2. Fusion Engineering Research Project

Fusion Engineering Research Project (FERP) started in FY2010 at NIFS. Along with a conceptual design of the helical fusion reactor FFHR-d1, the project is conducting research on the technology of key components, such as the superconducting magnet, the blanket, the first wall, and the divertor. The project has 13 tasks and 44 sub-tasks with domestic and international collaborations.

Overview of the Helical Fusion Reactor Design

The conceptual design studies on the helical fusion reactor have been intensively conducted by FERP.

The latest design, FFHR-d1, is four times as large as LHD, and the major radius of the helical coils is 15.6 m. The toroidal magnetic field is 4.7 T. The operation point has been explored by a design integration code, HELIOSCOPE, using the “Direct Profile Extrapolation” (DPE) of the LHD plasma parameters. A self-consistently obtained operation secures the energy multiplication factor Q ~ 10. The confinement improvement in the ongoing deuterium plasma experiments, when confirmed, should lead to the self-ignition ($Q = \infty$). The startup simulation is described in the Column.

For the engineering design of FFHR-d1, a number of innovative ideas have been proposed from the following three purposes: (1) to overcome the difficulties related with the construction and maintenance of three-dimensionally complicated large structures, (2) to enhance the passive safety, and (3) to improve the plant efficiency. These are summarized in Fig. 1.

For the superconducting magnet, the High-Temperature Superconductor (HTS) is considered as a counter option to the cable-in-conduit conductors with Low-Temperature Superconducting (LTS) Nb_3Sn strands. One of the purposes for selecting HTS is to facilitate the three-dimensional continuous winding of the helical coils by connecting segmented conductors which are bent and helically twisted individually beforehand. The mechanical lap joint technique with low joint resistance has been developed at Tohoku University. The ReBCO tapes are simply stacked in a copper stabilizer and a thick stainless steel jacket, thus the conductor is called Stacked Tapes Assembled in Rigid Structure or STARS. A ~3 m-long short sample STARS conductor successfully achieved 100 kA current at the bias magnetic field of 5 T and a temperature of 20 K.

For the tritium breeding blanket design, we have chosen the liquid blanket option with molten salt from the viewpoint of passive safety. The present selection of molten salt is FLiNaBe, which has the melting point at 580 K. In order to increase the hydrogen solubility, an innovative idea was proposed to include metal powders, such as titanium. An increase of hydrogen solubility over five orders of magnitude has been confirmed in an experiment, which makes tritium permeation barrier less necessary for the coating on the walls of cooling pipes.

It is essential in fusion blankets to use structural materials whose radioactivity is low and decays swiftly after irradiation by neutrons. Fabrication and characterization of Oxide Dispersion Strengthened ferritic steel (ODS steel) is also performed. Both 9-Cr and 12-Cr ODS steels were fabricated and their characterization, including residual impurity effects and joining properties, was performed. Evaluation of the low-activation vanadium alloy, NIFS-HEAT2, strengthened by ODS method, has been carried out in collaboration with universities. This is for the purpose of (3), that is to increase the plant efficiency to secure the power conversion ratio to be >40% by raising the temperature to >1000 K.

For the helical divertor, the water-cooled tungsten monoblocks is considered. It is expected that a copper-alloy could be applied by placing diverter tiles at the backside of blankets where the incident neutron flux is sufficiently reduced. The peak divertor heat load on this divertor is expected to exceed >20 MW/m^2 because of the non-uniform divertor heat load profile. The maintenance scheme for the full-helical divertor is also a difficult issue. In order to solve these problems, a new concept of liquid metal limiter/divertor, REVOLVER-D, has been proposed. In this system, ten units of molten tin shower jets (falls) are installed on the inboard side of the torus to intersect the ergodic layer. It is proposed that the vertical flow of Tin jets could be stabilized using metal chains embedded in the jets. This system works as an ergodic limiter, and the conventional full-helical divertor becomes less necessary, although they could, or should, be still situated at the backside of the liquid divertor. Neutral particles are expected to be efficiently evacuated through the gaps between liquid metal showers.

Maintenance is one of the important and difficult issues to realize the helical fusion reactor. For the blanket, a toroidally segmented system, T-SHELL, was proposed divided in every 3 degrees in the toroidal angle. Another new idea is the cartridge-type blanket concept, CARDISTRY-B. The discussion about the maintenance both for the blanket and the divertor is ongoing.

The hydrogen isotopes, deuterium and tritium, will be utilized as a fuel in fusion power plants. Tritium is a radioactive isotope and therefore should be managed with safety. The project includes development of tritium handling and safety technologies, such as tritium decontamination and an advanced tritium removal system. Safety strategy for radiation facilities is also an important issue. Those collaboration experiments are performed using tritium facilities at universities.

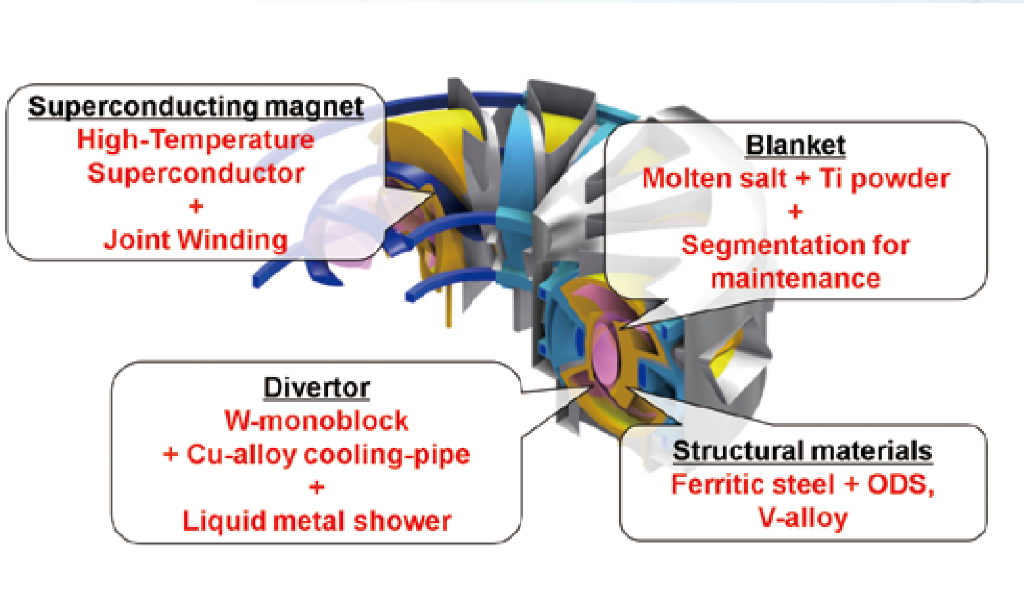


Fig. 1 Innovative ideas to be considered for the helical fusion reactor FFHR-d1.

Highlight

Fusion Startup Simulation of FFHR-d1

Regarding the helical reactor design, conceptual design activity of the LHD-type helical fusion reactor FFHR-d1 has been advanced. Regarding the core plasma design, the plasma operation regime of one of the design options of FFHR-d1 with high magnetic field (FFHR-d1B) was closely examined in view of MHD equilibrium/stability, neoclassical transport, density limit, helium impurity fraction, alpha energy loss and bootstrap current. Temporal evolution of the plasma density and temperature was calculated by the 1D simulation code based on the LHD experimental observations and the result was confirmed by the detailed physics analysis tools developed as the integrated physics analysis suite TASK3D-a by the Numerical Experimental Reactor Research Project. As a result, steady-state operation with a fusion gain of Q ~ 10 was confirmed within the physics parameter regime which has been already achieved in the LHD experiment: Mercier parameter $D_I < 0.3$ at the rotational transform of 1 surface and the energy loss by neoclassical transport is half the total absorbed power, meaning that the energy loss by anomalous transport is suppressed to the same as the loss by the neoclassical transport. Figure 1 shows the time evolution of the plasma and externally controlled parameters. The effects of the plasma and engineering design parameters on the operation regime were quantified. For example, the operation regime would drastically expand towards the region of high fusion gain by simultaneously relaxing the constraints of MHD stability and transport loss by means of the configuration optimization as shown in the POPCON plot. This study provides the design direction and the physics and engineering R&D issues of LHD-type helical reactors. The design requirements for the proper control of the plasma operation were identified, and the study also contributes to the overall plant system design. The developed calculation tool will guide the development of a real-time predictive simulation tool of the core plasma which will aid the plasma operation control of future fusion power plants.

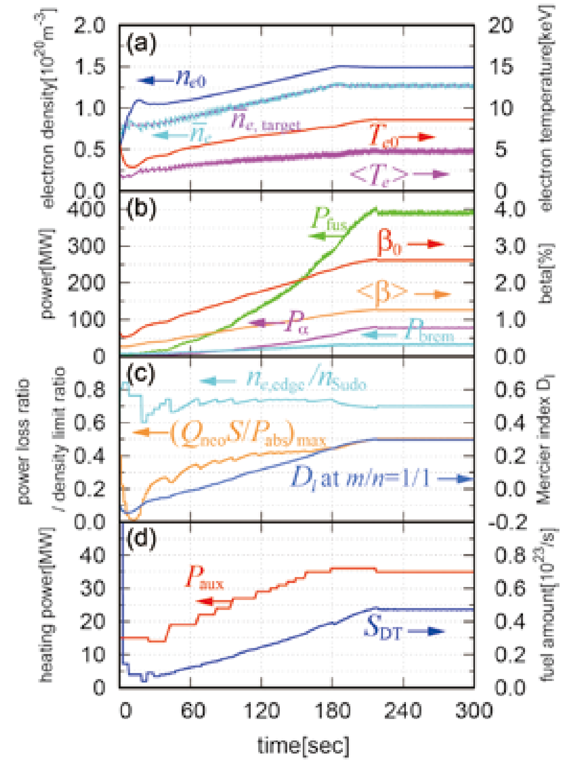


Fig. 1 Time evolution of (a) electron density and temperature, (b) fusion power, alpha power, bremsstrahlung loss and beta value, (c) ratio of neoclassical energy loss to the total absorbed power, ratio of the edge electron density to Sudo density limit and Mercier index and (d) external heating power and the injected fuel amount in the Q ~ 10 operation of FFHR-d1B.

R&D on Magnet

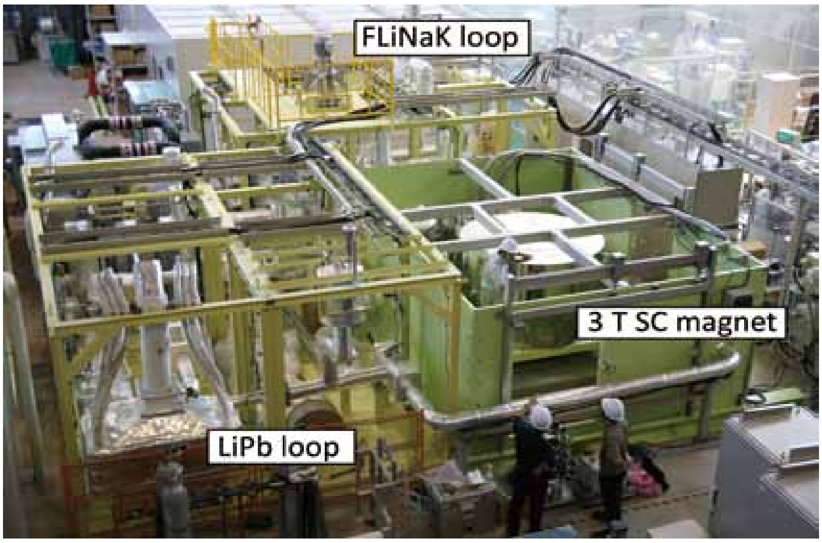
Research activities on applied superconductivity and cryogenic engineering focus on the large-scale superconductors of 100 kA-class current capacities at high magnetic fields of >13 T. Research is being conducted, in collaboration with universities and institutions, for developing indirectly-cooled low-temperature superconductor (LTS) and high-temperature superconductor (HTS). A new superconducting test facility with a 13 T magnetic field, a variable-temperature of 4.2 - 50 K, large bore of ϕ0.7 m, and 50 kA sample current has been installed (Fig. 2, left) together with a temperature controlled helium refrigerator facility. The excitation test of this 13-T coil is planned to be carried out within FY2017.

A highlight of the magnet task in FY2016 was the acceptance test of one of the Central Solenoid (CS) module coils of JT-60SA tokamak machine carried out in the Superconducting Magnet Laboratory of NIFS under the framework of collaboration between NIFS and National Institutes for Quantum and Radiological Science and Technology (QST). The test was done prior to the installation of the modules into the JT-60 SA complex which is under construction at QST. The CS module coil (weight 20 tons, diameter 2 m, Fig. 2, right) was cooled to ~4 K and was energized up to the conductor current of 10 kA. The basic properties of the superconducting coil were examined and it was successfully confirmed that the design specifications were satisfied.

Fig. 2 (Left) The 13-T solenoid at NIFS. (Right) Installation work of the JT60-SA CS module coil into the experimental facility at the NIFS Superconducting Magnet Laboratory.

R&D on Blankets

In order to develop the liquid breeder blanket for FFHR-d1, a large-scale forced-convection twin-loop facility of heat and hydrogen, “Oroshhi-2” (Fig. 3, up), was constructed equipped with a superconducting magnet to apply uniform perpendicular magnetic field of 3 T to the flow of either Flinak or LiPb. Using this facility, in FY2016, the MHD pressure drop in a flow of liquid LiPb was measured through a two-sectioned bending tube in collaboration with Kyoto University (Fig. 3, down). The relationship between the pressure drop and flow rate has been confirmed for the first time in the world. The experimental data will be used for validation of elaborate numerical simulation of the MHD effect so that a more accurate blanket design will become available. Coating and surface modification methods were also investigated for the mitigation of MHD pressure drop, tritium permeation reduction and corrosion protection for liquid breeder blankets.



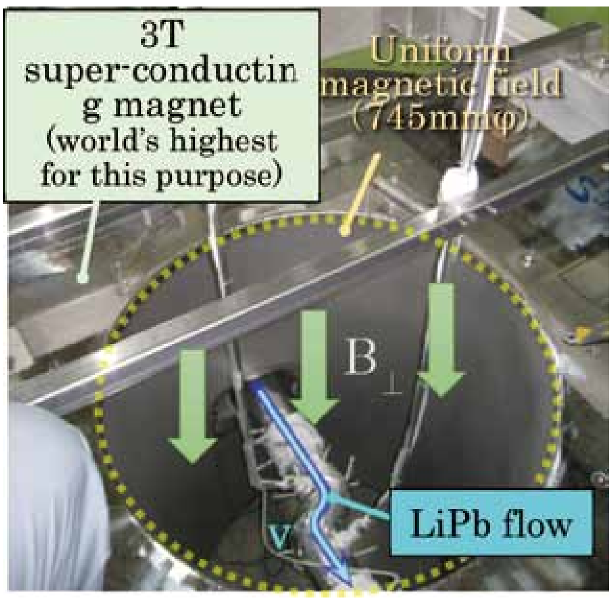


Fig. 3 (Up) The Oroshhi-2 twin loop facility. (Down) A two-sectioned bending tube installed in the superconducting magnet of Oroshhi-2 facility (collaboration with Kyoto University).

R&D on Divertor

Divertor heat flux in fusion reactors is considered to become even higher than 20 MW/m^2 in steady state. Important subjects in the research and development for coping with this high heat flux are the material selection, bonding technology between armor tiles and coolant pipes, and design studies of the 3D-shape helical divertor with precise neutronics analysis. Improvement of copper alloys is being examined in respect to high temperature mechanical properties and radiation resistance. Fabrication process using mechanical alloying and hot isostatic pressing (HIP) is being investigated. Characterization of welding and HIP joints are also carried out for ODS steels and ferritic steels. The ACT-2 electron beam facility (maximum power: 300 kW, Fig. 4, left) is used to apply >10 MW/m^2 of heat flux to test various samples, such as a tungsten block brazed with a copper alloy (Fig. 4, right). In order to solve the difficult problems of intense heat flux on divertors, another possibility is to use liquid metal, as is described above, and some basic studies are being carried out.

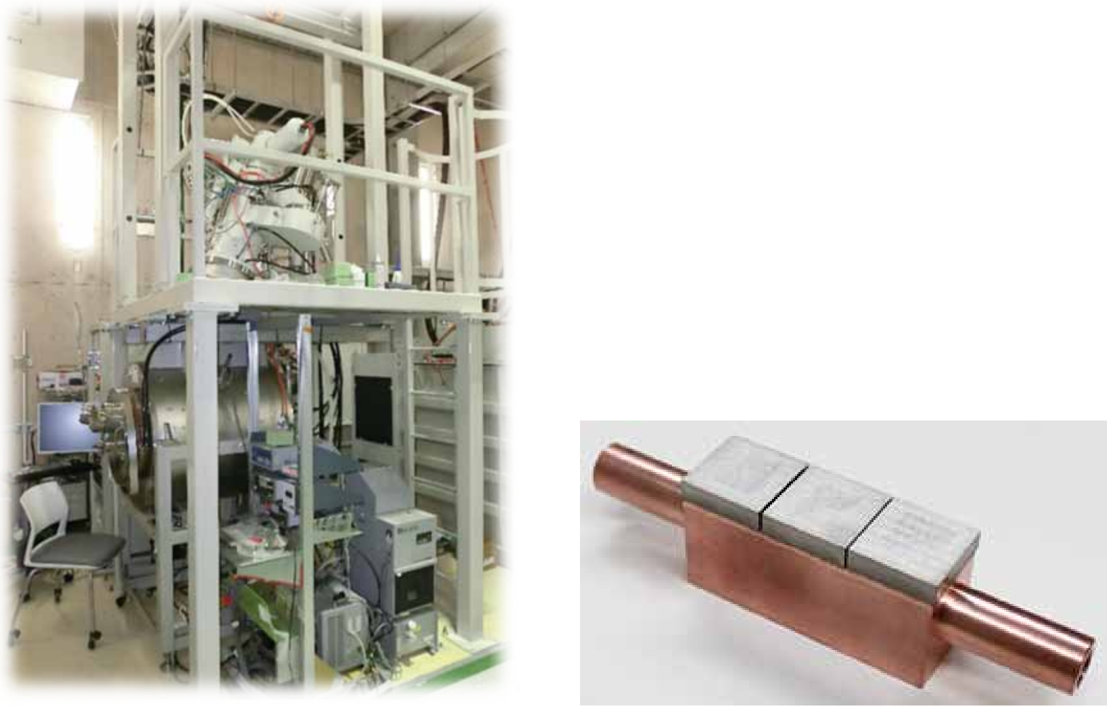


Fig. 4 (Left) The ACT-2 electron-beam facility with maximum power of 300 kW. (Right) Diveror block testing sample fabricated by employing an advanced bonding technique with brazing between tungsten and copper alloy.

Highlight

Dissimilar Bonding Techniques for Reactor Materials

Dissimilar-metals joint will be used for the first wall structure and cooling channel connection. The former requires large area and three-dimensional-shape bonding, while the latter needs robust welding to resist the coolant pressure. Two bonding processes, hot iso-static pressing (HIP) and electron-beam welding (EBW), are selected, because they are suitable for such structures, respectively. HIP and EBW joints were fabricated using 9Cr-ODS steel (Fe-9Cr-2W-0.1C-Ti-Y) and JLF-1 non-ODS steel (Fe-9Cr-2W-0.1C-V-Ta) (Fig. 1, up). Effect of the process parameters, such as HIP temperature and post-weld heat treatment conditions, were investigated.

Dissimilar-metals joints between NIFS-HEAT2 (NH2) and Hastelloy-X were fabricated by EBW with and without filler metal, such as pure Ni and pure Cu (Fig. 1, down). The joint without any filler fractured by weld cracking due to intermetallic formation accompanied by significant hardening in the weld metal. Pure Ni filler eliminated most of the hardening of the weld metal, however a several-micron-thick interlayer zone with still significant hardness was produced between NH2 base metal and the weld metal. In impact tests, the interlayer induced brittle fracture and lead to very low fracture energy. On the other hand, the hard and brittle interlayer disappeared, when pure Cu filler metal was used and Cu content in the weld metal was 95 % in mass percent. As a result, fracture energy was much improved by using the Cu filler metal. However, pure Cu is soft, reduced strength of the weld metal, and induces fracture in the weld metal under lower stress than the tensile strength of the base metal of NH2.

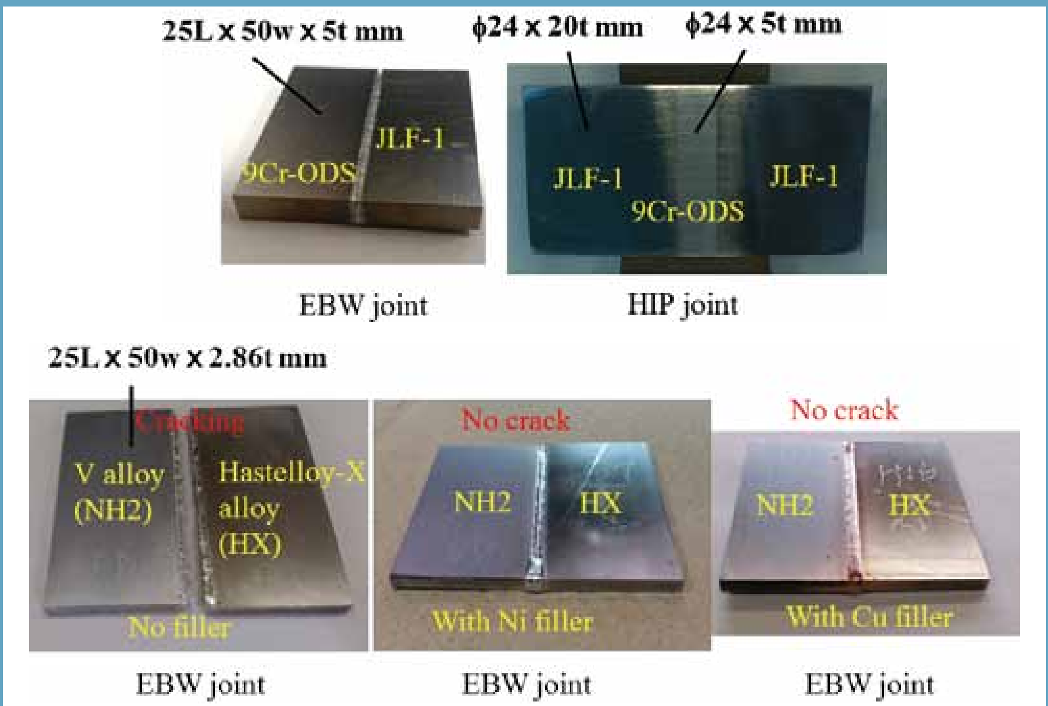


Fig. 1 (Up) Dissimilar bonding of RAFM ODS using EBW and HIP. (Down) Dissimilar bonding of Vanadium alloy NIFS-HEAT-2 using EBW with and without filler materials.

“LHD-Project Research Collaboration” with Universities

The aim of the LHD-Project Research Collaboration is to develop both technological and scientific foundations which are useful for the LHD project and universities. The characteristics of this collaboration program is that R&D’s are performed in universities and/or institutes outside NIFS. The advantage of this type of collaboration is that research collaborators can devote themselves to R&D’s more efficiently and enthusiastically by spending more time than by coming to NIFS. For the fusion engineering category, eighteen subjects were approved in FY 2016. Brief summaries are given for the following two subjects.

<u>Development of compact divertor plasma simulator for hot laboratory</u>

A compact divertor plasma simulator (Fig. 2) has been developed and installed in a radiation control area in International Research Center for Nuclear Materials Science of the Institute for Materials Research of Tohoku University. This device can generate steady-state deuterium plasma with electron density higher than 10^{18} m^{-3}. The sample stage equipped with an air-cooling system makes sample temperature almost constant during plasma exposure. Plasma-irradiated samples are transported to infrared heater for Thermal Desorption Spectroscopy (TDS) analysis without exposure to air.

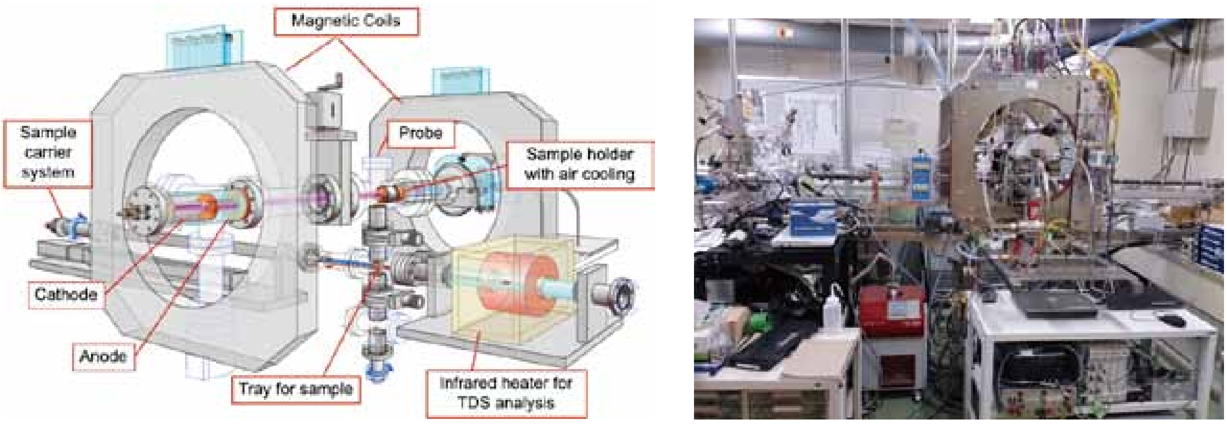


Fig. 2 Schematic drawing (left) and photo (right) of the compact divertor plasma simulator (CDPS).

<u>Development of helical winding using advanced superconductors to be used in high magnetic field</u>

For applying high-temperature superconducting (HTS) tapes for fusion magnets, the in-plane curvature of helical coils may cause a decrease in the critical current due to the edgewise bending strain. In order to establish the winding technique of helical coils without plastic deformation of HTS tapes, particularly with YBCO coated conductors, a prototype winding machine (Fig. 3) was constructed. Feasibility of geodesic winding, which can minimize the in-plane curvature variations has been investigated. As a result, the torsion control schemes based on the simultaneous four-spindle angle control system are visually confirmed.

(T. Muroga)

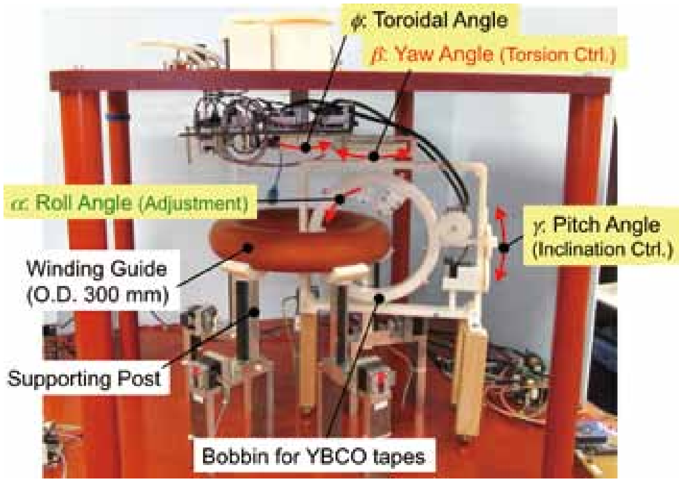


Fig. 3 Prototype helical winding machine using HTS tapes.