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ORIGINAL RESEARCH

Present State of Chinese Magnetic Fusion Development and Future Plans

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Abstract

Chinese magnetic confinement fusion (MCF) development has made significant progress during the past decade. With successful construction and operation of the EAST superconducting tokamak, China is playing a key role in advanced steady-state operations towards the next step ITER. The Chinese Fusion Engineering Testing Reactor (CFETR) is the next device for the Chinese MCF program which aims to bridge the gaps between the fusion experiment ITER and the demonstration reactor DEMO. Fusion power of CFETR will be in the range of 200 MW to over 1 GW. It will be operated in two phases: Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power up to 200 MW. The second phase aims for DEMO validation with a fusion power over 1 GW. The Chinese government has approved to proceed with the CFETR engineering design, and the project started on December 2017. Roadmap of Chinese MCF, gaps for construction and operation of CFETR, efforts to fill these gaps and speedup the fusion energy application in China are presented.

Keywords Fusion · Tokamak · Reactor

Introduction

Energy shortage and environmental pollution are two critical issues for human beings in the twenty-first century, especially for fast developing countries, such as China. The Chinese economy has been achieving 8–9% annual growth for the past 30 years. Aiming at a moderately developed economy for China as the target in 2050, the total energy requirements will increase by a factor of two in the next 30 years. Meanwhile, China's resources are poorly balanced. It produces up to 11% of the world's coal, 13% of its hydropower, but only 2.5% oil and 1.2% gas worldwide. This means that Chinese energy consumption structure is and will continue to be mainly based on coal, nearly 67% at the moment and targeting 50% in 2050 which is extremely challenging. At the same time, China suffers from poor efficiency in turning fossil energy into economic output, just about one seventh of that in Japan. China is already becoming the largest CO_2 producing country due to the fast development.

There is an urgent need for new sustainable energy to meet the fast growing demand for clean energy. It seems at present that we, humankind, do not have many choices in energy. Fusion is one of the few options to satisfy the requirement of large scale sustainable energy generation and global warming suppression. It has been proved to be a potential source of secure, inexhaustible and environmentally friendly energy, and; therefore, it must be developed as quickly as possible.

Fusion research started 50 years ago and significant progress had been made since then. More than 10 MW fusion power has been produced in TFTR [1] and JET [2–4], the world largest tokamak. Long pulse high performance H-mode discharges have been routinely obtained in JT-60U [5]. High performance hot plasma has been kept for more than 1 h in the Large Helical Device (LHD) [6, 7]. EAST [8, 9] and KSTAR [10, 11], the new generation of fully superconducting tokamaks, have operated for 10 years and significant progress has been obtained for steady-state high performance operation. Fusion research

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comes to a new era with joint efforts by China, Europe, India, Japan, Korea, Russia and the United States on the International Thermonuclear Experimental Reactor (ITER) [12, 13].

Through international collaboration, ITER brings fusion research to a state of readiness to explore burning plasma, which is the condition for future reactors. For the coming 10 years, the world fusion community will make efforts to accomplish ITER construction and start operation. Meanwhile, almost every ITER partner is considering to start a DEMO program with successful experience gained from ITER. Each party from ITER partners has its own thoughts on how to approach DEMO and different efforts have been made. The Chinese approach, future development roadmap, efforts and future plans will be presented in this paper.

Present State of CN MCF

Chinese MCF started in the 1960s. Even though it was very difficult from the very beginning, it has slowly but steadily grown with two major research institutes (Southwestern Institute of Physics, SWIP, and Academy of Science Institute of Plasma Physics, ASIPP) playing key roles for MCF development. For training of fusion scientists, plasma physics and fusion technology groups have been established in the University of Science and Technology, Tsinghua University, Dalian University of Technology, and Huazhong University of Science and Technology. Plasma and fusion related courses have been set up since the 1980s.

The Chinese MCF program aims at developing fusion energy from its very beginning. It passed three major phases. The first phase started from 1970 to 1990 when basic plasma physics and multi approaches were adopted. Small tokomaks (CT-6, HT-6, KT-5 [14], HL-1 [15]), mirrors, and a spherical torus have been built. The main efforts focused on basic plasma physics and training young scientists. The second phase stared from early 1990 until the midpoint of 2000. During this period, Chinese MCF constructed mid-class tokamak HT-6M [16], HL-1M [17] and started superconducting tokamak activities. With help from Russian scientists, the T-7 tokamak with a superconducting toroidal field coil was successfully modified to become HT-7 [18, 19] with modern heating and current drive and diagnostics. The HL-2A tokamak has been built, which was modified from the German ASDEX. Significant progress has been made marked by the first H-mode plasma in HL-2A [20, 21] and 400 s 1 keV discharge in HT-7 [22]. Meanwhile, the first fully superconducting tokamak EAST was successfully constructed and the first plasma was achieved in 2006 [23].

The final phase stared from the end of 2006 when the Chinese government formally joined the ITER project. This is a turning point for the Chinese MCF program with fast development. A fully planned MCF program named the Chinese magnetic confined fusion energy project (CN-MCF), which consisted of a domestic fusion program and ITER participation has been started since then with stable funding from the central government.

The domestic part of the CN-MCF project consists of basic plasma and education, theory and simulation, with experiments and technology development. The top ten Chinese universities have been getting involved from the beginning of 2007 in basic plasma physics research and fusion education. Fusion talent training programs started with the joint efforts from the Ministry of Science and Technology, Ministry of Education, Chinese Academy of Sciences, and Chinese nuclear cooperation. Excellent students from the undergraduate level to the postdoctoral level have been selected with stable funding for joining special research activities. Over 200 master and 150 Ph.D. students each year have been involved in fusion research. The activities in the universities quickly move forward by building small basic machines for physics understanding, new diagnostics, alternative approaches, plasma wall interaction, material, and technology development, which complement the main efforts with two major facilities, EAST and HL-2A. A large-scale simulation program has started based on strong international cooperation. Several topics have been selected in different universities and two major fusion research institutes. With stable funding from the central government, both EAST and HL-2A tokomaks enhanced their research capacities with advanced H&CD, diagnostics, and large research teams.

Very good results have been obtained which addressed some very important key issues for future ITER. HL-2A focused on edge plasma physics [24] and MHD control [25], especially on ELM (Edge Localized Mode) mitigation [26, 27]. Large ELMs can lead to rapid erosion of first wall material, which is not acceptable for DEMO operation. Several ways have been developed towards robust control of ELMs. By using supersonic beam injection, ELM amplitude could be reduced by a factor of 3 and ELM frequency increased dramatically as shown in Fig. 1 [28]. A grassy ELM H-mode could be achieved by using PAM LHCD antenna in HL-2A as shown in Fig. 2 [29]. Utilizing its superconducting long-pulse capability, EAST concentrates on exploring advanced high performance steady-state plasma operations. MA plasma and up to 400 s divertor plasma discharges have been routinely obtained in EAST. The research topics in the CN-MCF project quickly moved to the frontier of world tokamak research. Over 500 Science Citation Index (SCI) journal papers have been published each year during the past 5 years.

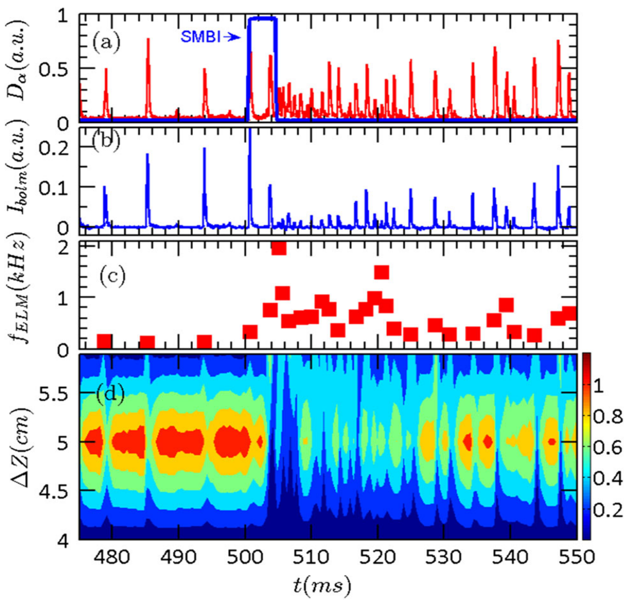

Fig. 1 ELM mitigation by SMBI

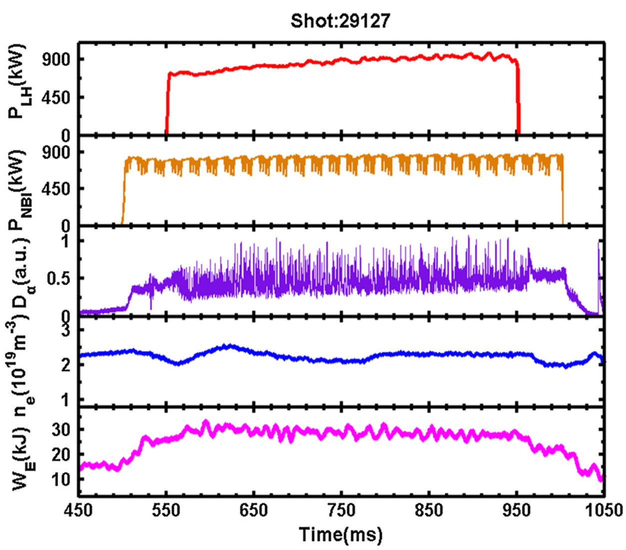

Fig. 2 Grass ELMH-mode by PAM antenna

Since China formally joined the ITER project, a domestic agency (DA) for ITER (ITER-CN DA) has been formally established in 2007 under the Ministry of Science and Technology. A well-established hardware fabrication program has been making steady progress in close cooperation with ITER-IO. There are 12 procurement arrangements (PA)s assigned to ITER-CNDA. Through joint effort within ITER-CN DA, two major leading institutes (ASIPP and SWIP) and Chinese industry, Chinese ITER PAs were delivered with fully qualified quality and always on schedule. TF conductor PA has been finished and 11 out of 12 PAs started massive production and more than 50% of total PAs have been completed. China also helps ITER-IO and other parties in successful construction of their components on time. China will finish all PAs with fully qualified quality on schedule towards the successful construction and operation of ITER.

CN-MCF Roadmap

After 3 years of discussion within the CN-MCF community, consensus has been achieved towards the early application of fusion energy in China. A CN-MCF roadmap is shown in Fig. 3. Based on the accomplishments of the last 50 years, international fusion society has reached a consensus on building a superconducting tokamak ITER. This has been done by participating in, sharing and assimilating the techniques in ITER. ITER technology provides the foundation for the CN-MCF roadmap. The next step fusion reactor will be the superconducting tokamak reactor. The advanced and reliable steady-state operation scenario will be achieved by a combination of bootstrap current and auxiliary driven current. The goal of tritium self-sufficiency will be obtained with the techniques of highly efficient advanced tritium breeding blankets and a large-scale tritium factory, together with advanced safety operation of burning plasma and material technology development. Efforts will be made to explore and tackle the unprecedented and important challenges of the advanced fusion power plant in the future with the aspects of science, technology, engineering and safety. Fusion power of CFETR will be in the range of 200 MW to over 1 GW. It will be operated in two phases. Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power up to 200 MW. The second phase aims for DEMO validation with a fusion power over 1 GW. By conducting engineering test and demo validation experiments respectively in two phases on CFETR, tremendous leaps from engineering test reactor to demo reactor could be achieved and eventually the prototype of power plant will be build based on CFETR.

The near-term targets of CN-MCF are aimed at (1) The establishment of an advanced platform (EAST, HL-2M, J-TEXT) for plasmas physics research in China. (2) Developing the key technologies for the construction of ITER and CFETR and (3) Design CFETR. The construction of CFETR is planned to be conducted in the 2020s and finished in the 2030s. The target fusion power of CFETR in phase I is about 100–200 MW. The steady state operation and tritium self-sufficiency will be explored in this phase in complement with ITER $Q=10$ operation. The Phase II of CFETR will be finished in the 2040s. Experimental studies conducted in phase II will validate the most important issues for tokamak DEMO under a fusion power output of

Fig. 3 Roadmap of the magnetic-confined fusion development in China

over 1 GW. The prototype of fusion power plant (PFPP), which will be built from 2050 to 2060, is going to be a final step in the roadmap towards a commercial power plant.

Progress of EAST and Role for ITER and CFETR

EAST shown in Fig. 4 is a modern superconducting tokamak with technologies similar to ITER. EAST has internationally unique capabilities to address many of the critical physics and technology issues that ITER will encounter. EAST is capable of pulse durations beyond 400 s with high power electron heating to challenge power and particle handling at high normalized levels (10–20 MW/m^{-2}) comparable to ITER.

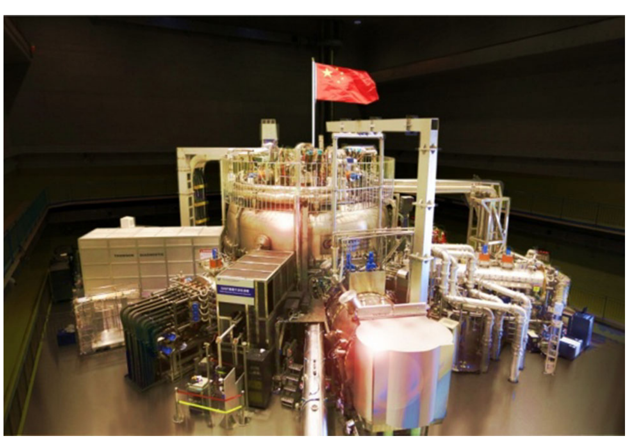

Fig. 4 EAST tokamak

Significant progress has been made on EAST during the past 10 years on both physics and technology fronts towards the long-pulse operation of high-confinement plasma regimes [30]. EAST has been upgraded with more than 30 MW of CW RF and long pulse ($\sim$ 100 s) NBI heating and current drive (H&CD) power, along with over 80 advanced diagnostics, two internal cryopumps, a top divertor with ITER-like W monoblock design and resonant magnetic perturbation (RMP) coils, which will enable EAST to investigate steady-state/400 s long-pulse H-mode operation with dominant electron heating. In addition, EAST will be facing challenges of power and particle handling with high normalized levels of particle and heat fluxes on the divertor comparable to ITER. Over 400 s divertor L-mode plasma discharges [31] and 100 s long-pulse H-modes with duration over 100 times of the current diffusion time have been achieved in EAST by the combination of lower hybrid current drive (LHCD), ECRH and RF heating as shown in Fig. 5 [32, 33].

During past 5 years, remarkable efforts have also been made in mitigating type-I ELMs in a stationary-state H-mode plasma with multi-pulses of supersonic molecular beam injection (SMBI), LHCD, lithium granule [32] and deuterium pellet injection, as well as RMPs, thus potentially offering a valuable means of heat-flux control for next-step long-pulse fusion devices. Long-pulse H-mode discharges with $H_{(98,y2)} \sim 1$ have been obtained either with

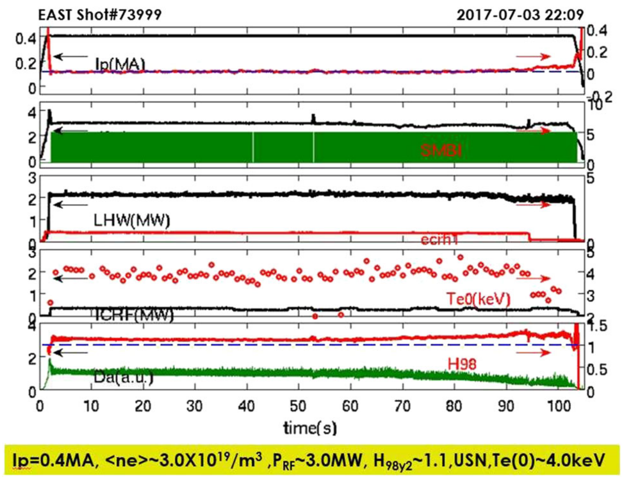

Fig. 5 EAST 100 s H-mode discharge

ELM mitigation or in a small EMLy regime accompanied by a new electrostatic edge coherent mode, which is present in the steep-gradient pedestal region and plays a dominant role in driving heat and particles outwards [33]. The peak heat load on the divertor, which is over 10 MW/m^2 during type-I ELMs, is reduced down to 2 MW/m^2 either by SMBI or LHCD. It is found that ELM mitigation with SMBI is due to enhanced particle transport in the pedestal, correlated with large-scale turbulence and strongly anti-correlated with small-scale turbulence, while LHCD can create 3-D distortion of magnetic topology similar to the conventional RMPs and exhibit a positive effect on mitigating the ELMs and redistributing power deposition on the divertor targets [34]. With continued strong and sustained government support, EAST could become a leading research device for plasma science and technology on critical issues affecting ITER in the near- and long-term under steady-state operation conditions.

EAST has made important contributions to ITER, especially in the area of operation with superconducting coils. Both high current (1 MA) and long pulse (400 s), albeit at reduced current, have been achieved. This demonstrates the ability to navigate the complex operating poloidal field coil constraints through the startup phase as well as the noise issues on magnetics associated with long pulse operation. Data from EAST along with results from KSTAR provides a sound foundation for ITER operation with superconducting coils for long pulse operation.

In addition to the high $Q=10$ operating mode on ITER, development of the steady-state operating mode ($Q=5$) in ITER is an important mission element. This entails combining high β_n, energy confinement time (H_{98} factor) and good MHD stability. Simulations of the ITER steady-state operating mode indicate that there is a complex interplay between MHD stability, fully non-inductive current drive and high Q operation. Further upgrades to the ITER heating and current drive system may be necessary and, in particular, the need for lower hybrid current drive is an outstanding question. EAST with its emphasis on RF heating tools will provide critical near-term results associated with the steady-state operating mode on ITER. These will be very challenging experiments entailing the integration of high performance operating conditions, high heat flux to plasma-facing components and RF heating and current drive.

Currently, EAST is utilizing its unique capability to study very long high performance (high β_n, and high bootstrap current fraction) pulses, in particular, with changes of current profile and its stability and control. Comparison of such experimental results with theoretical modeling will be an important contribution to ITER operation.

Evaluation of long-pulse high performance discharges with respect to the prediction of disruptions and consequent mitigation will be a long-term benefit to ITER. Whether lower hybrid current drive will provide significant seed current for runaway electron generation will be assessed in the future which could provide a much faster control method than massive gas injection.

EAST will also address another key research topic on ITER, namely long pulse operation with high heat flux (>10 MW/m^2) to the plasma facing components. Data on long-pulse operation will augment and extend results from AUG and JET with tungsten. They will evaluate the technology used on ITER for both the tungsten mono-block and the in-vessel coils. One area where the experimental activities will extend the earlier results is in hydrogenic retention and dust production with a tungsten divertor, including technology developments to remove dust. These are important topics for ITER.

For the future Chinese Fusion Engineering Testing Reactor (CFETR), EAST will also play a key role for truly steady-state operation and power and particle handling on the divertor. Steady-state high performance operation is the top priority of EAST. Figure 4 is a 100 s H-mode with RF only, which is one of several possible operation scenarios for CFETR. Up to 80 s duration with loop voltage kept at zero, externally driven current at 75% and bootstrap current at 25% are achieved, which could be used for a CFETR early operation scenario. Advanced divertor configurations (Snow-flake, quasi-snow-flake, radiation divertor) have been tested which showed a heat load reduction of 3–10 times on EAST up to 20 s pulse length. Technology development for handling higher heat load is also under way. Both ITER mono-block W tile and flat W tile have been fully tested up to 20 MW/m^2 for more than 5000 cycles without failure. The combination between physical advanced divertor and technological improvement will provide a more solid base for solving the power and particle handling problems for the CFETR Phase II when fusion power will be over 1 GW and power exhaust will be a great challenge.

CFETR

CFETR is the next device for CN-MCF program which aims to bridge the gaps between the fusion experiment ITER and DEMO [35–37]. CFETR will be operated in two phases: Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power up to 200 MW [38]. The second phase aims for DEMO validation with a fusion power over 1 GW.

CFETR's objectives are as follows:

- A good complementarity with ITER
- Demonstration of a full cycle of fusion energy
- Demonstration of a full fuel cycle of Tritium aiming at a tritium breeding ratio (TBR) over 1.0
- Long pulse or steady-state operation with duty cycle about 0.3–0.5
- Based on the existing ITER physics and technology together with advanced new technology development
- Exploring physical and technical options for DEMO with an easily changeable internal components by remote handling technique
- Addressing the DEMO relevant issue via a step by step approach
- Exploring the technical solution for licensing a DEMO fusion device

Phase I of CFETR is mainly based on the present physics and technology of ITER with moderate fusion power up to 200 MW. Detailed concept design and some R&D including full size VV manufacture, 1/6 CS Nb3Sn coil fabrication and testing, key T technology development, fist wall (FW) and divertor materials have been carried out. The site view of CFETR is shown in Fig. 6.

Present design and R&D activities are moving towards phase II with fusion power over 1 GW. Operation scenarios have been assessed by integrated modeling based on advanced H-mode physics with high magnetic fields (up to 7 T). Advanced configuration of divertor (field expansion with impurity radiation to reduce heat load) together with technical improvement of heat load removing capacity (over 20 MW/m^2) are adopted for the phase II divertor solution.

High Tic superconducting magnets made of BSCCO (Bismuth–Strontium–Calcium–Copper–Oxide) 2212 cable-in-conduit-conductor (CICC) magnets [39] were chosen for

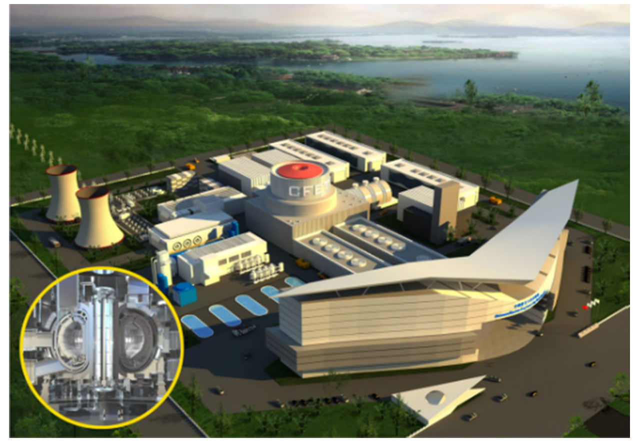

Fig. 6 Bird's view of CFETR site

the possible solution of the CS coil. Present progress of the 2212 whirl and the CICC conductor is very promising, which is well beyond present Nb_3Sn performance. A high power CW 4.6 GHz tube for LHCD has been developed. The 170–230 GHz 1 MW gyrotrons are under development. He cooled and water cooled configurations are two primary candidates for blankets. Vertical replacement of blankets and bottom replacement of divertor are the stratagem of RH. Eight different RH tools have been developed with a payload of a few tens of kg at the moment [40, 41]. Large scale RH manipulators will be developed.

CFETR concept design has been carried out for nearly 4 years. A previous approach is based on a smaller machine with major radius R=5.7 m, a=1.6 m and B_T=4–5 T [37]. Plasma performance with fusion power up to 200–500 MW can be obtained for the first phase. For smooth transfer to Phase II with the same machine, aggressive advanced plasma performance has to be chosen to meet its objective and is also very difficult for power handling on divertor and blanket, and material damage by high heat flux and neutron.

For meeting both phase I and II targets with achievable technical solutions, a new design has been made by choosing a larger machine with R=7–7.2 m, a=2.2 m, B_T= 6–7 T. Over 1GW fusion power can be produced and technically feasible which will be transferred from Phase I to Phase II with the same machine.

Table 1 shows the main operation scenarios of phase I (A1–A2) and phase II (A3–A4) for steady-state and hybrid mode with ohmic heating (A5). All the plasma parameters in Table 1 show reasonable values which either have a sound experimental database or can be achievable within future efforts. The whole research period of CFETR experiments will be placed in a staged process. CFETR plasma will start from a very conservative starting point, say 50–100 MW fusion power after the H/D phase of full H&CD power commissioning. CFETR can start operation

Table 1 CFETR plasma performance for Phase I (case A, B) and II (C)

CFETR		A.1 200 MW SS	A.2 500 MW SS	A.3 1 GW SS	Case A.4 1.5 GW SS DEMO	Case A.5 1 GW hybrid
Minor radius	A m	2.19	2.19	2.19	2.19	2.19
Major radius	Ro m	7.00	7.00	7.00	7.00	7.00
Plasma elongation	κ	2.00	2.00	2.00	2.00	2.00
Fusion power	Pf MW	196	503	1003	1446	1053
Power to run plant	PI MW	181	190	179	165	183
Gain for plant	Plant	0.67	1.34	2.63	3.99	2.70
Fusion/Paul	Plasma	2.84	7.29	17.21	30.31	17.74
Net electric power	Pent MW	− 60	65	292	494	310
Neutron load	Pan/Wall	0.21	0.53	1.06	1.53	1.11
Normalized beta	Beta	1.27	1.69	2.50	3.0	2.10
Bootstrap fraction	fibs	0.40	0.50	0.60	0.75	0.60
Ohmic fraction	form	0.0	0.0	0.0	0.0	0.30
Plasma current	$I_{p,}$ MA	12.01	12.93	12.93	11.91	12.93
Field on axis	B_o, T	6.0	6.54	6.54	7.05	6.54
Field at conductor	B_{Sc}, T	12.48	13.62	13.62	14.68	13.62
Ion temperature	Ti(0), keV	26.00	32.00	37.00	34.00	17.50
Electron temperature	Te(0), keV	26.00	32.00	37.00	34.00	17.50
Electron density	$n(0)\ 10\times^{14}$	0.47	0.64	0.78	1.02	1.51
Ratio to GW limit	Near/mgr.	0.39	0.50	0.61	0.86	1.17
Z_{eff}	Z_{eff}	2.45	2.45	2.45	2.45	2.45
Stored energy	MJ	276	465	657	789	599
Auxiliary POWER	Paul MW	69	69	58	48	59
TauE_net(s)	Tau_E_net	2.95	3.65	3.76	3.45	3.10
H factor	H_{ITER}98Y2	1.19	1.47	1.43	1.43	1.17
Power over R	P/R	8.74	10.97	13.89	18.20	16.06
q95 Iter	q95_iter	6.07	6.15	6.15	7.20	6.15

without waiting for ITER $Q=10$ DT data for its first phase. After phase I, together with ITER $Q=10$ 400 s experience, CFETR could move forward smoothly to its second phase for getting fusion power over 1 GW. The major challenges are steady-state operation, tritium breeding and power and particle handling with fusion power over 1 GW in an integrated way.

For Phase I, a conservative plasma performance has been chosen which is based on the achievable experimental database shown in Table 1 case A1 (200 MW) and A2 (500 MW). Efforts will be focused on the tritium breeding and long pulse or steady state operation. The challenges for material, heat exhaust, and plasma disruption are relatively weak compared with ITER.

For phase II with fusion power over 1GW, operation scenarios have been assessed based on the present advanced high confinement with higher bootstrap current fraction (60–75%), high magnetic fields. Low plasma current (~ 12 MA) was chosen for easy long pulse or steady state operation, and less disruption mitigation challenge. High frequency electron cyclotron resonance heating (170–230 GHz) and lower hybrid current drive (5–7.5 GHz) from high field side launch together with off-axis negative-ion neutral beam injection will be used for achieving steady-state advanced operation. Advanced configuration of divertor (field expansion with impurity radiation to reduce heat load) together with technical improvement of heat load removing capacity (over 20 MW/m^2) are adopted for the phase II divertor solution.

Steady-State Operation

CFETR steady-state operation is achieved by using a combination of three heating and current drive systems, i.e., neutral beam (NB), electron cyclotron waves (EC) and lower hybrid waves (LH), each offering unique properties. Self-consistent modeling is performed using a multi-dimensional code suite anchored by the transport code

TGYRO with the physics integration facilitated by an automated framework OMFIT [42, 43]. Within the code suite, physics-based models TGLF and NEO are used to calculate turbulent and neoclassical transport, the evaluation of sources and sinks as well as the plasma current evolution are performed using ONETWO, and the equilibrium is updated using EFIT. The pedestal is consistent with the EPED model. Finally, when iterating among transport, equilibrium and H&CD toward steady state, the auxiliary power is continuously adjusted to keep the plasma fully non-inductive.

Two basic steady state scenarios have been developed [38]. The first scenario with steady state plasma can be maintained by off-axis NBI, top launch EC together with bootstrap current drive. The second scenario is using high field side LHCD, top launch EC together with bootstrap current drive. Furthermore, external CD is used to control $q_{\min}>2$ to avoid low n MHD mode instabilities that might lead to disruptions, and to access possible improved confinement at higher $\beta_{N.}$

The results for the larger device are consistent with those of the smaller device ($R=5.7$, $a=1.7$) in that higher NB power, which drives stronger rotation, improves confinement leading to higher temperatures thus higher fusion gain. Furthermore, a higher β_N together with a higher q_{95} (due to a higher B_T and lower I_p) enables a higher bootstrap current fraction of 75%. Because of the much lower auxiliary power required, even if the achievable confinement is lower, we can still approach the target fusion power by moderately increasing the auxiliary power and/or plasma density. Based on this result, we would expect the achievable fusion gain/power for the larger device to increase substantially if β_N could be further increased to $\sim$ 3 and beyond as shown in Table 1.

T Breeding

T breeding is the one of the most important issues in CFETR. Efforts have been made in both blanket design and T-plant design to get optimized T breeding. Two options of blankets have been designed [44–46] and compared for both physics and technical issues involved in each option. The helium gas cooled T breeding blanket was chosen as the premier candidate blanket configuration. Water-cooled coolant blanket is chosen as a second choice. Both options can meet TBR>1 requirement under 3D simulation.

For helium cooled ceramic breeder blanket design, a modularized breeding unit and a multi-layer back plates manifold have been used for concept design. Li_4SiO_4 was chosen as the tritium breeder, Be as the neutron multiplier, RAFM steel as the structural material, and tungsten as the armor material of the first wall. An 8 MPa helium gas was chosen as coolant with 300 °C inlet/500 °C outlet. With optimized design, 3-D neutronic simulation shows that TBR in the blanket could be 1.21 for Phase I (200 MW fusion power) and 1.15 for phase II (1 GW) [44, 47].

For water cooled ceramic breeder blanket design, the cooling plates and the breeder zone are parallel to the FW. A compact coolant design was chosen, which could enlarge the breeder zone. Purged gas is directed in the toroidal direction to reduce its pressure drop. Li_2TiO_3 are chosen as the tritium breeder and $Be_{12}Ti$ as the neutron multiplier, RAFM steel as the structural material, and tungsten as the armor material of the first wall. The 15.5 MPa pressured water was chosen as coolant with 285 °C inlet/325 °C outlet. With optimized design, 3-D neutronic simulation shows that TBR in the blanket could be 1.2 for Phase I (200 MW fusion power) and 1.1 for phase II (1 GW) [48, 49].

Divertor and Heat Exhaust

During phase II of CFETR, heat load on the divertor will exceed 20 MW/m^2 if a standard SN divertor configuration is adopted, which is beyond the present technology limit. For reducing such high heat load and particle flux on the divertor, two extra superconducting coils (DC1 and DC2) shown in Fig. 7 have been chosen for advance divertor design. By using these two coils together with other PF coils, either an x-divertor or a snowflake divertor [50] can

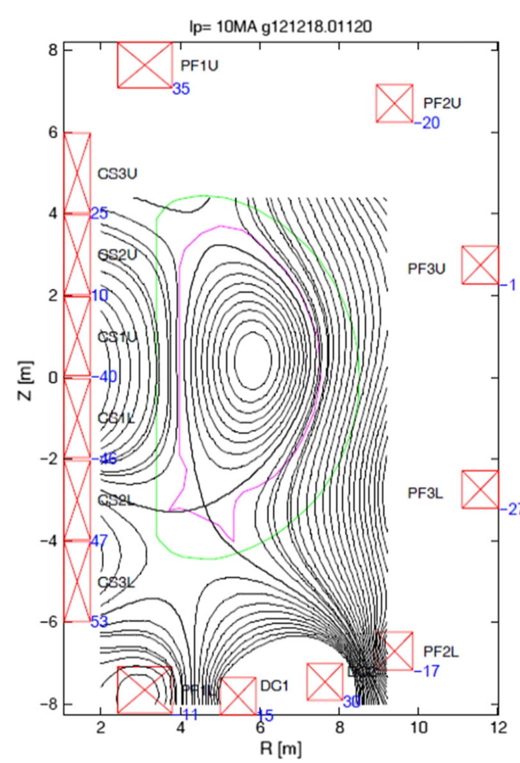

Fig. 7 SF divertor

be realized with field expansion 3–8 times. Figure 7 shows a configuration of CFETR snowflake (SF) with 12 MA plasma current and field expansion about 4.6.

Externally seeded impurities in the core can help partially radiate the heat before it reaches the divertor. The seeded impurities, however, cannot be so large as to negatively impact the plasma performance in the core. We have investigated the maximum core impurity radiation achievable for the CFETR baseline without degrading the core performance. A chain of events specific to steady-state operation occurs when argon is injected. The added impurity raises Z_{eff}, which lowers the current drive efficiency. By using impurity seeding, up to 30% of the core heat could be radiated without significantly affecting the core parameters [51].

Gaps for CFETR and Efforts Made

There are still many technical difficulties for successful construction and operation of CFETR in plasma performance, enabling technologies, material and component performance and safety issues, and time is needed for further development. According to the input requirement of DEMO and power plant as shown in Fig. 8 (r-solution is desirable, R-solution is a requirement), some key scientific and technical issues will be addressed in ITER as shown in the column under ITER. There are still many issues which only can be solved until DEMO and power plant are eventually as shown as in the right 3 columns.

For Phase I of CFETR, since plasma performance is relatively low compared with ITER and present advanced tokamaks, such as DIII-D and ASDEX-U, successful construction and operation is not so challenging due to the strong scientific and technological basis of ITER except for high field superconducting magnets. The major gaps for Phase I are the steady-state or long pulse operation (few hours, longer than T fuel cycle), T breeding. Due to conservation plasma parameters (shown in Table 1), power exhaust and material would not be an obstacle except for plasma wall interaction. There are no experiments at the moment for a W divertor operating at 1000 °C for hours. The recycling, particle balance and impurity (sputtering, erosion) behaviors during hours of operation remain unclear. Experimental validation needs to be done in the near future on EAST and a CW high flux plasma wall interaction devices.

For CFETR Phase II, all scientific items in Fig. 8 will be highly challenging together with the advanced breeding blanket, which should not only breed enough tritium, but also perform at very high thermal electricity conversion efficiency. Efforts are urgently needed to speed up the CFETR development.

To speed up fusion development and fill the gaps especially for CFETR Phase II, the following essential steps have been started simultaneously:

1. Strengthen CFETR physics activities. This includes large scale simulation and modeling of CFETR plasma performance in an integrated manner together with experimental validation. Theory and simulation activities, which are carried out in different universities and

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Plasma performance	Disruption avoidance	2	3		R	R	R
	Steady -state operation	2	3		r	r	r
	Divertor performance	1	3		R	R	R
	Burning plasma (Q>10)		3		R	R	R
	Start up	1	3		R	R	R
	Power plant plasma performance	1	3		r	R	R
Enabling technologies	Superconducting machine	2	3		R	R	R
	Heating, current drive and fuelling	1	2		3	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Remote ha ndling	1	2		R	R	R
Materials, Component performance & lifetime	Materials characterisation			3	R	R	R
	Plasma -facing surface	1	2		3	4	R
	FW/blanket/divertor materials		1	1	3	4	R
	FW/blanket/divertor components		1	1	2	3	R
	T self sufficiency		1		3	R	R
Final Goal	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

Fig. 8 Present state and Gap analysis towards fusion power plant [52]. Input: r-solution is desirable, R-solution is a requirement. Output: 1-will help to solve the issue, 2-may solve the issue, 3-should solve the issue, 4-must solve the issue

institutes, will be more focused on the mission and task of CFETR. Resources will relocated with CFETR priority and a well centralized CFETR team will be further enhanced.

2. Start CFETR detail engineering design. Although full burning plasma physics cannot be comprehensively addressed before ITER begins its long-pulse high-power DT experiments, many efforts for such an integrated detailed CFETR design (both for physics and an engineering point of view) are still required for advanced plasma control, steady-state operation, heating and current drive, fuel cycle, remote handling and maintenance, tritium breeding, electrical power generation, etc. In particular, reliability, availability and maintainability are important to minimize the maintenance time together with safety and licensing issues. Industry must play a major role in this work to ensure the feasibility and standards, in particular in the area of nuclear technologies, and to become acquainted with the system requirements of a fusion power plant. A joint team with fusion scientists and engineers, professionals from nuclear industry and safety has to be formed with the permission from CN-MOST for CFETR engineering design which is essential towards the realization of future commercial fusion power plants under a full industrial partnership. The CFETR engineering design effort has been granted by CN-MOST and the project started on December 2017.
3. Orienting EAST, HL-2A experiments towards CFETR experimental scenarios. EAST and HL-2A will increasingly become testbeds for simulation of future CFETR experimental tasks and operation scenarios. A joint experimental team together with modeling and simulation is to be formed during the next 5 years to addressing some key issues for future CFETR operation, such as advanced steady-state operation, ELM mitigation and control, off-normal event, energetic particle and runaway electron, particle and heat fluxes control.
4. Start large scale R&D for successful construction and operation of CFETR
   Large scale R&D will begin in the next 5 years based on the previous small scale activities, such as MW CW gyrotron development, negative neutral beam injection system, superconducting magnets which including 2212HTc, Nb_3Sn magnets, remote handling (RH) system, and breeding blankets (He and water cooled). Several testing facilities will be built for simulating the future CFTER operation without a nuclear environment, such as superconducting testing facility (for full size CS and TF testing), vacuum vessels for installing and removal by RH, a T exhaust testing facility, TBM fuel handling testing system, and small fluencs (10^{11}–10^{12}) DT neutron system for small blanket testing. With successful construction of these R&D and testing facilities, together with further testing and operation, a more solid basis will be established for startup of CFETR construction.
5. Speedup material application
   During the past 3 years, the Chinese fusion material community has made a preliminary CN material development roadmap which includes simulation and modeling, material (plasma facing material, structure material, functional material, superconducting materials) manufacture, material validation and a material testing facility. Detailed development plans are laid out for the next 5 years and each of the following 10 years. The main efforts will focus on material production, understanding of damage phenomena and experimental validation. First, the wall will use W alloy, structure material will be ODS RAFM steel which can stand up to 100 dpa at final target, ODS Cu and Cu alloy will be used for the heat sink with 20 dpa for the next 10 years and finally targeting 100 dpa for a possible solution. Future activities for fusion materials will be enhanced and moved forward with an integrated domestic effort.

Strengthen International Cooperation

ITER is a good example for broad international cooperation between countries consisting of countries for more than half the population of the world. During the past 10 years, very productive cooperation has been established between ITER parties and IO. For speedup of early application of fusion energy in China, international cooperation is one of most important key elements for success.

China is already getting benefit via joining the ITER project and also making its unique contributions for the ITER project. By continuing to strengthen the cooperation within the ITER project, China could acquire more knowledge on the technologies in construction of ITER for the next decade, which is over 70% of technology for construction of CFETR.

The Chinese domestic fusion program also gets benefit from wide international cooperation. China has a long history of very productive cooperation with US, Russia, EU and Japan. US-C cooperation continues for over 30 years. T-7 had been modified to HT-7 and ASDEX had been modified to HL-2A which made significant contributions not only in promoting the CN-MCF development but also for training a large amount fusion scientists and engineers. During past 5 years, the A3 program within Japan, Korea, and China progressed well. Joint laboratories between CN-France, CN-US have been established. Recently, very close and productive cooperation between DIII-D and EAST, EAST/HL-2A and WEST/ASDEX-U have been carried out, and specified joint working task groups have been built. Remote participation and 3rd shift operation of EAST from US and EU have been routinely carried out, and very productive results have been produced. For the next 10 years, this cooperation will be continued and strengthened by joint funding from each government.

CFETR welcomes any participant not only from ITER parties, but also to those who are willing to join and contribute. Joint efforts for design of CFETR started 5 years ago. Joint design teams between CN-GA, ASIPP–PPPL have been built and good results have been produced. Good cooperation between EU-DEMO and CFETR teams also started few years ago with routinely held bilateral workshops for design. For the next 5 years, the CN-MCF program is fully open to international parties or individuals to participate in CFETR design and R&D activities which will be certainly very helpful and more productive to speedup the CFETR project.

Conclusion

CN-MCF development has made significant progress for the past decade when China joined the ITER project in 2006. With successful construction and operation of the EAST superconducting tokamak, China is playing a key role for advanced steady-state operation towards the next step ITER. After joining the ITER project, the Chinese fusion industry also made significant progress on ITER PA with qualified components received by ITER-IO. Over 50% of CN-ITER-PA has been finished. The CN-MCF community has made a clear roadmap towards the early application of fusion energy in China. CFETR is the next device for the CN-MCF program which aims to bridge the gaps between the fusion experiment ITER and the demonstration reactor DEMO. CFETR will be operated in two phases: Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power up to 200 MW. The second phase aims for DEMO validation with a fusion power over 1 GW. Concept design of CFETR has been finished and small scale R&D started 5 years ago and has progressed well. The detailed engineering design, integrated simulation, and large scale R&D will continue for filling the gaps for construction and operation of CFETR. With strong government support and joint efforts between university, research institutes, industry and international partners, CN-MCF program will focus on establishing a more solid technical basis for successful operation of ITER and starting the construction of CFETR.

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