Review of ITER Physics Issues and Possible Approaches to their Solution

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Abstract. ITER will demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes by nominal operation of 500 MW fusion power for 400 s. With its all superconducting coil technology, ITER will be capable of moving toward steady-state, high gain operation for fusion power. While the physics basis for ITER's nominal inductive operation is well established, the physics basis for steady-state is currently being developed [1]. Ongoing tokamak research programs must continue to contribute strongly during ITER construction to various physics issues whose resolution will improve both the inductive and steady-state operation of ITER.

1. Introduction

ITER will demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes by nominal operation of 500 MW fusion power for 400 s. With its all superconducting coil technology, ITER will be capable of moving toward steady-state, high gain operation for fusion power. While the physics basis for ITER's nominal inductive operation is well established, the physics basis for steady-state is currently being developed [1]. Ongoing tokamak research programs must continue to contribute strongly during ITER construction to various physics issues whose resolution will improve both the inductive and steady-state operation of ITER.

2. Stability Issues

Plasma stability will determine the operating space of ITER and the frequency of interruption of the research program for maintenance. ITER's research program will make critical advances for some stability issues.

2.1. The Resistive Wall Mode

Unstable resistive wall modes (RWMs), a branch of the external kink mode, cause disruptions that limit the plasma pressure and therefore the fusion power. The pressure limit can be expressed as a limit on the normalized beta quantity $\beta_N = \beta_T/(I/aB_T)$. Conventional tokamak operation lies below $\beta_N < 2.8$, the pressure limit for free boundary plasmas. ITER's nominal operating regime (β_N~2) is below this limit. The importance of the quantity β_N arises from the basic equilibrium relation $\beta_P \, \beta_T = 25\,[(1+\kappa^2)/2]\,(\beta_N/100)^2$. The fraction of the plasma current the plasma self-generates, the bootstrap fraction, is given by $f_{bs} = C_{bs}\, \beta_P / A^{1/2}$. The fusion power is proportional to $\beta_T^2 \times B_T^4$. The desires for high fusion power and high bootstrap fractions compete against each other; simultaneous high gain, steady-state operation in ITER (or any tokamak) requires increasing the RWM stability limit β_N. Fortunately the RWM can be stabilized if the plasma is rotating sufficiently fast inside a close fitting conducting metal wall [2]. With wall stabilization, β_N might be doubled, which from the basic equilibrium relation would allow either 4 times the bootstrap fraction at constant β_T or many times the fusion power at constant bootstrap fraction, huge potential gains. In ITER, higher β_N through wall stabilization would enable achieving ITER's fusion power goal at lower magnetic field and current or utilizing higher bootstrap fractions to realize steady-state operating scenarios at full fusion power.

However plasma rotation in ITER is estimated to be just below the threshold for rotational stabilization of the RWM (Fig. 1, [3]), about 2% of the Alfven speed. This estimate of the rotation speed balances neutral beam momentum input against diffusive momentum loss. While this 'standard model' probably contains the major physics elements, the model is questionable because plasmas with no momentum input have been found to rotate rapidly [4] and because the momentum diffusivity is just assumed comparable to the ion thermal diffusivity. More research is needed on the physics of what makes plasmas rotate. These caveats having been stated, in the context of this standard model doubling the momentum input into ITER may be enough to ensure rotational stabilization of the RWM. The momentum input to ITER is low because the neutral beam energy is so high, 1 MeV from each of two beamlines, and the ratio of momentum in a beam to power is M/P = $(2m/E)^{1/2}$. If the third beam envisioned in the ITER design were at 250 keV but the same power as one of the 1 MeV beams (which requires four times the ion source area), then the momentum input to the plasma would be doubled. Alternatively two beams at 500 keV and a third beam at 250 keV (with higher ion source areas) might be considered. However, recent data from DIII-D being presented at this conference imply the threshold for rotational stabilization may be much lower than previously thought [5]. Lower beam energies would also remove current drive from the center of the plasma making it easier to achieve advanced, high $q_{\min}$ operating modes and may also improve confinement through increased rotation. Tradeoff calculations need to be done about overall neutral beam current drive and its radial profile. ITER should closely monitor the rapidly evolving situation in regards to required rotation.

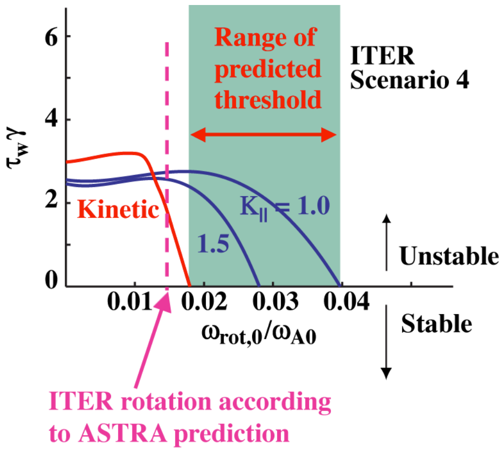

FIG. 1. Predictions of the threshold for rotational stabilization of the RWM and the rotation speed in ITER. Central rotation is given.

In the event that rotation cannot be increased in ITER, then direct feedback stabilization of the RWM by a suitable set of non-axisymmetric coils may be possible but remains to be proven by ongoing research. A suitable set of coils might still be incorporated into the present design by placing the coils essentially as liners inside the large midplane ports. The required currents in the coils are small. High bandwidth is needed in these coils for direct feedback stabilization. Even with sufficient plasma rotation, such a set of coils would be needed to counteract resonant field amplification (RFA) of error fields by a stable RWM [6]. The RFA can drive up non-axisymmetric fields in the plasma which create drag, lowering the rotation, and leading to an RWM disruption. Counter-acting RFA requires much lower bandwidth than direct feedback stabilization, but the required coils are essentially the same. Since the gains in ITER from wall stabilization would be great and the gains from this physics toward a DEMO even greater, and since it appears a reasonable location for the coils is possible, ITER should very seriously consider installing RWM feedback coils.

2.2. The Neoclassical Tearing Mode

The neoclassical tearing mode (NTM) is made unstable by a seed island which flattens the pressure gradient across the island removing the bootstrap current in the island. The resulting helical perturbation to the plasma current makes the island grow. The low order islands $m/n = 2/1$ and to a lesser extent $m/n = 3/2$ decrease confinement. If the 2/1 island grows too large and the plasma rotation gets too low, the island can lock, leading to disruption. The

NTM is expected to be unstable at low beta in ITER and is likely to be the principal performance limiting instability [1] so stabilization or avoidance (by high $q_{\min}$) is important. Stabilization by localized electron cyclotron current drive (ECCD) replacing the missing bootstrap current in the island works but we need to quantify the power needed in ITER. The issue is how well the EC waves can be focused to deposit current just in the island. The port for the EC launchers is too far above the midplane for optimal focusing. The focusing from that port achievable with remote steering launchers was so broad that more than the available 24 MW source was required to stabilize the NTM if the power were left on continuously [7]. In principle, the power requirement can be halved if the EC is 50% modulated to inject only into the island O-points as they rotate by. While three machines have shown stabilization with the EC on continuously [8], only ASDEX Upgrade has so far shown some promise with modulation [9]. Recent designs with front steering mirrors achieve adequate focusing for CW ECCD [10] and address the serious force and heat load challenges. An alternative to ECCD stabilization is just to use off-axis auxiliary and bootstrap current drive to prevent $q_{\min}$ from falling below 2 (for the 2/1 mode) or 3/2 (for the 3/2 mode).

2.3. Disruption Mitigation

With up to 350 MJ of stored energy in ITER, disruptions can have serious consequences [11]. The machine has been designed to withstand the EM force loads from transfer of the plasma current to the vacuum vessel wall. The heat pulses to the divertor will significantly erode the divertor plates, leading to frequent interruptions of the research program for vessel reconditioning and divertor repairs [1]. The generation of several MA runaway electron beams in disruptions threatens serious damage to in-vessel components. Successful mitigation of all three consequences of disruptions (EM forces, heat pulses, runaway electrons) with massive gas injection has been shown to work [12]. The essential feature is to get the Rosenbluth density of electrons (free or bound on partially ionized impurity atoms) into the plasma in a millisecond [13]. While early data were consistent with the high density neutral gas streams penetrating to the plasma center [14], more recent studies [15] have shown the injected gas apparently does not penetrate far into the plasma. But fast MHD mixing of the partially ionized gas into the core plasma is indicated. Unfortunately, it will be difficult to project such an MHD mixing process to ITER. Experimental research and theory calculations on the massive gas injection technique should continue. But the required massive gas injection system for ITER is sufficiently low cost that such a system should be included in the baseline plan now. Research toward liquid jet injection for disruption mitigation should be undertaken as a backup option.

2.4. Edge Localized Modes

ITER's baseline operating mode will be with an H-mode edge transport barrier. This edge transport barrier causes a large pressure gradient from the divertor separatrix plasma boundary up to the top of an edge pressure pedestal. Current theoretical understanding is that transport from turbulence is 'stiff' in that small increases in temperature gradients dramatically increase the transport in the core plasma; hence, the core plasma has effectively upper bounds on the temperature gradients. With such a stiff model, the overall energy gain in ITER is controlled by the height of the edge H-mode pressure pedestal. The larger the pedestal the better. But the large edge pressure gradients and bootstrap current densities in the edge transport barrier cause regular periodic edge instabilities [edge localized modes (ELMs)] of less than a millisecond duration that produce high heat pulses on the divertor surfaces which may cause excessive erosion of the divertor plates in ITER. There is also concern about ELM energy landing on surfaces other than the divertor. The stability limits to edge pressure gradients and current densities are understood but the width of the H-mode transport barrier region is not [1,16].

While there are many ELM types and many variations of their size, duration, and frequency, projections of the expected Type I ELM regime to ITER still imply the ablation threshold will be exceeded [1]. A promising active intervention is pellet pacemaking, in which a periodic stream of shallowly penetrating pellets is injected into the edge transport barrier with each pellet triggering an ELM [17]. With sufficiently frequent pellets, the ELM size can be made low enough.

However, the best solution is to completely eliminate ELMs while retaining the H-mode edge. Three methods have been found: the enhanced D-alpha mode (EDA) in Alcator C-Mod [18], the quiescent H-mode (QH-mode) [19] discovered on DIII-D and reproduced on ASDEX Upgrade, JT-60U, and JET; and the use of edge resonant magnetic perturbations on DIII-D [20]. The EDA mode requires much higher edge collisionality than will be the case in ITER.

The QH-mode suppresses ELMs in ITER's range of edge collisionality. The physics of this mode probably lies either in the plasma edge rotation or the orbit loss of neutral beam ions charging the edge plasma negative. QH-mode is most easily produced with counter-neutral beam injection, leading to the orbit loss hypothesis. JT-60U has produced QH-mode with balanced neutral beam injection and even co-injection, although ripple loss of fast ions may be important in JT-60U. Now that both JT-60U and DIII-D are equipped with both co-and counter-neutral beams (and have the additional variation of ripple loss), the physics of QH-mode should be understood in the next 2-3 years, allowing its extrapolation to ITER. If the physics is in the edge rotation, it may be difficult to accomplish this mode in ITER. If the physics is in the edge orbit loss of injected fast ions, the required hardware for QH-mode in ITER might be as simple a low energy (< 80 keV) positive ion counter injected neutral beam of possibly 5 MW or alternatively turning the diagnostic neutral beam slightly to the counter direction.

The suppression of ELMS by edge resonant magnetic perturbations has also been shown to work at ITER's edge collisionality. The edge RMP lowers the edge pressure gradient and current density to just below the stability limit. While work remains on understanding the physics, the required coil set for ITER probably has to produce magnetic perturbations confined in space to the outer 10% of the plasma (in order not to degrade confinement or rotation) and resonant at surfaces with q between 3 and 4.5. Such a coil system has to be relatively close to the plasma and have a high order structure. Finding locations for such coil systems on ITER may be difficult.

2.5. Alfvén Modes

ITER will afford a new and unique capability to study how the energetic alpha particle population will drive a wide variety of Alfven mode instabilities. ITER will make the *dominant research contribution* in this area. Current devices can study aspects of this physics using surrogate energetic particle populations produced by ICRF and neutral beams. Research in this area is rapidly expanding owing to improvements in various core plasma turbulence diagnostics that have enabled much more Alfven mode activity to be seen than was previously detectable on edge magnetic diagnostics [21]. Important new ideas are coming forward about possible core plasma contributions to driving these modes and the possible role of small spatial scale Alfven modes in core plasma transport [22]. The main action item for ITER in this area is to assure an adequate set of alpha diagnostics. Current plans have an adequate set for confined alphas but an inadequate set for escaping alphas.

3. Confinement at Low ρ* and ν*

ITER is the only machine in which we can learn the transport properties of tokamak plasmas at low $\rho*$ (the ratio of gyroradius to system size) and $\nu*$ (the collisionality). Hence ITER has a special obligation to study transport from turbulence in this fusion power plant relevant regime. Research in current devices needs to focus on transport in the electron channel, cases with $T_e = T_i$, and how stiff the transport really is. The disagreement in the dimensionless scaling results and the ITER global database scalings, particularly in beta dependence, should be resolved [23]. With a stiff transport model, the energy gain is controlled by the height of the edge pedestal pressure. We can calculate the limiting edge pedestal pressure gradient, but we need research on what determines the pedestal width [16]. The L-H transition threshold physics needs renewed effort, perhaps illuminated by internal transport barrier formation studies. The adequacy of ITER's turbulence fluctuation diagnostics to meet this major mission of the device is a concern.

4. Plasma Boundary Issues

4.1. First Wall Materials

The best mix of first wall materials to use in a fusion power system is a crucial issue. It is a crucial issue for the successful operation of ITER and ITER must provide the research basis for the materials choices for an eventual DEMO. The current design position for ITER is a beryllium main chamber wall, carbon in the high heat flux divertor zones, and tungsten in the less active divertor zones (Fig. 2). This mixed materials compromise uses the low Z beryllium in the main chamber because material eroded from the main chamber wall most easily gets into the core plasma, but low Z materials are tolerable in the core. It uses carbon in the high heat flux divertor zones because carbon has been shown best capable of handling both the steady-state and pulsed heat loads (from ELMs and disruptions). The introduction of tungsten is mainly for its future fusion power plant relevance owing to its low erosion by plasma fluxes.

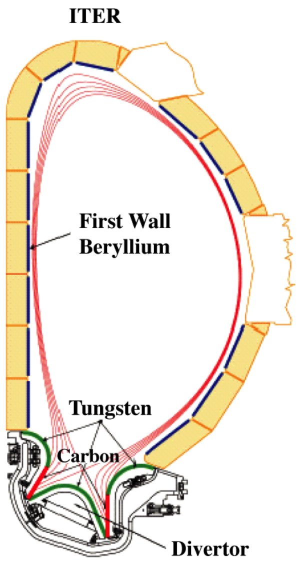

FIG. 2. ITER first wall and divertor materials.

All of these materials have limitations. The beryllium wall is predicted to have surface melting from the intense radiation flash (350 MJ in 1 ms) from a successful massive gas injection disruption mitigation [24]. Some resolution of this conflict is needed since disruption mitigation will be essential to a successful research program in ITER. More detailed considerations are being made [11]. Carbon materials have the problem of co-deposition of tritium on surfaces, possibly leading to an unacceptable tritium inventory in ITER. Predictions from experimental data and modeling of the full power burn time to reach the tritium inventory limit in ITER have ranged widely from early estimates of 10,000 s (25 discharges) to recently 200,000 seconds (500 full power and pulse length discharges, a few years) based on JT-60U results (Fig. 3) [1,25].* Operation of surfaces at high temperatures can inhibit co-deposition [26]; oxygen baking is being developed as a way of periodically removing tritium co-deposited with carbon. Tungsten is very deleterious to plasma confinement owing to the large radiation it produces in the core plasma. Continued research is

* The extrapolation of JT-60U results [25] to ITER was made by M. Shimada as follows: In ITER, ~ 30 MW of power reaches the target; plasma temperature at the target is 10 eV; the particle flux at the target is 2.64E24/s (heat transmission coefficient=7.1); chemical sputtering yield of 0.006 [25]. Then the carbon generation rate is 1.58E22/s. If codeposited DT/C is < 0.04 [25] The DT deposition rate is 6.33E20/s and the T deposition rate is 3.17E20/s. The accumulated T deposition for 2E5 s is 6.34E25 (atoms), which is 316 g.

needed on the tritium codeposition issue for carbon and on the melting, cracking, and plasma impact issues for metals, and on the mechanical and chemical properties of mixed material redeposition layers and the ability to remove tritium from them. Given the uncertainties in this area and the need for ITER to test materials for DEMO that may be different than what is optimal for ITER achieving its critical early objectives, it would be advantageous to be able to change the first wall materials in less than one year and at a reasonable cost. Perhaps two such changes might occur in the lifetime of the project.

4.2. Divertor Physics

ITER is the only machine that can display what edge, scrapeoff layer, and divertor physics will be at high absolute densities and simultaneously low collisionalities. The boundary plasma is too complex to construct dimensionless parameter scaling rules for how to project results in current machines to ITER. Results in current machines vary from laminar type SOL flow and divertor recycling at ITER's collisionality to clumpy cross-field transport with substantial main chamber wall interaction at ITER's absolute density. Without scaling rules, benchmarked codes to calculate ITER's boundary plasma properties are the only recourse to carry forward physics from current machines. Comparison of the unique boundary physics data from ITER to today's smaller, current tokamaks should allow the determination of how divertor properties can be extrapolated from one machine generation to the next.

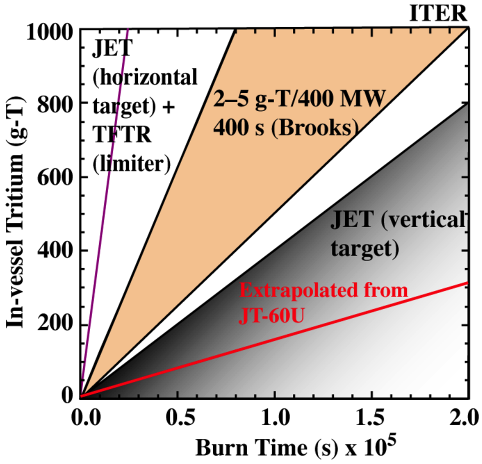

FIG. 3. Various predictions of T inventory in ITER versus seconds of full power operation.

4.3. Pellet Fueling

How to get fueling deep into the core plasma is still an issue [1]. Substantially increased penetration of pellet mass into the core plasma has been found by injecting pellets from the high field side instead of the low field side. However the circuitous routing of pellets to get to the high field side limits the pellet speed to 300 m/s and such pellets in ITER only penetrate one-third of the way into the plasma. New approaches may be needed.

5. Advanced Operating Scenarios and Integration Issues

5.1. Advanced Hybrid and Steady-State Scenarios

The study of the complex feedback loops involving the current profile, bootstrap current, alpha heating, transport barriers, instabilities, etc. in high performance, burning, self-heated, steady-state plasmas will be a *unique research contribution of ITER*. Current research is making exciting progress. Hybrid modes in which the OH transformer supplies some current drive extrapolate to ITER's full fusion power in several thousand second pulses [1]. Such modes may enable achieving ITER's performance goals at lower plasma current. Discharges equivalent to ITER's true steady-state Q=5 mode with 100% non-inductive current drive have been demonstrated [27]. Prospects look good for even higher performance, but wall stabilization will be needed.

In general these modes utilize startup procedures that retard the penetration of plasma current to keep a minimum-q value high (at least 1, maybe above 2) by getting to a full bore and diverted plasma as early as possible in the discharge and using heating and off-axis current drive during the current rise phase. In contrast, the current startup plan for ITER grows a small plasma from the outer limiter, a process which encourages current penetration. If sawteeth start before the plasma is diverted, then it will be difficult to get the current back out of the plasma center and the sawteeth will trigger NTMs. If heating must be applied in the rampup phase, the earliest possible formation of the divertor magnetic configuration will reduce the impurity accumulation that would occur in a limiter configuration. Alternative startup scenarios should be investigated for ITER; they will likely have important hardware implications for power handling, slew rates of coils, etc.

5.2. ITER Auxiliary Heating Systems

The auxiliary heating systems on ITER were chosen when central plasma heating was the main concern. Alterations of the mix of systems or additions to perhaps a total 130 MW [28] might focus on optimizing current drive and current profile control by more ECCD, adding LHCD, and increasing the momentum input for RWM stabilization to enable higher bootstrap fraction with high fusion gain. The general thrust toward advanced steady-state plasmas is toward moving plasma current from the center to the outer half of the plasma cross-section, leading to more interest in increased ECCD or lower hybrid. ICRF is only useful for central heating and current drive. The physics of coupling of ICRF waves through edge plasma needs work. Since H-mode cannot be obtained in plasmas limited at the outer midplane, the compatibility of LH with advanced modes, including H-mode, needs to be demonstrated if the LH launcher is closely coupled to the outer midplane plasma. Or methods must be shown to couple power through a large gap between the plasma and the launcher [29]. LH would be the most efficient means of driving off-axis current.

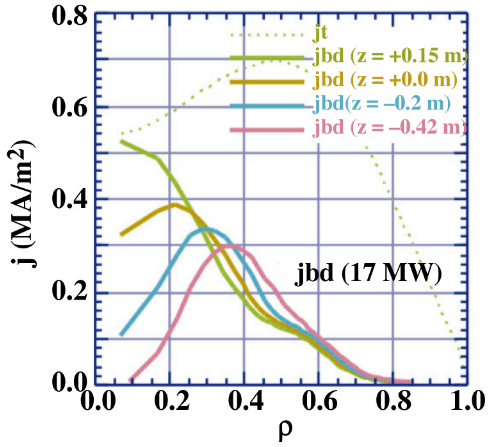

FIG. 4. Off-axis current drive calculated in ITER from tilted NBI.

For neutral beams, the 1 MeV beam energy was chosen mainly to deliver heating power to the plasma center. The 1 MeV NBI system can produce some off-axis current drive (Fig. 4) by tilting the beamline so it aims off-axis. However these calculated predictions are called into question by the recent results from ASDEX Upgrade [30] which show that at high power the off-axis current drive is not obtained. The off-axis current drive capability of lower energy beams should be evaluated. Lower energy beams also inject more momentum for rotation (see RWM discussion above).

5.3. A Comprehensive Simulation Code

A comprehensive simulation code including both engineering and physics should be developed to integrate the physics elements discussed above and to assess planned operation. Extensive validation of the code against existing experiments will be required and ITER might play a central role in coordinating the international efforts in this area.

6. Summary

Ongoing tokamak research programs must continue to contribute strongly during ITER construction to various physics issues whose resolution will improve both the inductive and steady-state operation of ITER. However at the present time, the ITER design needs to take account of research advances that have been made in RWM stabilization, NTM stabilization, disruption mitigation, stabilization of ELMs, and advanced performance modes with their special startup and off-axis current drive requirements. Other important issues are in the ability to change the first wall materials, the adequacy of turbulence and escaping alpha diagnostics, and the need for a comprehensive simulation code. The research program on ITER will make unique contributions in alpha physics, the confinement at low $\rho*$ (the ratio of gyroradius to system size) and $\nu*$ (the collisionality), in divertor physics at low collisionality and high absolute density, and in the study of the complex feedback loops involving the current profile, bootstrap current, alpha heating, transport barriers, instabilities, etc. in high performance steady-state plasmas.

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