The world tokamak community has confidence that ITER will be successfully constructed without obstruction; the target for this is 2025. The scientific goal of ITER is to demonstrate the feasibility of 500 MW fusion power with Q = 10 lasting for 400 seconds. There are still some risks in meeting this goal. The frontier and the most important issues for the next 10 years are the control of off-normal events, confinement and transport physics of the burning plasma, high-power particle and heat flux exhaust on the divertor, energetic particle behavior, high-performance long pulse and steady-state operation, and tritium breeding and retention.

Transient events can release plasma energy in a very short time and could damage plasma-facing components easily. These phenomena include minor and major disruptions and very localized plasma edge instabilities. Any major disruption in ITER could easily destroy some of the plasma-facing components, and therefore it is essential to develop robust ways to avoid or mitigate disruptions to prevent plasma-facing material from being damaged, to allow achievement of high-performance plasma and to ensure continuous operation in ITER. An elevated effort is needed in theory, modeling and technology for more advanced control to achieve a reliable, disruption-free operation scenario with 99% reliability.

In ITER, a burning plasma will be tested while the plasma is heated by fusion-born alpha particles. Energetic alpha particles from fusion reactions heat and sustain the burning plasma, but they also bring plasma instabilities. It is clear that a significant loss of alpha/fast ions may degrade the plasma heating and current drive efficiency and may lead to significant loss of plasma performance. Theory and modeling predictions of alpha particle control and reliable alpha particle diagnostics are needed to be explored with simple technical solutions in preparation for the ITER burning plasma physics experiments in the near future.

High-performance steady-state H-mode operation with a fusion power of 500 MW over 400 s is the premier goal of ITER operation, which needs integrated control for many physical quantities from both the core and edge. To maintain high core plasma performance, efficient plasma heating and current drive in the H-mode scenario, high bootstrap current fraction and low impurity concentration are required simultaneously. In the edge and scrape-off layer regions, it is essential to have a controllable plasma surface interaction by using strong gas puffing to keep a low ion temperature to sustain low impurity generation and physical sputtering on the tungsten divertor. The current superconducting tokamak devices such as EAST and KSTAR should make continued effort on these problems and achieve steady-state high-performance plasmas.

In ITER, over 10 MW/m² peak heat fluxes that nearly reach the present technological limit are foreseen and the ITER mono-block divertor solution is not sufficient for DEMO. The future DEMO divertor working conditions are very challenging and need both new physical (detached plasma and new divertor configuration) and technical
Large-scale R&D that addresses the major challenges for CFETR should start as soon as possible, such as high-performance superconducting magnets including high-Tc magnets, advanced new Nb3Sn magnets, megawatt continuous wave gyrotron development, a steady-state negative neutral beam injection system, a remote handling (RH) system, a new DEMO divertor, advanced fusion materials and breeding blankets. Several testing facilities are needed for simulating future CFETR operation without a nuclear environment, such as a superconducting testing facility, a vacuum vessel for installation and removal by RH, a tritium exhaust testing facility, and a DT neutron system for blanket testing. With successful construction of these R&D and testing facilities, together with further engineering testing, large-scale simulation and modeling and tokamak experiments, a more solid basis will be established for the beginning of CFETR construction.

**REFERENCES**

1. Hawryluk RJ, Batha S and Blanchard W et al. Rev Mod Phys 1998; 70: 537–87.
2. Keilhacker M, Watkins ML and the JET Team. Nucl Fusion 1999; 39: 209–34.
3. Oyama N, Isayama A and Matsunaga G et al. Nucl Fusion 2009; 49: 065026.
4. Wan Y, Li J and Weng P et al. Plasma Sci Technol 2006; 8: 253–4.
5. Lee GS, Kim J and Hwang SM et al. Nucl Fusion 2000; 40: 575–82.
6. ITER Physics. Nucl Fusion 1999; 39: 2137–74.

**Figure 1.** Roadmap for Chinese magnetic confined fusion (MCF) development.

Roadmap for Chinese MCF development

ITER

- Phase I: Q=1-5, steady state, TBR>1, >200 MW, 10 dpa
- Phase II: DEMO validation, Q>10; CW, 1 GW, >50 dpa

CFETR

- Phase I: Q=10, 400 s, 500 MW, hybrid burning plasma
- Phase II: Q=5, 3000 s, 350 MW, steady-state burning plasma

EAST

- Advanced PFC, steady-state advanced operation

HL-2M

- Advanced divertor, high power H&CD, diagnostics

J-TEXT

- Disruption mitigation, basic plasma

Roadmap for Chinese magnetic confined fusion (MCF) development.