MW, would be similar to that of D–T plasmas with a low fusion power, e.g. 200 MW. During this phase, remaining uncertainties of plasma surface interactions must be significantly reduced. Control methods of plasma surface interactions and large transient events such as vertical displacement events, disruptions and large edge localized mode (ELM) will have to be established so that possible erosion and damage could be minimized in D–T burning plasmas.

3.3. Burning phase

After establishing deuterium plasmas, a small amount of tritium will be introduced and it is assumed that demonstration of D–T burning with a short pulse will be relatively easily achieved. The fusion power gain Q and the burn duration will be increased gradually. In order to achieve burning plasmas with dominant alpha-particle heating, Q must be 10 or higher. Operation conditions depend largely on confinement. If the confinement is better, $Q = 10$ can be achieved at a lower fusion power with a lower plasma density. It also depends on the heating methods. $Q = 10$ would be achieved at around 200 MW with a plasma current of 15 MA, a confinement enhancement factor of $H_{H98(y,2)} = 1$ and a plasma density of $6.5 \times 10^{19}\ m^{-3}$. The total heating power in this case is 60 MW and the power across the separatrix is about 45 MW. These values are only two times higher than in the present large tokamaks whose major radius is about a half or so of that of ITER. The heat flux to the divertor is lower than the acceptable value, i.e. 60 MW. The heat pulse due to a Type-I ELM is estimated to be 14 MJ, which is slightly higher than the acceptable value, i.e. 12 MJ. Assuming 40% of power loss across the separatrix is due to the ELM, the loss is 18 MW and the natural frequency of the ELM is 1.4 Hz. In order to reduce the heat pulse down to a sufficiently low value, the frequency must be higher than 2 Hz. This could be achieved by fuel ice pellets injected from the high field side. The ablation position of fuel pellets is deeper than one half of the plasma pedestal width (Fig. 3) which is consistent with the requirement to induce ELMs. Normally, the frequency is higher than 2 Hz and so the heat pulse due to ELMs should not be a serious problem in this low power operation. Fundamental burning plasma physics with dominant alpha-particle heating, and methods of control of burning plasmas, including edge and divertor plasmas, could be studied in this kind of operation with less demanding conditions to the plasma-facing components. The fusion power will be gradually increased and various operation modes will be expanded. In these operations, predictions from

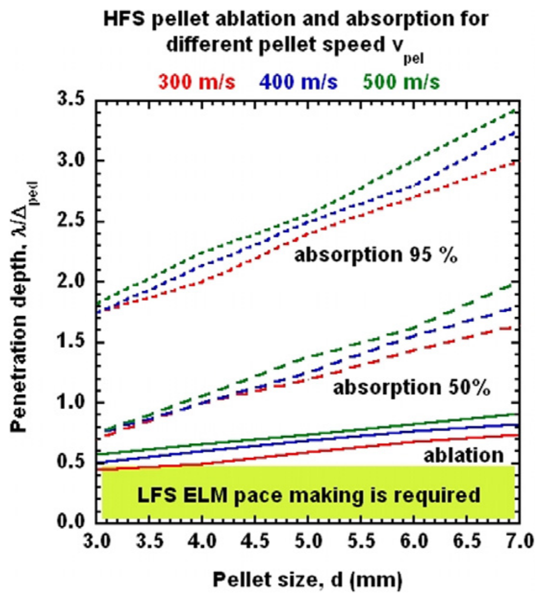

Fig. 3. Ablation or absorption depth (λ) normalized by plasma pedestal width (Δ_{ped}) of a deuterium ice pellet injected from high toroidal field side with a plasma current of 15 MA and fusion power of 400 MW.

the current knowledge would not be accurate and controlling plasma surface interactions will become more challenging. In the course of these studies, more knowledge will be accumulated which is needed for future development of the operation space in ITER as well as for the design of the next generation reactor.

3.4. Large transient events

Plasma disruptions and vertical displacement events will be induced during the investigation of the possible operation space. It will be unavoidable to have some sublimation or melting of plasma-facing materials. These events will reduce the efficiency of the operations. It is essential to develop reliable methods to avoid or to mitigate these events as much as possible but without losing operational flexibility in the plasma research phase. It is also necessary to develop a reliable and efficient method to recover the wall condition and to achieve a normal operation. Research in this area is ongoing in existing devices but still further effort is needed to develop the capability of predictions of these events and their control methods. Type-I ELMs could also be large transient events and it is essential to develop scenarios with small ELMs which do not induce sublimation or melting of the plasma facing materials.

3.5. Issues associated with long operation

3.5.1. Tritium retention

Retention of fuel gas is one of the classical issues in a magnetic fusion device. In the ITER safety analysis, 1 kg of mobilisable tritium is assumed in the vacuum vessel in addition to tritium in the primary vacuum pumps. Analysis with conservative assumptions shows a sufficiently large safety margin. Therefore, the maximum inventory of tritium in the vacuum vessel could be 1 kg or possibly a higher value from the safety point of view.

In some experiments with CFC limiters, a very high retention rate (0.4–0.5) has been observed. Similar rates have been also observed in a divertor tokamak. In a paper [8] of JET, retention rates of 0.1–0.03, depending on divertor configurations, are reported. By assuming these retention rates in ITER, 1 kg of tritium will be retained after injecting 10–33 kg which corresponds to 200–660 shots of the reference burning operation at 400 MW with duration of 400 s. Assuming a few cleaning operations, this would be marginally acceptable for the first few years of initial D–T operations. But this is not reasonable for the following experimental periods in which 2000–3000 equivalent shots are planned per year. In JT-60U experiments, saturation of retention is observed after injecting fuel particle of $\sim 2 \times 10^{22}$ in a long pulse operation with a high power. This is explained by the operation temperature and by the geometry and the small amount of carbon dust due to good alignment of targets. If this is the case, a more optimistic situation would be expected for ITER [9]. However, these largely scattered values in different machines and with different divertor configurations have not been well explained and analysis also gives large scattering [10].

Carbon deposition between gaps of plasma-facing components and/or sublimation of carbon due to large transient heat loads such as large ELMs, can increase the retention of tritium and so such events should be avoided as much as possible as mentioned in Section 3.3. In ITER, the first wall will be made of Be which will be eroded and flow into the divertor. The tritium retention in the redeposited Be layers may be largely reduced [11]. Optimization of the geometry of the divertor is also important. Carbon deposition is observed largely in line-of-sight from the place of origin [8]. Therefore, it is important to avoid line-of-sight into the under side of the dome so that deposition areas could be easily exposed to plasma discharges (both tokamak and conditioning discharges) and/or other possible cleaning methods. This study is ongoing with the new simulation result of the divertor which requires a smaller conductance to the underside of the dome [12].

Certainly, the tritium retention is one of the most critical issues in ITER and requires more effort to reduce uncertainties. It is also important to optimize the divertor design to reduce the retention and to facilitate the removal of the retained tritium. Work on both aspects is ongoing.

3.5.2. Other issues

It is essential for the high neutron fluence test as well as for the future reactor to have a long lifetime of the plasma-facing components, especially the divertor target. After developing reliable operation modes, a metallic divertor target such as W should meet this objective. Further study is needed on high-Z impurity control and on the avoidance of deep melted layers which may cause serious adverse effects by creating irregular surfaces.

Carbon and metallic dust will be produced during normal operations and transient phenomena. The amount of activated dust must be limited. Another issue of dust is its potential reaction with leaked water and production of hydrogen. This hydrogen, or dust itself, is potentially a course of explosion if there is also air leakage into the vacuum vessel. ITER has been designed in such a manner in order to prevent explosion or to avoid serious potential consequences of explosions. However, it is important to understand the characteristics of dust, to measure and to remove it. It is also important to avoid degradation of in-vessel diagnostic components, such as mirrors, due to dust, material deposition and erosion.

A mixture of materials between carbon, beryllium and tungsten may occur after long operations and could lead to effects not presently considered. For example, it has been pointed out that a metallic alloy of W and Be has a low melting temperature. Further investigation will be needed to understand what effects will occur, how serious they will be and when they may appear.

4. Plasma surface interaction issues in a next fusion reactor

The central programmatic aim of ITER operations is the preparation of the physics basis for a next fusion reactor, e.g. DEMO. No concrete design for it yet exists. However, it is generally understood that about 1 GW of the net electrical power will be necessary. It is expected that the plasma current and plasma size will be similar in ITER and DEMO. However, compared to ITER, DEMO will need higher fusion power P_{fus}, higher Q and higher operational reliability. To achieve higher P_{fus}, a higher plasma pressure (higher $\beta \times B^2$), a higher density and higher radiative cooling will be required. To increase Q, a higher normalized beta β_N, and, in steady-state regime, higher bootstrap current fraction f_{BS} are necessary. To improve operational reliability, large plasma perturbations, including disruptions and Type-I ELMs, should be avoided.

Physics parameters of the core plasma such as temperature, density, collisionality, normalized Larmor radius, ratio of the required normalized beta to the ideal normalized beta and the required confinement enhancement factor HH are very similar between ITER and DEMO [13] as shown in Table 2. The most distinctive feature of a DEMO fusion reactor is a much higher heating power (alpha pulse additional heating), typically 500 MW or higher. Since the reactor divertor target may be only slightly larger than that of ITER, the technological constraints for power handling on the plasma-facing component will limit the power to around 100 MW or less. A substantial fraction, i.e. about 80% or higher value, of the plasma exhaust power must therefore be distributed over the plasma-facing surfaces by plasma and divertor radiation with impurity seeding, or other advanced high-heat-flux handling techniques should be developed. Demonstration of very high radiation fraction

Table 2
Typical parameters of ITER and DEMO [13]

	ITER			Example of DEMO
	Reference operation	High power	Steady-state	
Plasma current (MA)	15	17	9	16.7
Plasma volume (m^3)	830	830	830	940
Fusion power (MW)	400	700	350	3000
HH factor	1	1	1.3–1.4	1.3
Temperature $\langle T \rangle$ (keV)	8–9	9–10	11–12	17
Normalized Larmor radius at pedestal $\rho_i^*(10^{-3})$	1.4	1.5	1.6	1.0
Collisionality at pedestal ν^*	0.039	0.034	0.028	0.015
Normalized beta β_N (β_N/ideal β_N)	1.8	2.2	2.6–2.8 ($\leqslant$0.8)	4.3 (0.78)
Density $\langle n \rangle$ (10^{19} m^{-3})	10	12.3	6.7	11.7
Total heating power (MW)	120	175	140	660
Radiation loss from core (MW)	33	55	45	170
Acceptable power to target (MW)	60	60	60	100
Radiation power in edge and divertor (MW)	60	85	75	390
Net operation length (days/year)	5–15			280

$\rho_i^* = 1.02 \times 10^{-4} \frac{\sqrt{MT_i}}{aB}$, $\nu^* = 1.37 \times 10^{-16} \frac{n}{T_i^2} \frac{qR}{\varepsilon^{1.5}}$.
T_i (eV), n (m^{-3}), B (T), a (m), R (m), q and ε at edge (near pedestal).

scenarios with high energy confinement will be an important item of the ITER experimental programme. ITER has a capability to test handling of about 200 MW (about 100 MW of fusion power pulse about 100 MW of additional heating power). This would permit a study of compatibility of radiative cooling and high performance core plasma in conditions relatively close to those expected in DEMO. But the condition is still far from that in DEMO.

Such simulation experiments in ITER will be indispensable to develop and to validate an integrated tokamak prediction code which would be the most important final product for the next step. In parallel, many elements of this code relating to plasma surface issues can be studied in devices other than ITER. The present tokamaks, especially AUG with full W wall and JET with ITER like wall, and the new generation of super-conducting tokamaks, such as EAST, KSTAR, STT-1 and JT-60SA, are essential to study these elements by selecting operation conditions for each research as well as parallel theoretical and basic research. Important elements in the plasma surface interaction area includes: (1) compatibility of high-Z material with high performance plasmas required in a reactor, (2) physics of scrape-off layer and divertor plasmas especially the effect of neutral–neutral collision and impurity transport, (3) optimization of radiative cooling at the core, periphery, edge and divertor, (4) control of large transient heat and particle loads, (5) steady-state heat and particle loads to plasma-facing surfaces including plasma density blobs, and (6) material selection especially from the following points: limit the tritium retention and permeation, long lifetime, compatibility with plasma, and avoid problems of material mixture.

By integrating the results of these studies as well as core plasma studies, a comprehensive simulation code will be developed. This will be used to predict reactor plasmas and plasma surface interaction for the next generation reactors.

5. Summary

Construction of ITER will start soon and the first plasma is expected to be obtained around 2016 and an extended burn of D–T plasma with a few hundred MW of fusion power by the end of 2020. It will be possible to achieve burning plasma with dominant alpha-particle heating at a fusion power of about 200 MW with a similar heat load condition of the present large tokomak experiment. Once this is achieved, the operation space will be gradually expanded. Various operation modes with higher fusion power and higher performance will be developed by controlling plasma surface interactions. In the course of these studies, knowledge of plasma surface interactions and their control will be built up. However, there is a significant difference between ITER and a next fusion reactor with regard to plasma surface interactions because of the large difference in requirements of the total heating power, lifetime of plasma-facing components and reliability in operation. Therefore, it is necessary to develop a tool to predict accurately the performance of a reactor core plasma and edge plasma including plasma surface interactions. This will be the most essential final product of the ITER Project. It will require the integration of results of many other researches in addition to the ITER program.

Acknowledgements

Fruitful discussion with Drs M. Shimada, M. Sugihara, A. Polevoi, G. Federici, V. Mukhovatov, A. Costley and A. Sakasai of the ITER International Team, Dr M. Kobayashi of NIFS and Dr S. Nishio of JAEA is gratefully acknowledged. This report was prepared as 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, India, 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.

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