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Paradigm Changes in High Temperature Plasma Physics Research

and Implications for ITER

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Paradigm Changes in High Temperature Plasma Physics Research and Implications for ITER

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Abstract

Significant high temperature plasma research in both the magnetic and inertial confinement regimes led to the official launching of the International Thermonuclear Experimental Reactor (ITER) project which is aimed at challenging controlled fusion power for human kind. In particular, such an endeavor originated from the fruitful research outcomes from the world wide magnetic confinement devices (primarily based on the Tokamak approach) mainly in advanced countries (US, EU, and Japan). In recent years, all new steady state capable Tokamak devices are operated and/or constructed in Asian countries and incidentally, the majority of the ITER consortium consists of Asian countries. This provides an opportunity to revisit the unresolved essential physics issues and/or extend the understanding of the transient physics to the required steady state operation so that ITER can benefit from these efforts. The core physics of a magnetically confined hot plasma has two essential components; plasma stability and cross-field energy transport physics. Complete understanding of these two areas is critical for the successful operation of ITER and perhaps, Demo reactor construction. In order to have stable high beta plasmas with a sufficiently long confinement time, the physics of an abrupt disruption and sudden deterioration of the energy transport must be understood and conquered. Physics issues associated with transient harmful MHD behavior and turbulence based energy transport are extremely complicated and theoretical understanding needs a clear validation and verification with a new research approach such as a multi-dimensional visualization.

Introduction

The International Thermonuclear Experimental Reactor (ITER) project [1] (see Fig.1), has officially been launched and consists of a consortium of seven countries with the aim of demonstrating the feasibility of this future energy source for human kind. The significant increase in the number of participating countries from Asia (China, India, Japan, and Korea) is mainly driven by the necessity of this precious energy resource. At the same time, it reflects the paradigm change in fusion plasma research which once was dominated by advanced countries (Russia, US, EU and Japan). Through intensive physics research involving large tokamak devices (TFTR[2], JET[3] and JT-60[4]) which have created the plasma conditions necessary for the optimum fusion reaction (~20keV; optimum cross-section for the Deuterium-Tritium fueling), the ITER project is well justified. However, the sustainment of magnetically confined hot plasmas is still a challenging physics problem in addition to the engineering issues which need to be resolved in the future. The physics of the containment of the high temperature plasma consists of two branches; cross field energy transport and stability physics. Transport physics is an understanding of the cross field energy transfer which is crucial for the compact fusion devices. For the larger devices like ITER, the primary physics issue is the

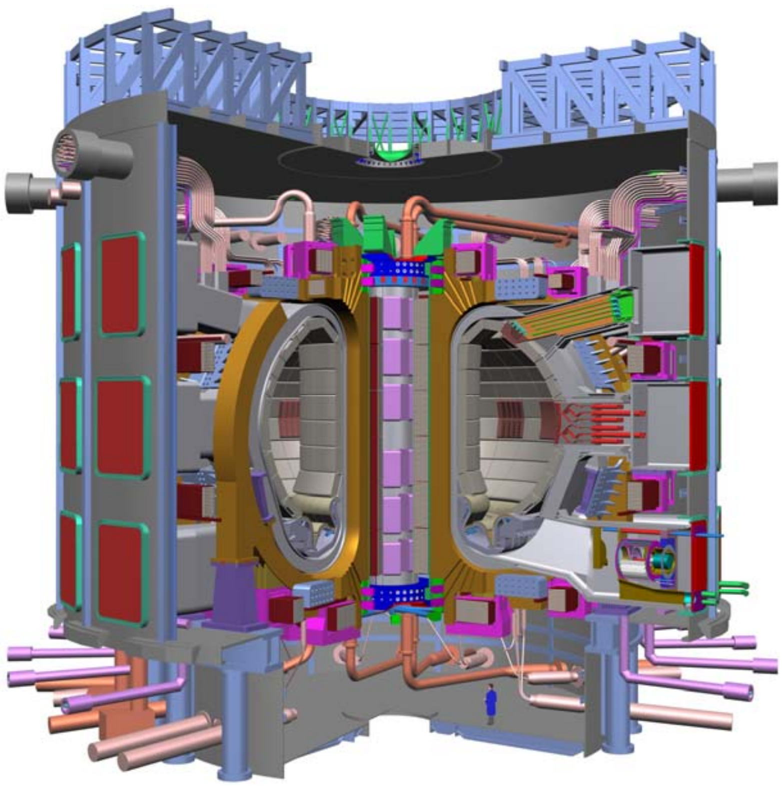
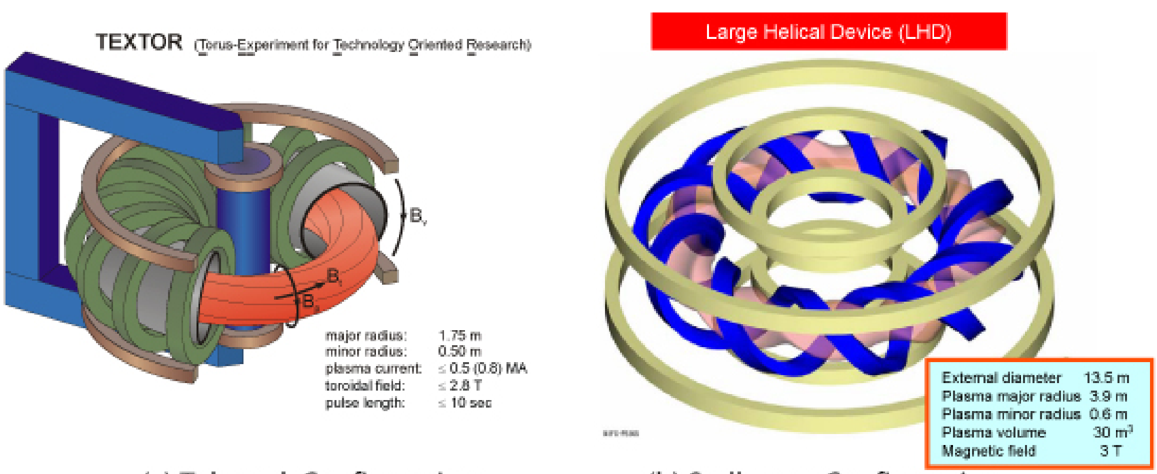


Fig.1. Schematic of the International Thermonuclear Experimental Reactor

stability. The stability problem arises mainly from the unstable growth of Magneto-Hydro-Dynamic (MHD) modes driven by free energies from the plasma pressure and/or current distribution in a closed magnetic configuration. Control of the harmful MHD mode is critical for the steady state operation of the high β plasmas; beta is the ratio between the plasma pressure and magnetic energy [$\beta = (2\mu_0 \sum_{j=i,e} n_j T_j / B^2)$] where n_j and T_j are number density and temperature of ions and electrons in the plasma and B is the magnetic field strength. Reliable control mechanisms are only possible from a full understanding of the physical mechanism of the explosive growth of those harmful MHD modes which can often lead to catastrophic plasma disruption. Physics modeling of the MHD modes in the hot plasma has been advanced significantly based on world-wide fusion research but is not conclusive yet which is what is needed to develop a precise remedy for the harmful instabilities in ITER. This is largely due to the underestimated complexity of phenomena which require much more sophisticated multi-dimensional diagnostic system to map out precisely the nature of the problem. As an example, a proof-of-principle state-of-the-art two dimensional millimeter wave "camera" system has been challenged to resolve the classical MHD problem in a Tokamak device and has provided clear conclusions for the disputed physics of the m=1 mode (sawtooth oscillation). A new approach for physics studies is essential so that the transient high beta plasmas achieved in previous generation Tokamaks can be extended to the steady state operation in the new superconducting tokamak devices such as KSTAR, EAST, SST and JT-60-SA in Asia. Successful test of the new physics on these devices will increase the chance of the success of the ITER project which will eventually lead to the successful Demo construction.

Paradigm change in fusion research and ITER

The successful demonstration of the uncontrolled fusion power (i.e., the H-bomb) has lead to extensive international plasma research activities for more than a half century. Explosive ideas of the controlled fusion faced a steep challenge. The promising results were from devices with a strong external magnetic field. Among numerous magnetic confinement concepts, a closed magnetic confinement system has been demonstrated to be more promising than the open systems. Among the closed systems, the most extensively investigated is the Tokamak plasma with a helical structure of the magnetic surface produced by a combination of an externally applied toroidal magnetic field and the magnetic field induced due to the self driven current as shown in Fig.2a. Due to the resultant helicity of the magnetic field, particle confinement is dramatically enhanced compared to a system with a simple toroidal magnetic field only. The tokamak concept has been the most popular fusion research subject due both to its relative simplicity and the fact that the plasma confinement characteristics in this device have been superior to those of other concepts. While the helicity induced by the internal plasma current distribution enhances the confinement of particles, it also can be a source of instability which can lead to the disruption. Also the induced plasma current by the external transformer is transient in Tokamak and requires a current drive system to operate in a steady state mode. The other alternative concept is the "Stellarator" which has the helical structure of the magnetic field produced by a complicated set of external magnetic coils so that no driven current is required as shown in Fig.2b. Since no driven plasma current is required, this device can be operated in a steady state mode, in principle. However, the



(a) Tokamak Configuration (b) Stellarater Configuration

Fig.2 Two representative magnetic confinement concepts coil structure is much more complex due to the fact the helicity of the magnetic field comes from the external coil windings and any small error in coil winding can result in an unstable plasma operation. Before the 1970s, numerous small Tokamak devices were built with the main emphasis on understanding the fundamentals of tokamak plasma physics. In early 1970, three large Tokamak projects were launched by the US, EU and Japan based on experience accumulated from these smaller devices. All devices were based on Cu coils and various current drive concepts have been explored to prolong the discharge period. Also, a variety of heating scenarios were tested to improve the plasma performance including high energy neutral beams and Radio Frequency heating. All three devices demonstrated that the Tokamak configuration can confine plasmas with the optimum conditions for the fusion reaction (energy confinement time of ~0.5 sec., ion temperatures up to ~40keV and core plasma density of ~1 x $10^{20}/m^3$). The Tokamak Fusion Test Reactor (TFTR) at Princeton, USA was the flagship of the US fusion effort. TFTR achieved a fusion power yield (Q=output power/input power) of ~0.3 at the end of the full D-T experimental campaign. With excellent diagnostic systems on this device, outstanding physics research results were produced. Two devices, Japanese Tokamak (JT)-60U, Naka, Japan and Joint European Tokamak (JET), Culham laboratory, UK are still in operation. JET also performed a full DT experiment and produced Q~0.7 and this device has been used to benchmark the ITER related physics and engineering experiment. JT-60U has mainly operated using Deuterium only and the extrapolated Q value was ~1.25. Note that the duration of the record fusion yield in each device was quite short compared to the energy confinement time and the high performance plasma often encountered harmful MHD instability at the peak of the performance. Similarly to the JET device, many ITER related experiments have been carried out. Research results from the three large Tokamak devices together with the study results from other smaller devices, produced reliable and convincing empirical scaling laws[5] that can project the energy confinement time which is the critical component of the fusion device performance as shown in Fig. 3. In parallel with the experimental progress, the progress of the theoretical understanding of the magnetically confined plasma has been significant mainly due to vital experimental results and computational capability. Based on the convincing but physically not well understood scaling law and remaining engineering studies, the ITER project which was first announced in 1984, was officially launched in 2006 by an international consortium consisting of EU, Japan, US, Korea, China, and India. The objective of this project is to study the feasibility of the fusion power reactor and the success will lead to the DEMO reactor construction in the future. It is notable that four out of the six participants are from Asian countries and incidentally, Asia has become the most active area for magnetic fusion research. Recent new steady state capable Tokamak fusion devices are operating and/or being built in Asia. These are EAST at Hefei, China, KSTAR at Daejon, Korea, SST-1 at India, and JT-60SA device at Naka, Japan by joint effort between Japan and EU. In addition to the JT-60SA, the Large Helical Device (LHD), NIFS, Japan has been operating and the feasibility of the Stellarator concept is being explored.

Accelerating economic growth and ever increasing energy consumption in Asian countries motivate a new investment for the quest for a new energy source. Through reliable engineering capability at a reasonable cost, each country has been able to operate and/or construct these complex research devices which once were only possible in advanced nations. A long period of the fusion research in advanced countries has produced many outstanding Asian researchers and engineers. They can organize new research teams together with their new young generation to accelerate their program so that the device is sufficiently matured in a short enough time to challenge the critical physics problems beneficial for ITER and DEMO reactors in the future.

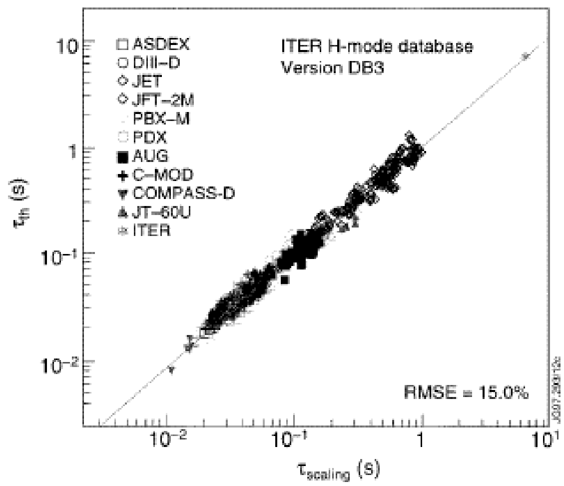


Figure 2. τ_{th} versus the scaling expression of (2) for the ELMy dataset.

Fig.3. Scaling law for the energy confinement time based on accumulated data from many devices (after J. Cordey)

Physics of the magnetically confined plasmas and complexity

The physics basis of the ITER project is largely dependent on scaling laws established using the accumulated confinement data during the last ~40 years of world wide Tokamak research as illustrated in Eq.(1).

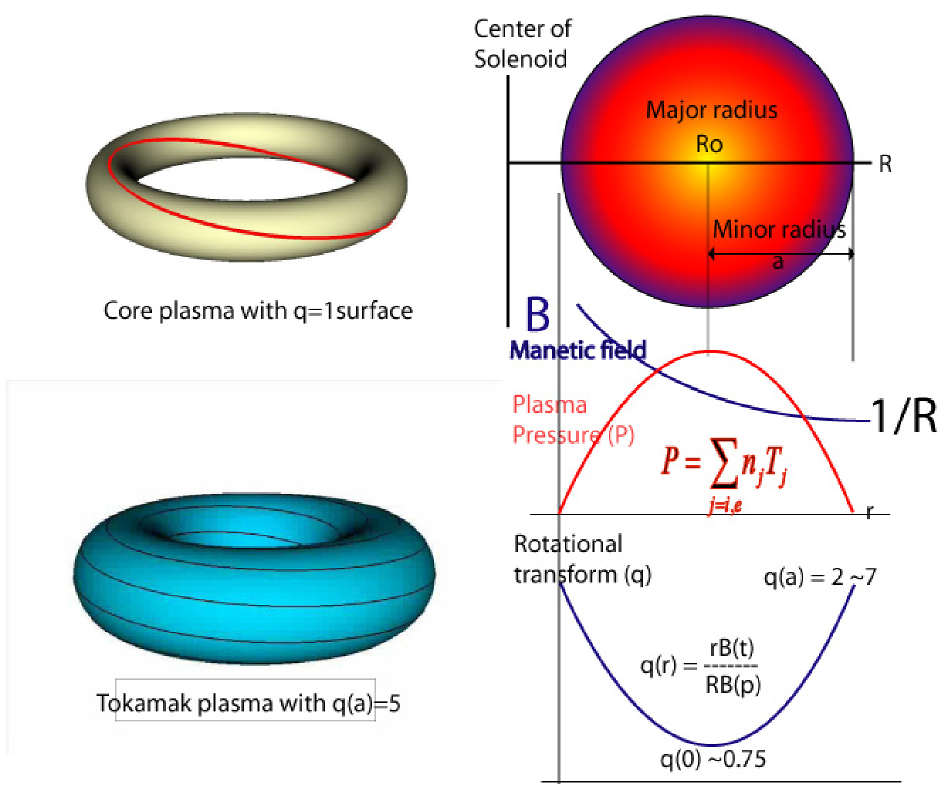
$$\tau_{th} = 0.029 I^{0.99} B^{-0.06} P^{-0.69} n^{0.61} R^{2.11} \varepsilon^{0.22} \kappa^{0.7} M^{0.11} \tag{1}$$

Here, I is the plasma current (MAmp), B is the strength of magnetic field (T), P is the heating power (MW), n is the plasma density ($/10^{19}/m^3$), R is the major radius (m), ε is triangularity, κ is elongation, and M is mass ratio to the Hydrogen. Physical understanding of the energy confinement time dependence on machine parameter such as magnetic field, plasma current, major radius, minor radius, geometric shape factors, etc., is important for the future improvement of the concept. The fundamental physics problem of the magnetic confinement concept including the Tokamak is two fold. As briefly discussed in the previous section, the duration of the peak performance of the discharge (highest fusion power yield) is often abruptly ended on a time scale shorter than the energy confinement time and this is largely due to the sudden growth of the harmful MHD instability. Throughout the research of the high β plasmas in existing Tokamak devices, a number of harmful MHD instabilities that can be a threat for the steady state operation have been identified. We are still in an infant stage for a full understanding which can lead to permanent remedies of these harmful MHD modes. Furthermore, it is essential to have a practical demonstration on these steady state capable devices prior to construction of ITER so that the success of ITER and devices beyond is guarantied. The other branch of physics which still needs a full understanding is the cross field energy transport problem through the micro-turbulence. In tokamak research, there are many acronyms for the confinement regimes operated in each device such as H-mode, L-mode, Supershot, etc. These names are purely based on empirical knowledge and there is no clear physical description of each operating regime. The difference in energy confinement time in each regime is roughly a factor of two. This difference may not be a significant factor when the practical number is compared to the scaling law which has a variation of a similar order on a linear scale. As we have empirically learned that the performance of the confined plasma is largely proportional to the size of the device (plasma volume), for the ITER size of the device, the energy confinement time may not be a critical issue if the optimistic confinement regime is chosen. However, an advanced compact fusion device needs to find a way to improve the energy confinement time. The most effective way to improve the performance in the smaller size device is to understand the basic physical mechanism of the energy confinement of the Tokamak plasma. Therefore, a full understanding of the physical mechanism of the cross-field mechanism based on micro-turbulence is not only beneficial for the ITER device but also for the advanced concepts.

The behavior of the hot plasma with an enormous excess free energy in a complex magnetic field is dynamic and vulnerable to numerous potential instabilities. In this modern era, one might expect that one is fully capable of simulating the entire toroidal plasma using massive parallel processors. Indeed, there are many publications featuring incredibly sophisticated time dependent motions of the plasma fluctuations and MHD modes. However, in spite of this great progress, we are still not able to confidently predict either the performance or the stability of the specified operating regimes of the Tokamak device. This is largely due to the fact that the complex physics introduced for the theoretical modeling is only partly verified with a limited diagnostic capability. Often the plasma instabilities arise from the nonlinear interaction of many physical mechanisms. Therefore, a decisive clarification of each mechanism verified by a firm diagnostic system step by step is essential for the valid outcome of the result. The dynamics of hot plasmas in a complex magnetic field are nontrivial to diagnose, since the growth time scale of the instability often extends down to the micro-second regime and the problem spans over more than a single dimension. As we aware, complex systems require accurate and detailed diagnostics as demonstrated in multi-dimensional visualization diagnostic systems in Magnetic Resonance Imaging (MRI) for human bodies which can address the problem accurately. There have been numerous efforts on visualization of the plasma motions through both active and passive topographic systems. First of all, a chordal measurement, whether performed with passive or active system, requires a significant number of views and chords to address the scale of the instability which can be as small as the orbit size of electrons and ions (~ mm scale). This is hard to achieve in the Tokamak environment due to the limitation in access. Another complication is that the tomography based on the emission is often a function of multiple plasma parameters and this makes the interpretation more complex. Therefore, these diagnostic systems have been used mostly for survey purposes and one has to be extremely careful in interpretation of the outcomes. It is thus imperative to be innovative in eliminating the ambiguity of the measurement in visualization.

New approach to study the physics of the Tokamak plasmas

The intrinsically enhanced energy confinement property of the Tokamak plasma compared with other magnetic confinement devices is largely due to the rotational transform induced by a self driven internal current. A common representation of the profile of the plasma current is via the so-called safety factor (q) profile. Here, q is simply a measure of the number of times a field line goes around a torus the long way (toroidal direction) for each time around the short way (the poloidal direction) as shown in Fig.4(a). In an ordinary tokamak plasma, the current density profile is positive definite toward the center as well as the pressure profile. The corresponding q - profile is monotonically increasing towards the edge as shown in Fig. 4(b).



(a) (b)

Fig. 4. (a) An example of the sheared Tokamak plasma magnetic surface at q(a)=5 at the edge and q=1 surface at the core plasma. (b) Poloidal cross section of the Tokamak plasma with a typical pressure and rotational transform profile shape

Under these conditions, the repetitive disruptive behavior of the plasma core within the q~1 layer, commonly referred to as "sawtooth oscillation", was discovered in the early days of fusion plasma research [6]. This is known as the *m/n=1/1* internal kink mode where *m* and *n* are poloidal and toroidal mode numbers, respectively. An excellent review of recent research in the field of sawtooth oscillations is given in Ref. [7]. High-resolution 2-D images of the electron temperature fluctuations during the sawtooth crash phase in TEXTOR have been measured by a 2-D electron cyclotron emission imaging (ECEI) system. The basic principle of the technique is similar to that of conventional 1-D ECE radiometers [8, 9]. The new feature of the ECEI diagnostic is that measurements are done in a 2-D matrix of sample volumes. Detail of the ECEI diagnostic can be found in Refs. [10,11]. Since the physics results from the ECEI system have been also published in Refs. [12, 13, 14], the results will only be briefly summarized in the following sections.

During the last ~30 years, there have been numerous physics models proposed to explain the sawtooth crash and three prominent theoretical models are summarized in this section. In the full reconnection model [12, 13], the plasma current density in the core region increases (q(0) drops below unity), and the *m/n=1/1* internal kink mode becomes unstable due to a pressure driven instability. Island formation starts due to an influx of the cooler part of the plasma outside the inversion radius through the magnetic reconnection, as soon as the pressure driven instability reconnects the magnetic field through the reconnection zone along the magnetic pitch of the q~1 surface. As the island (the region with q~1) grows, the hot spot (the region with q<1) gets smaller and it is eventually eliminated and the island fully occupies the core on a reconnection time scale defined as $\tau_c \approx \frac{1}{2}\sqrt{\tau_A^* \cdot \tau_\eta}$, where τ_A^* is the modified Alfven transit time and τ_η is the resistive diffusion time in Refs. [15,16]. Second, the quasi-interchange model [17] differs significantly from the full reconnection model and does not require any magnetic field reconnection process. The core plasma having a flat q-profile (q~1) inside the inversion radius becomes unstable due to a slight change of the magnetic pitch angle. In this model, there is no pressure driven instability. As the hot spot deforms into a crescent shape, the cooler outside portion of the plasma is convectively inducted into the core region, resulting in a flattening of the core pressure profile. The distinctively different evolution of the hot spot and/or cold island formation between the quasi-interchange model and full reconnection model could not be conclusively identified due to the lack of reliable 2-D experimental tools.

The evolution of the hot spot/island in the early stage of the precursor period is compared with the relevant images from the full reconnection model and the quasi-interchange model in Fig. 5. In the full reconnection model, the formation of the island is an indication of the topological change of the magnetic field through the reconnection at the low field side. 2-D images from the simulation results in Ref. 16 are directly compared with the relevant experimental images as shown in Fig. 5a. The shape and growth of the island in Fig. 5a are strikingly similar to those from simulation results of the full reconnection model. On the other hand, the shape of the hot spot is circular and it swells as it approaches the crash time, whereas the hot spot in the model is shrinking as the island grows in simulation. In the experimental result, there is no indication of a heat flow until the reconnection through the sharp temperature point takes place. In the full reconnection model, the formation of the island is the beginning of the reconnection

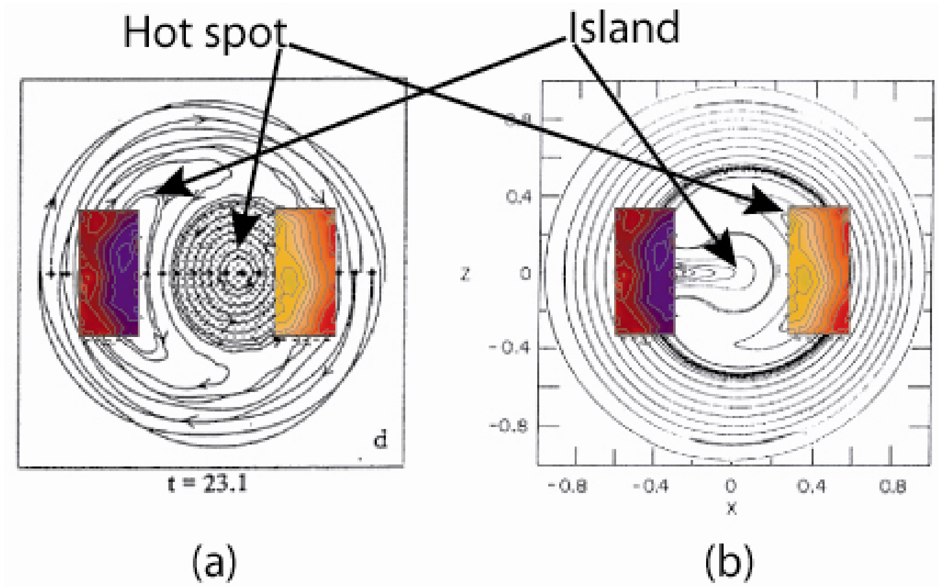


Fig.5. The measured partial 2-D images of hot spot (orange-yellow) and cold island (blue) are overlaid with the simulation results of the prominent theoretical models. (a) Full reconnection model (b) Quasi-interchange model.

process since it is assumed that the island is the result of a topological change of the magnetic field structure. The reconnection time, estimated based on the definition of the characteristic time (τ_c) which starts with the time when island formation is observed (precursor) and ends with the full island formation in the core for this experimental condition, is ~600 μs which is consistent with the estimated value of τ_c. However, it is notable that a trace of heat flow outside of the inversion radius was routinely observed in the later stage of the precursor. If the time when a trace of the heat flow outside of the inversion radius is detected is regarded as the beginning of the reconnection process, the reconnection time is less than ~100 μs. This observation suggests a new physical mechanism which may delay the reconnection process until a critical time while the island grows. Alternatively, the reconnection process is based on two distinctive phases; the first phase is an extremely weak reconnection, while a stronger reconnection driven by a pressure mode follows in the second phase. Often, the "crash time" is referred to as the time period from the maximum value of $T_e(0)$ to the minimum value of $T_e(0)$ when there are no precursors whereas the characteristic reconnection time (τ_c) is referred to as the time period from the moment when the island is formed during the precursor phase (indication of reconnection at the lower field side) during the precursor phase to the moment when the island is fully established. In the quasi-interchange model, the hot spot deforms into a crescent shape due to magnetic instability and the cooler parts of the plasma are convectively induced to the concave side of the crescent shaped hot spot as shown in Fig. 5b. Therefore, any magnetic field line reconnection process is not required to explain the sawtooth oscillation. It is clear that the observed partial 2-D image of the hot spot is a part of the circle and not a part of the crescent shape. Therefore, the time evolution of the measured *m/n=1/1* mode does not resemble the images of the hot spot from this model. The time evolution of the island (cold spot) in the experimental 2-D images is distinctively different from this model. Furthermore, the localized reconnection does occur with a sharp pressure point and heat flow crosses the inversion radius whereas this model does not require any type of reconnection process. Since the occurrence frequency of the full reconnection type of the sawtooth crash is dominant, the pressure instability driven reconnection may be the dominant mechanism compared to the magnetic instability.

Observation of a localized electron temperature bulge [18, 19] at the low field side on the poloidal plane in the Tokamak Fusion Test Reactor (TFTR) device has been interpreted as caused by a finite pressure effect on the sawtooth oscillation [20]. Here, a steep pressure gradient near the temperature bulge at the low field side leads to a global stochasticity of the magnetic field which is thought to be necessary in order to reconcile the small change of the current density and the fast change of the pressure during the reconnection time observed in finite β plasmas. Finally, the pressure driven ballooning mode instability was introduced to account for the observed disruptions lead by a sawtooth crash in the high beta (β_p ~ 1 and $\beta_t(0)$ ~ 4%) plasmas [21] in TFTR. These modes are more pronounced at the bad curvature side of the magnetic surface (low field side of the torus). Also, a 3-D local reconnection model where the reconnection zone is localized in the toroidal plane with many assumptions has been proposed in Ref. [19]. In plasmas with a moderate beta (β_p ~ 0.4 and $\beta_t(0)$ ~ 1%), where the present 2-D imaging measurements were conducted, the level of the ballooning modes and global stochasticity of magnetic field lines that are strongly coupled with the pressure surfaces, is moderate compared to those at high beta plasmas as demonstrated in Ref. [22]. All models developed to explain the sawtooth oscillation are based on numerous assumptions, and thus there is a strong need to compare them with precise experimental results.

The sharp temperature point or "pressure finger" accompanied with the swelling of the *m/n=1/1* mode at the low field side of the torus is the signature of the ballooning mode model. Dispersion of the heat is dominated by the global stochastic magnetic field in this model. The magnitude of the "pressure finger" and the global stochasticity of the magnetic field are small at the moderate plasma beta. In Fig. 6, the observed 2-D images of the reconnection processes on the poloidal plane are compared to those from the simulation results of the ballooning mode model [19] for a similar plasma beta (β_p=0.4 and β_t~2%). Three 2-D images (before the presence of the ballooning mode, ballooning mode, and crash phase) are directly compared to the 2-D pressure pattern of the ballooning mode in the bad curvature (low field) side from the simulation [19]. The pressure bulge with a smooth surface before the development of the ballooning mode is quite similar as shown in the top frame of Fig. 6b. In the middle frame of Fig.3b, the sharp temperature point is strikingly similar to the ballooning mode from the simulation. While the stochastic behavior is dominant in the pressure pattern of the simulation, the experimentally measured heat flow patterns are highly collective as shown in the bottom of Fig. 6b. At the good curvature side of the torus (high field side), the measured 2-D image before development of the ballooning mode is quite similar as shown in the top frame of the Fig. 6a. In the middle frame of Fig. 6a, instead of the "pressure finger" as shown in the low field side, the *m/n=1/1* mode is indented toward the center while the observed 2-D image of the sharp temperature point resembles that of the low field side. Like the low field case, the global stochasticity of the pressure pattern is dominant in simulation while the heat flow is highly collective in the high field side (bottom of the Fig. 6a).

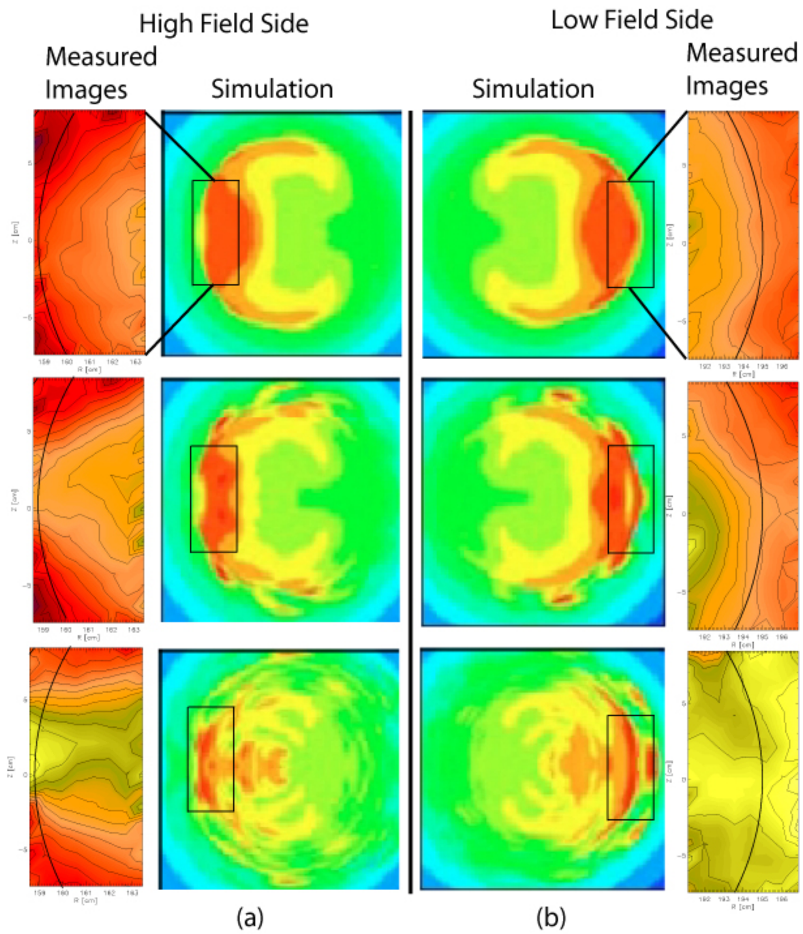


Fig. 6. The measured time dependent 2-D images of hot spots near the q~1 surface are compared the simulation results of the well established Ballooning mode model. (a) Discrepancy is clear at the high field side where there should be no reconnection process (b) Experimental images are consistent with simulation results except that there is no clear global magnetic field line stochasticity in experimental data.

Summary

Progress in fusion plasma research has been dramatic; consequently, the ITER project has been officially launched by an international consortium to explore a feasibility of the fusion power for mankind. The majority of the participating countries are from Asia and all new steady state capable Tokamak devices are operated or being built in Asian countries. This is an indication of a paradigm shift in fusion research until the ITER is fully functional. The weakness of the foundation of the ITER is a firm physics of understanding, since it is heavily reliant on empirical scaling law based on data accumulated during the last half century. There are a number of physics issues as well as engineering problems to be resolved in order to improve the margin of the ITER and to provide the necessary confidence to build the Demo fusion reactor. With the progress of diagnostic technology and computation capability which were not available a decade ago, a new way of studying the complex dynamics of hot plasmas is introduced. Multi-dimensional visualization of the critical physics will verify the proposed theoretical models. Eventually a fusion device based on the first principle physics understanding can be initiated.

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