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Development of Key Technologies for Steady State Tokamak Reactor in JAERI

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Abstract

For steady state operation of the fusion plasma, technologies on high power negative ion based neutral beam injector (N-NBI), radio frequency heating system (RF), breeding blanket system, divertor system and super conducting magnet system have been developed in Japan Atomic Energy Research Institute (JAERI). In the development of the negative ion beam, the beam energy(1 MeV) required for the current drive in tokamak reactor has been demonstrated. The current density(20mA/cm^2) and the pulse duration of 140 hours, have been demonstrated individually. A high energy neutral beam of 350 keV, 5.2 MW has been successfully injected into JT-60 plasma from the N-NBI system which is the first NBI using the negative ions. Concerning the development of RF heating system, high power RF of 170 GHz, 0.5 MW, 6.2 s and 170 GHz, 450 kW, 8 s (3.6 MJ) have been successfully generated from the ITER prototype gyrotron. A breeding blanket first wall model (113 mm × 200 mm × 18 mm thick) made of reduced activation ferritic steel F82H has endured more than 5000 heat cycles at the surface heat flux of 2.7 MW/m^2. Near full-scale divertor mock-ups have been fabricated, and demonstrated to withstand a steady-state heat load of 5 MW/m^2 for more than 3,000 cycles, and 20 MW/m^2, 10 s for 1,000 cycles, which meets the ITER requirements. Super conducting magnet is one of the essential elements for steady state fusion reactor. A Central Solenoid Model Coil for ITER has successfully been fabricated which can be operated at a current of 46 kA, generating magnetic field of 13 T with the field variation speed of 1 T/sec.

Keywords:
tokamak, steady state, NBI, RF, blanket, divertor, superconducting magnet, negative ion, gyrotron

1. Introduction

For the steady state operation of the tokamak reactor, current drive systems to sustain plasma continuously, first wall and plasma facing components to withstand high heat flux from the plasma, and super conducting magnets for plasma confinement are essential.

A high power negative ion based neutral beam injector (N-NBI) is one of the candidates for the current drive system. Required beam energy and beam current of the negative ions are MeV class and several tens of ampere. To realize such powerful NBI, production and acceleration of the high power negative ion beam is one of the most crucial subjects in the development. An electron cyclotron radio frequency (ECRF) heating system is also one of the candidates to heat and sustain CW plasma. Development of a high power RF generator at hundred GHz band is a key issue to realize the system.

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In the development of the tokamak itself, blanket system and divertor system which receive a high heat flux are the crucial components to confine the steady state fusion plasma. Strong magnetic field is necessary to confine a high density and large volume fusion plasma stably. A large scale super conducting magnets is required to confine fusion plasma.

In the present paper, recent progress of R&D on additional heating system of NBI and RF heating system, blanket system, divertor system and super conducting magnet system is reported.

2. R&D Results for Steady State Tokamak

2.1 Negative ion based neutral beam injector

A high power negative ion source is a key element in the negative ion based NBI. Required beam energy and current are MeV class and several tens of ampere for D^-/H^- ions. Development of the high current negative ions has been concentrated on the cesium seeded volume type negative ion source. A high current of negative ions up to 18.5 A, H^- has been produced by optimizing the operating parameters of the JT-60 negative ion source [1]. This is almost a half of the negative ion current required for the ITER NBI. In order to demonstrate steady state negative ion beam generation, a small negative ion source has been operated for 140 hours continuously [2].

Concerning the high energy acceleration of the negative ions, a hydrogen negative ions have been accelerated up to the energy of 1 MeV in a prototype 1 MeV accelerator which consists of a multi-aperture, five stage electrostatic accelerator [3] under the ITER program. Figure 1 shows the prototype 1 MeV accelerator. It is experimentally confirmed that the negative ion beam optics agrees well with the design value by the beam trajectory simulation.

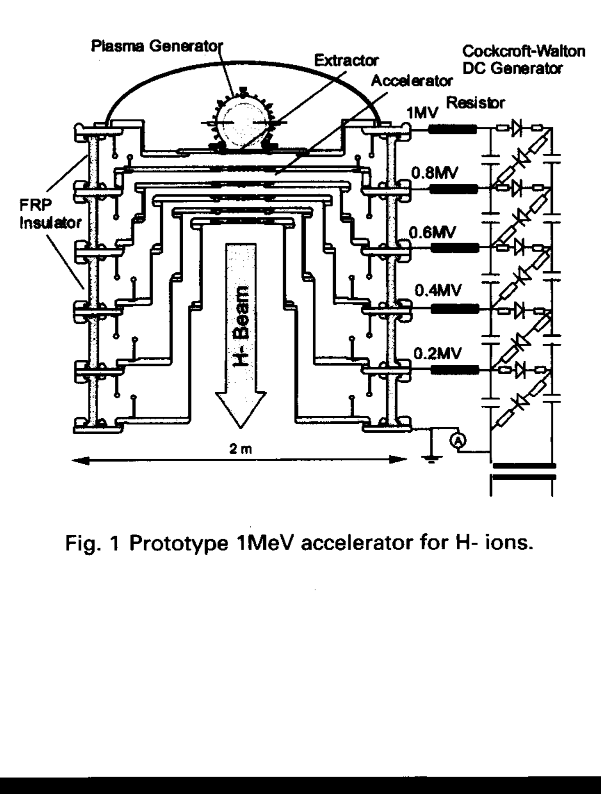

Fig. 1 Prototype 1MeV accelerator for H- ions.

The first NBI system based on the negative ions has been successfully developed for JT-60. The system aims at injecting 500 keV, 10 MW neutral beam for 10 s using two ion sources. The system is under operation for plasma heating and current drive experiment. A 350 keV, 5.2 MW neutral beam has already been injected into the plasma [4].

Figure 2 shows the progress of the negative ion beam development. There are two approaches, i.e. high current approach and high energy approach. Based on the results of 1 MeV acceleration and the high current production in the JT-60 ion source, 1 MeV/40 A ion source for fusion tokamak will be realized.

2.2 ECRF system

The most critical issue of the ECRF system is a gyrotron. Major milestone of the ITER gyrotron development is 1 MW, CW with 50 % efficiency at 170 GHz. There are three major points to develop the ITER gyrotron. The first is a high order volume mode cavity for 1 MW power in CW. The second is energy recovery system to improve the efficiency of the gyrotron which

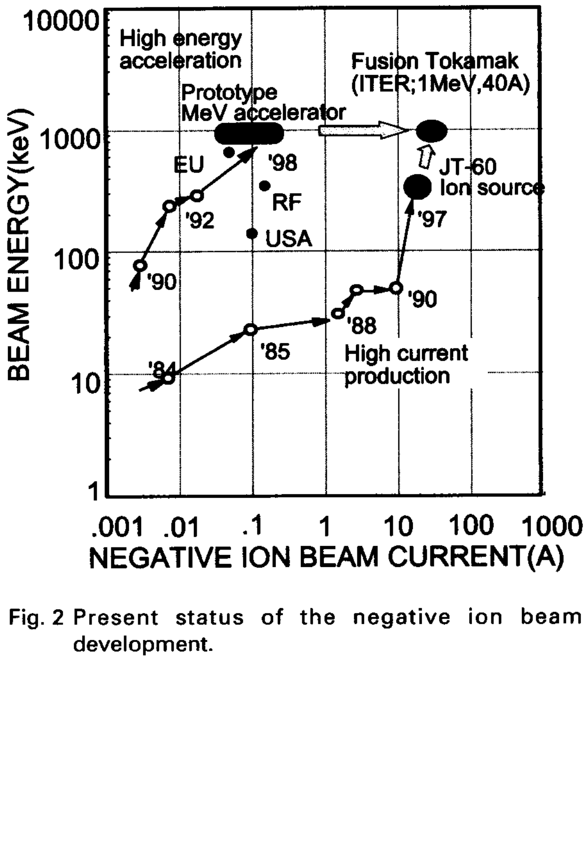

Fig. 2 Present status of the negative ion beam development.

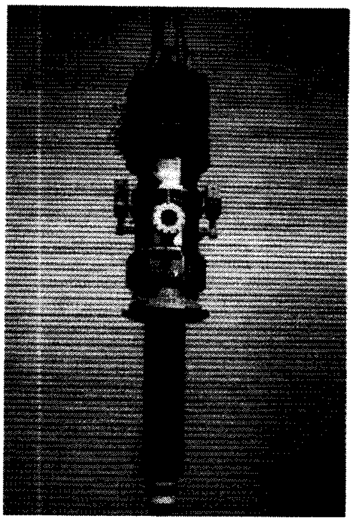

Fig. 3 World first diamond window gyrotron.

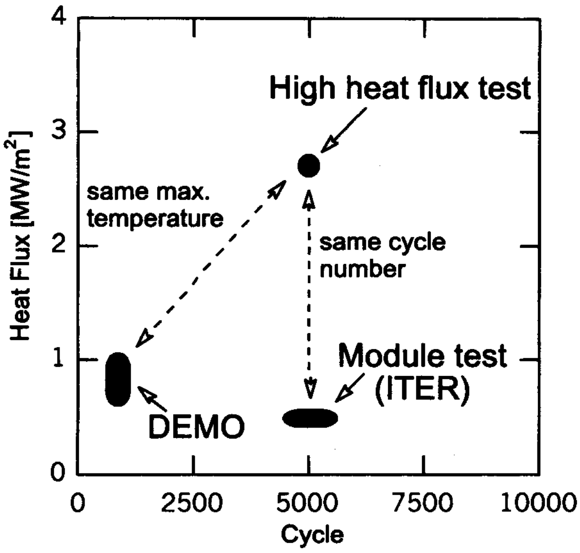

Fig. 4 Condition of the high heat fluz test.

was demonstrated at 110 GHz [5]. The third key element is output window.

A high volume mode cavity gyrotron is necessary to output 1 MW, CW at 170 GHz to reduce the heat load to the cavity. The conventional cylindrical cavity and $TE_{31,8}$ mode were selected by optimization. The ITER prototype gyrotron with this $TE_{31,8}$ cavity had been tested but power output was limited to 500 kW/0.7 sec because of the temperature rise of the sapphire or Si_3N_4 window. A new window has been developed to overcome the limitation with synthetic Chemical Vapor Deposition (CVD) diamond, which has ideal performance for a gyrotron window [6].

The picture of the 170 GHz gyrotron with the diamond window is shown in Fig.3. After the quick conditioning of the gyrotron, the pulse length gradually extended up to 6.2 sec with the output power $P_{rf} \sim 0.52$ MW [7]. The efficiency reached to 32 % with a depressed collector. Temperature rise of the window at 500 kW was up to ~150°C at the center. It was almost saturated after 5 sec. The loss tangent of the diamond disk of this gyrotron is relatively high as $\tan\delta \sim 1.3 \times 10^{-4}$. High quality diamond disk whose $\tan\delta$ is 2×10^{-5} has already been developed. If it is used, temperature increase at 1 MW would be only 30°C in steady state. These novel technologies promise the prospect of achieving 1 MW, CW at 170 GHz gyrotron for steady state operation.

2.3 Blanket system

A ceramic breeder blanket development has been widely conducted in JAERI from fundamental researches to engineering researches with scaleable models. Two types of DEMO blanket systems of water cooled blanket and helium cooled blanket have been designed. They are consistent with the design for Steady State Tokamak Reactor (SSTR) [8] which has been proposed by JAERI as a reference DEMO fusion reactor. Both of them utilize packed small pebbles of breeder Li_2O or Li_2TiO_3 as a candidate and neutron multiplier (Be). The first wall model made of low activation material ferritic steels such as F82H and JLF1 have been fabricated and tested.

In the fabrication technology of the blanket/first wall, HIP method has been progressed. A HIP-bonded F82H first wall panel was successfully fabricated [9]. A high heat flux test of the first wall panel has been performed to examine the thermo-mechanical strength. Test conditions applied were the maximum heat flux of 2.7 MW/m^2 to accelerate the fatigue test up to 5000 cycles. The maximum temperature of the panel was ~ 460°C under this heat flux. Through this test campaign, cracks were not observed on the surface of the panel. Further, degradation in heat removal performance was not observed. Fatigue lifetime of the panel was found to be longer than the fatigue data obtained by a mechanical test of the material.

2.4 Plasma facing components

Plasma facing components in fusion reactors, such as ITER, should be actively cooled to withstand heat

Table 1 Major design parameter of plasma facing components.

	Divertor	First Wall
• Load Conditions		
No. of Cycles	3,000*	13,000
Steady-state Heat Flux, MW/m^2	5-10	0.5
Transient Heat Flux, MW/m^2	20	-
No./duration of Transient, #/s	300*/10	-
BPP Radiation Damage, dpa	0.3	1
Disruption Energy, MJ/m^2	100	1
VDE Energy, MJ/m^2	-	20-60
No./duration of Disruption, #/s	300*/0.1-3	500/0.025
No./duration of VDE, #/s	-	10/0.3-1
Energy/duration of Run-away Electrons, $MJ/m^2/s$	-	50-100/0.3
• Cooling Conditions		
Coolant	Light Water	Light Water
Inlet Temperature, °C	140	140
Inlet Pressure, MPa	4	4
• Materials		
Armor	CFC/W	Be
Cooling Tube / Heatsink	Cu alloy / Cu alloy	SS316L / Cu alloy
Structural Material	SS316L	SS316L

* with three component replacements

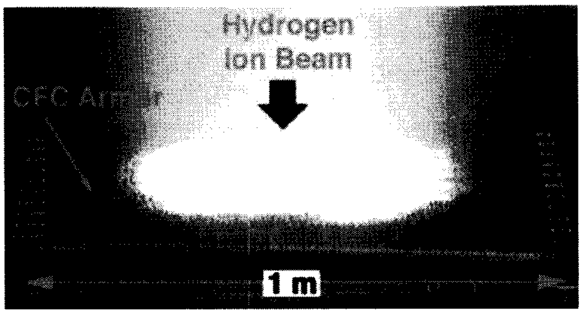

Fig. 5 Heating tests on the divertor mock-up.

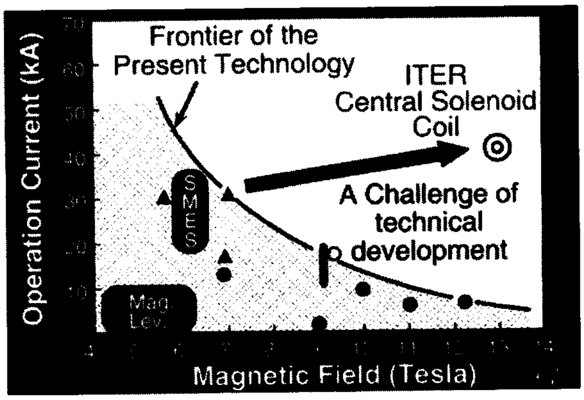

Fig. 6 Development of the CS coil.

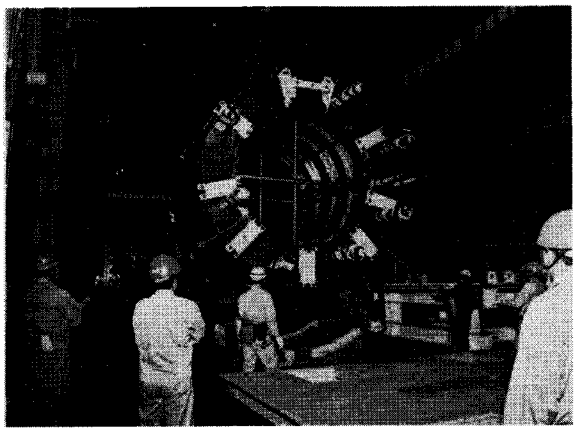

Fig. 7 The CS model coil for outer module.

loads for steady state operation. Major design parameters of the plasma facing components of ITER are summarized in Table 1 [10]. To satisfy these requirements, high thermal conductivity CFCs, high performance cooling tubes, and reduced radio-active bonding techniques have been developed [11]. The divertor full-scale mock-ups have been fabricated and tested in a high heat flux test facility as shown in Fig. 5. The mock-ups have successfully withstood both a heat load of 5 MW/m^2, 30 s for more than 3,000 cycles, and a heat load of 20 MW/m^2, 10 s for more than 1,000 cycles [12]. These heating conditions are relevant to the ITER steady state heat load condition, and the ITER transient heat load condition, respectively.

2.5 7Superconducting magnet

The superconducting magnet technology for the construction of the ITER requires challenging efforts for the new technology as shown in Fig. 6. For this purpose, the ITER Central Solenoid (CS) Model Coil [13] which is a 13 T, 1 T/s, 46 kA, 640 MJ, 110 ton Nb3Sn coil was designed and fabricated through difficult researches and development by an international cooperation.

Through this project, the following new technologies were developed; 1) high performance Nb3Sn strands [14] suitable for operation at 13 T, 2) technologies to fabricate and wind 46 kA huge conductor whose cross section of 50 mm × 50 mm with a high accuracy, 3) heat treatment of the winding without any cracking of conductor jacket made of Incoloy 908 [15], 4) conductor insulation after the heat treatment of Nb3Sn, 5) precise assembling technology of multi-layer windings, 6) reliable formation of the ground insulation by vacuum impregnation that can withstand 31 kV DC. Figure 7 shows the completed the CS Model Coil Outer Module whose outer diameter is 3.6 m that is almost identical to that of the CS coil of the compact ITER.

3. Conclusion

Present status of the key technologies for the steady state fusion plasma have been reported. Development of the steady state machine is strongly encouraged by these successful R&D results.

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