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ITER research plan of plasma–wall interaction

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ABSTRACT

This paper describes an important part of ITER Research Plan, on plasma–wall interaction (PWI). In order to maximize the flexibility of the machine during the initial operation (H and D phases), CFC will be used for the targets. Tungsten will be used for the other plasma-facing components of the divertor. In order to minimize the tritium retention, tungsten will fully cover the divertor targets before the DT phase. Extrapolation of heat loads on plasma-facing components (PFCs) during disruption and ELMs to ITER parameters indicates serious consequences of these phenomena. Therefore schemes for prediction and mitigation or avoidance of these phenomena need to be developed during construction and demonstrated in the early phase of ITER operation. T-retention and dust have important impacts on safety. Therefore the methods of measurement and removal of tritium and dust must be developed during construction and demonstrated in the early phase of ITER operation.

1. Introduction

The plasma–wall interaction (PWI) has made a significant progress during the last three decades. Understanding of PWI is proven to be essential to improve the performance of the plasma. Good reviews have been published in this area [1–3]. Based on the understanding developed so far, the projection to ITER is made, showing the critical role of PWI in ITER operation [3,4].

The ITER Research Plan (IRP) is a component of the new ITER Baseline Documentation. The ITER Research Plan provides a framework linking and integrating the current priorities of the programme with the preparation for future commissioning and exploitation. A major aim of the Research Plan is to integrate the researchers in the Members' fusion communities into the planning and preparation for ITER operation: encourage active participation in current R&D activities required to prepare for successful implementation of ITER operation and exploitation. This paper aims to present the outline of the research plan related to plasma–wall interaction. Section 2 discusses the PFC development strategy, Section 3 near-term R&D issues to be addressed in the next ~3 years and Section 4 longer-term R&D issues. The near-term R&D issues are driven by Procurement Arrangements, particularly with confirmation of the target material for the initial DT phase. The longer-term R&D needs are mostly driven by the preparations for operation and exploitation. The R&D issues discussed in this paper were stimulated by the discussion at ITPA meetings and conferences and papers relevant to this field. These R&D issues will be revised in future as more progress is made in experiments and analyses. The R&D can be carried out through voluntary work coordinated through ITPA and ITPA/IEA cross-machine experiments and Physics Tasks agreed between Domestic Agencies and ITER Organization.

2. The PFC development strategy

Fig. 1 outlines a tentative ITER operation schedule during the first 10 years. It consists of hydrogen phase, deuterium phase, DT phase (inductive) and DT phase (non-inductive). The need to provide flexibility in plasma operation while meeting the Regulatory Requirements is a unique aspect of the ITER device. To address key considerations, i.e. to maximize the flexibility during the start-up phase and to minimize the tritium inventory during the DT phase, a new strategy has been adopted for ITER operation:

(1) plasma operation will start with Be/CFC/W distribution of PFCs as previously foreseen;
(2) a change to a full W-divertor will be carried out before DT operation (possibly before D operation).

This strategy has been incorporated in the IRP, assuming that H-mode access is possible only in deuterium. It is needed to establish reliable ELM and disruption mitigation, and the time of development determines the time at which the divertor is replaced. It is assumed that the changeover can be accomplished in ~6 months.

This strategy allows time to learn how to mitigate ELMs and disruptions since CFC is more robust to type I ELMs and disruptions. This strategy also enables characterisation of hydrogenic retention for CFC in H-mode. Physics programme with W-divertor ideally

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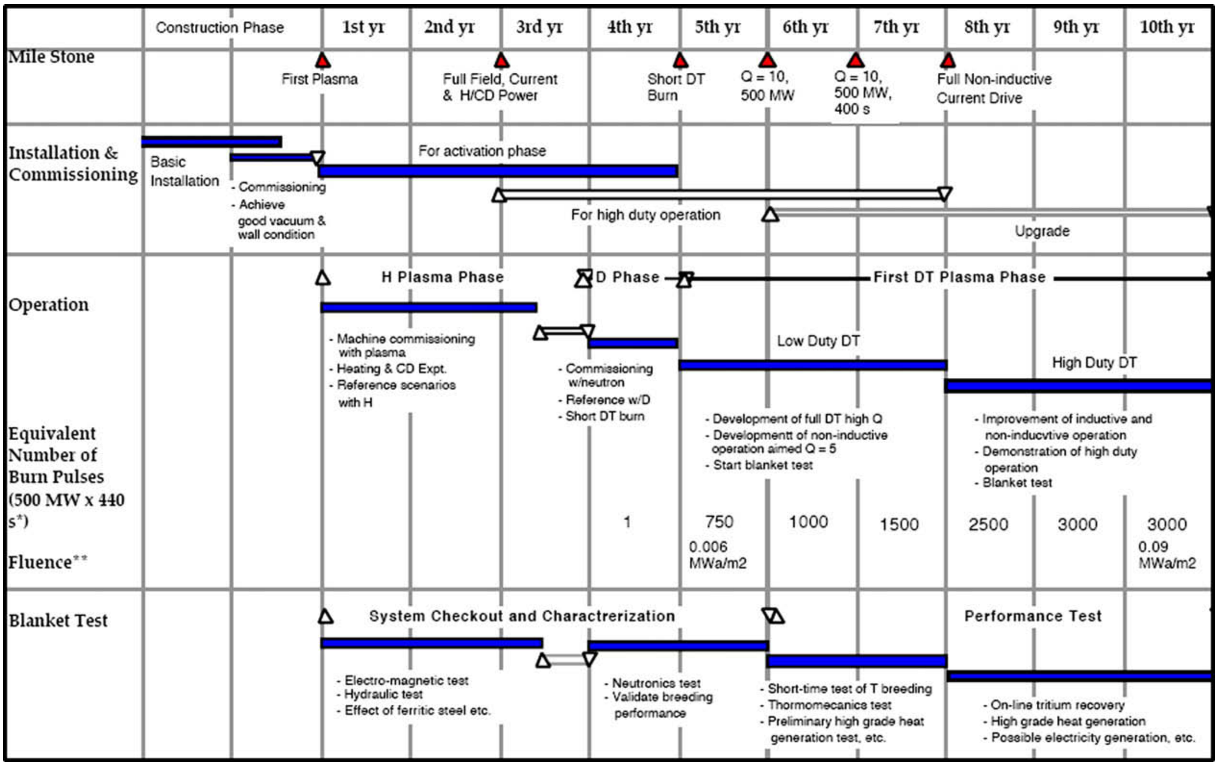

Fig. 1. ITER operation schedule during the first 10 years (tentative).

can start at the end of CFC, e.g. already with knowledge of how to ameliorate the ELMs and mitigate disruptions.

Due to the uncertainties of the operation with tungsten target, ITER retains the option of running DT discharges with CFC targets. In-vessel removal of T is possible by oxidation but co-lateral effects (e.g. recovery of the discharge conditions), safety issues and reprocessing of tritiated water (DTO) remains to be evaluated.

3. Near-term R&D issues

3.1. T-retention control

As a nuclear device, tritium retention control is a major subject for ITER. Tritium retention inside the vacuum vessel should be kept lower than 1 kg. With consideration of measurement uncertainty and the retention in the cryopumps, the tritium level in the vacuum vessel should be kept under ~700 g. Recent estimation [5] suggests that this level of tritium retention could be reached after $\sim 5 \times 10^5$ s or 1250 discharges, each with 400 s DT burn time, if tungsten is used for the whole divertor PFC and beryllium for the first wall, and after $\sim 10^5$ s or ~250 such discharges if CFC is used for the divertor targets, tungsten for the baffle and dome and beryllium for the first wall. However, the uncertainties in extrapolation are large due to uncertainties in; carbon chemical erosion for ITER divertor conditions, wall fluxes during the steady state, possible role of beryllium deposition suppressing chemical erosion under D + Ne (Ar) and role of ELMs and disruptions.

Recently, a strategy for T-retention (and dust – see below) control [6] has been accepted into the ITER Baseline. Part of the strategy calls for a 'good housekeeping' approach [7] in which a variety of techniques are employed to reduce the accumulated in-vessel tritium. Two of these are the use of isotope exchange and glow discharge conditioning, the latter only possible in ITER on a regular basis (e.g. inter-shot) with ICRF or ECRF conditioning. Such conditioning may also be required for ease of discharge start-up. Improved definition of requirements for RF conditioning is urgently called for, since IC antenna Procurement Arrangement is scheduled in 2009. There are a number of outstanding issues relating to the use of this technique on ITER and efforts are urgently required to advance the level of understanding and ensure that it will be a viable option. In particular:

- RF wall conditioning.
  - The feasibility of using the ITER RF heating systems for wall conditioning without damaging the antenna and other components of the machine.
  - The range of parameters for RF heating for conditioning (frequency, power, gas composition, gas throughput, poloidal fields to control the wetted areas).

- T-retention control.
  - The feasibility of removing tritium retained during burning plasma phases by running D only phases at the ends of DT discharges.
  - The feasibility of removing tritium from co-deposited surfaces by radiation flushing of wall surfaces using mitigated disruptions.
  - The feasibility of oxidation to remove tritium from co-deposited surfaces.
  - The rate of isotopic exchange between H, D and T on the co-deposited surfaces formed during the hydrogen and deuterium phases.
  - The contribution of castellation structure and edges on tritium retention and the effects of tritium removal schemes on the tritium trapped there.
  - Development of diagnostics for in-vessel tritium measurement.
  - Evaluation of tritium removal efficiency at divertor baking at 350 °C for Be, W, C and mixed material.

3.2. Tungsten

As mentioned in Section 2, the baseline research plan calls for the installation of full tungsten divertor before the start of DT operation. Since the experience with PFC made of tungsten and other high-Z materials is limited, and the required R&D programme needs to be completed around 2013 to enable the Procurement Arrangement issued in early 2016, an accelerated research is required to assess the scientific feasibility of full tungsten divertor, in the aspects including:

- Evaluation of the operation range for ohmic and additionally heated X-point ramp-up/ramp-down with high-Z divertor targets including impurity seeding.
- Development of understanding of tungsten behaviour in the edge, SOL and at the PFC surface.
  - Evaluation of expulsion of W impurities from the pedestal under the action of ELMs compared with the influx provoked by the ELM-enhanced erosion.
  - The influence of ELM-induced impurity influxes on plasma performance.
  - The effects of W nanostructures under hydrogenic and helium plasma bombardments on erosion.
  - The effects of castellation structure and divertor target damage on device operability and operational space.
  - Properties of mixed-material surface PFCs under ITER-relevant loads and mixed plasma species.
  - Evaluation of tungsten dust and wavy surface layers after disruption, their impact on operation and the possible schemes for clean-up.
- Modelling of divertor and edge including the effects of tungsten impurities.
- Assessment of impurity production by use of ICRH with high-Z PFCs and the role of sputtering on the high-Z targets connected to antenna shields along field lines.
- Evaluation of the permeability of tritium through tungsten at temperatures relevant to divertor targets and the effect of neutron irradiation on T-retention.

3.3. Dust

Like T-retention, a strategy to deal with in-vessel dust accumulation [6] has recently been accepted into the ITER Baseline. A significant effort is required in the next few years to improve understanding of the main mechanisms of dust production, both in steady state and during ITER-relevant transients for all materials (C, Be, W). In the next 2–3 years it is important that progress be made in as many as possible of the areas listed below:

- Development of understanding of dust production mechanism and dust transport.
  - Dust production rates from solid Be and deposited Be layers of thicknesses expected in ITER on both W and CFC substrates under heat loads appropriate to disruptive and ELM driven transients (both for controlled and uncontrolled ELMs and for natural and mitigated disruptions). Characterisation of ejection velocities and size distributions of dust. Benchmark of dust size distributions and velocities using numerical simulation.
  - Link gross erosion measurements with dust quantities to establish the conversion factor from erosion to dust production.
  - Cross-machine studies of deliberate dust injection to investigate dust launch velocities and subsequent transport. Benchmarking against dust transport models.
- Assessment of dust removal with tokamak discharges.
  - Exposure of tokamak generated deposits (carbon in the short term before JET operation with the ILW) to ITER-relevant transient heat loads and analysis of generated dust.
  - Study response of deposited layers to flash heating (as in mitigated disruptions).
  - Control of dust at hot surfaces by e.g. sweeping the separatrix over the hot surfaces.
- Development of dust diagnostics.

3.4. Heat-fluxes to plasma-facing surfaces

Revised heat load specifications, based on extrapolations from results obtained in operating devices, for SOL and divertor parallel power fluxes during steady state and transients [4] have recently been accepted into the ITER Baseline. However, since the uncertainty is large and the physical understanding is under development in many cases, improved characterisation of first wall/divertor heat loads, particularly during ELM/ disruption mitigation is needed.

Longer timescale transients (i.e. durations of seconds), which exceed the steady-state heat exhaust capability of the active cooling by even factors of 3–4 cannot be tolerated for more than a few seconds at most. On the question of the short timescale transients, outstanding issues remain concerning controlled ELM variability and first wall loading during mitigated disruptions. In order to improve confidence in the new heat load specifications as the first wall design is finalised (next 1–2 years), further studies are requested in tokamaks which address the following areas:

- Characterisation of ELM heat loads.
  - Cross machine comparisons of ELM statistics, including controlled ELMs (waiting times between ELMs, poloidal distribution and sizes of ELM wall footprints), for evaluation of natural variability in wall and divertor heat and particle fluxes. Continuation of efforts to scale ELM radial velocity with ELM size, device size and other dimensional/dimensionless pedestal parameters.
  - Improved characterisation of divertor power loading during ELMs mitigated or suppressed by resonant magnetic perturbations and comparison with models.
- Characterisation of inter-ELM SOL transport.
  - Further measurements of inter-ELM far SOL cross-field heat and particle fluxes and effective convective velocities and continued SOL turbulence modelling (extension to 3D) towards predictive capability for ITER.
  - Begin studies of transient divertor reattachment, particularly in response to loss of gas or extrinsic impurity seeding or due, for example to confinement transients such as H–L transitions, ITB collapses etc, including the timescales for reattachment and the distributions/magnitudes of divertor radiation.
  - Resolution of the discrepancy between the numerical codes and experiments on the detachment conditions.
  - The power e-folding length scaling for extrapolation to ITER including isotope scaling and in high power regimes.
  - Further investigation of SOL limiter profiles during start-up and ramp-down in ITER-relevant scenarios and the model validation including scenarios with variable numbers of limiters for inner or outer wall start-up.
- Evaluation and extrapolation to ITER of the degree of toroidal asymmetry in the intense radiation flash during mitigated disruptions in the case of a single injection point.

3.5. Erosion

The ITER first wall components will be shaped, to avoid plasma exposure of PFC edges, permitting steady-state and transient heat fluxes to be tolerated and to open the possibility of using the first wall PFCs as limiters during the initial and final phases of ITER discharges. Shaping of components automatically leads to regions of erosion and re-deposition. It is thus important that efforts begin in operating devices to investigate the effect of first wall shaping on local erosion and re-deposition:

- Cross machine comparisons of main wall co-deposition on limiting surfaces.
- Development of local models taking into account surface shaping and attempting to predict the observed deposition patterns for both steady (diverted operation) and during limiter start-up/ramp-down phases.
- Estimates of the quantities of tritium that could be trapped in these first wall deposits.
- Efforts should also continue to characterise outer and inner divertor erosion, the movement of impurities from main chamber to divertor and from divertor to divertor, including a full description of drift physics in the modelling packages and benchmark against experiment.

4. Longer-term R&D issues

The longer-term R&D needs are driven more by the preparations for operation and exploitation:

- Contribution to development of integrated scenarios consistent with PWI requirements.
- Evaluation of the influence of all-metal walls on plasma–wall interaction phenomena in ITER-relevant regimes.
  - Evaluation and control of impurity ingression especially at low/modest densities during start-up/shut-down or during steady state.
  - Development of operation scheme to accelerate recovery of wall conditions (wavy surface and dust) after disruption.
  - Evaluation of impurity accumulation during steady-state operation and the measures for its control.
- Development of improved modelling capability for PWI phenomena.
  - Evaluation of the effects of SOL flows, drifts, blobs, kinetics, carbon hydrides.
  - Inclusion of PWI modelling into integrated modelling.
- Improved disruption control.
  - Improved prediction of disruption e.g. with real time stability calculation.
  - Incorporation of the stability evaluation in the control algorithm.
- Improved ELM control.
  - Development of ELM-free improved confinement regimes.
- Inclusion of PWI control into integrated plasma control scheme.
- Development of global measurement schemes for tritium retention and dust.
- Development of fueling methods with better penetration into the plasma (e.g. supersonic gas or compact toroid injection).

5. Summary

Although the PWI field has witnessed a significant progress in the last three decades, it requires further research efforts since the uncertainties are still large and PWI processes will play a critical role in ITER. In this paper, near-term and longer-term R&D issues are discussed. The near-term R&D issues are driven by Procurement Arrangements and particularly in relation to the selection of divertor target material during the initial DT phase. The near-term R&D issues include T-retention control, tungsten, dust, heat flux on PFC and erosion. The longer-term R&D issues are driven more by the operation and exploitation of ITER.

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