

NIFS工学プロジェクトでのヘリカル炉設計と工学基盤 Helical reactor design and engineering research bases in NIFS FER Project

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Fusion Engineering Research Project (FERP) has been launched from the FY2010 in NIFS. This project focuses on both the conceptual design of a steady-state fusion demonstration reactor and various engineering research and development, which are needed before entering into the engineering design activities for DEMO as shown in Fig.1. Therefore, this project consists of three research groups, (1) reactor system design, (2) superconducting magnets, and (3) in-vessel components, with the total 13 task and 44 sub-task groups as shown in Fig.2.

The LHD-type device does not need plasma current, and this excellent feature gives a great advantage for realizing a steady-state reactor. Therefore, along with a conceptual design of the helical DEMO reactor FFHR-d1 by integrating design bases established so far on the designs of the FFHR series for commercial power plants, the project is carrying out research on key components, such as the superconducting coil system, high performance blanket, first wall and divertors, and so on. As the center of fusion engineering research for universities, the project enhances domestic and international cooperation to advance reactor design studies as well as to expand basic research lying in interdisciplinary areas.

The large-scale superconducting magnet system requires high-performance superconductors of 100 kA-class current capacities. Research is being conducted for developing such an advanced conductor using metallic low-temperature superconducting materials like Nb_3Sn and Nb_3Al and/or high-temperature superconducting materials like YBCO. Actual environment testing is also carried out to estimate the characteristics of superconducting materials under conditions with cryogenic temperatures, intense magnetic field and neutron irradiation. Components in the magnet systems are subject to huge electromagnetic force. Research is ongoing so as to precisely evaluate the expected stress on component materials and to seek the optimum coil supporting structure. The engineering design of the winding and fabrication method for magnets is also in progress.

It is essential in fusion blankets to use structural materials whose radioactivity is low and decays swiftly after irradiation by neutrons. Vanadium alloys are one of the major candidate materials. A low-activation

vanadium alloy (NIFS-HEAT) was produced and its evaluation has been carried out in collaboration with universities. Development of various components using vanadium alloys is also progressing along with further improvement of the material. The experimental results confirmed that creep deformation is effectively suppressed by strengthening the material with high-density precipitation.

The blanket is a key component to shield neutrons, to convert fusion energy to thermal output, and to breed tritium fuels. In order to develop liquid breeder blankets using molten salts or liquid metals, experimental investigations are being carried out on the high temperature strength of materials and control of corrosion, which are important for a long-life blanket. One of the integration studies is constructing a forced convection type molten salt loop Orosh²i-1. In this work, feasibility study on material science including chemistry, fluid dynamics and thermo-mechanical engineering for the fusion blanket are integrated.

Divertor heat flux in a steady-state fusion demonstration reactor is considered to be maximum $10MW/m^2$ in a steady state. Here extremely high heat-tolerant divertor plates need to be developed. Three important subjects in the research and development are material selection, development in bonding technology between armor tiles and coolant systems, and design studies of the 3D-shape of the helical divertor with sufficient pumping ducts.

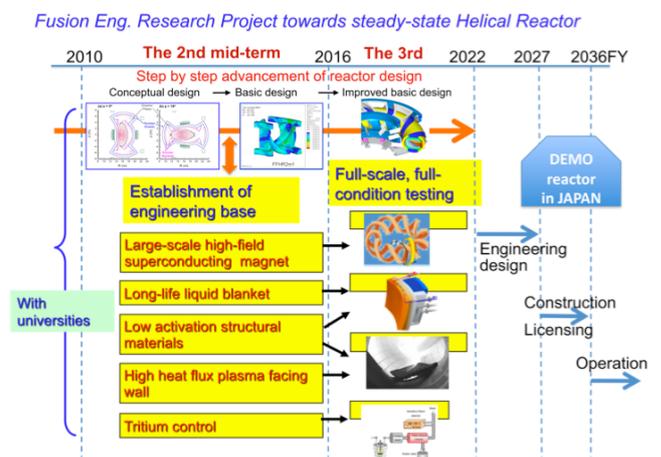


Fig.1 NIFS FERP activities towards DEMO reactor

The first wall of a magnetic fusion reactor is required to maintain high vacuum as well as to act as part of the blanket structure, facing the edge plasma, under 14MeV neutron irradiation at elevated temperatures. The choice of material for the first wall is thus of critical importance for the successful development of fusion energy.

Hydrogen isotopes such as deuterium and tritium will be utilized as a fuel in fusion 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. A sensitive detector is also under development for safety management of tritium. Those collaboration experiments are performed using tritium facilities of universities.

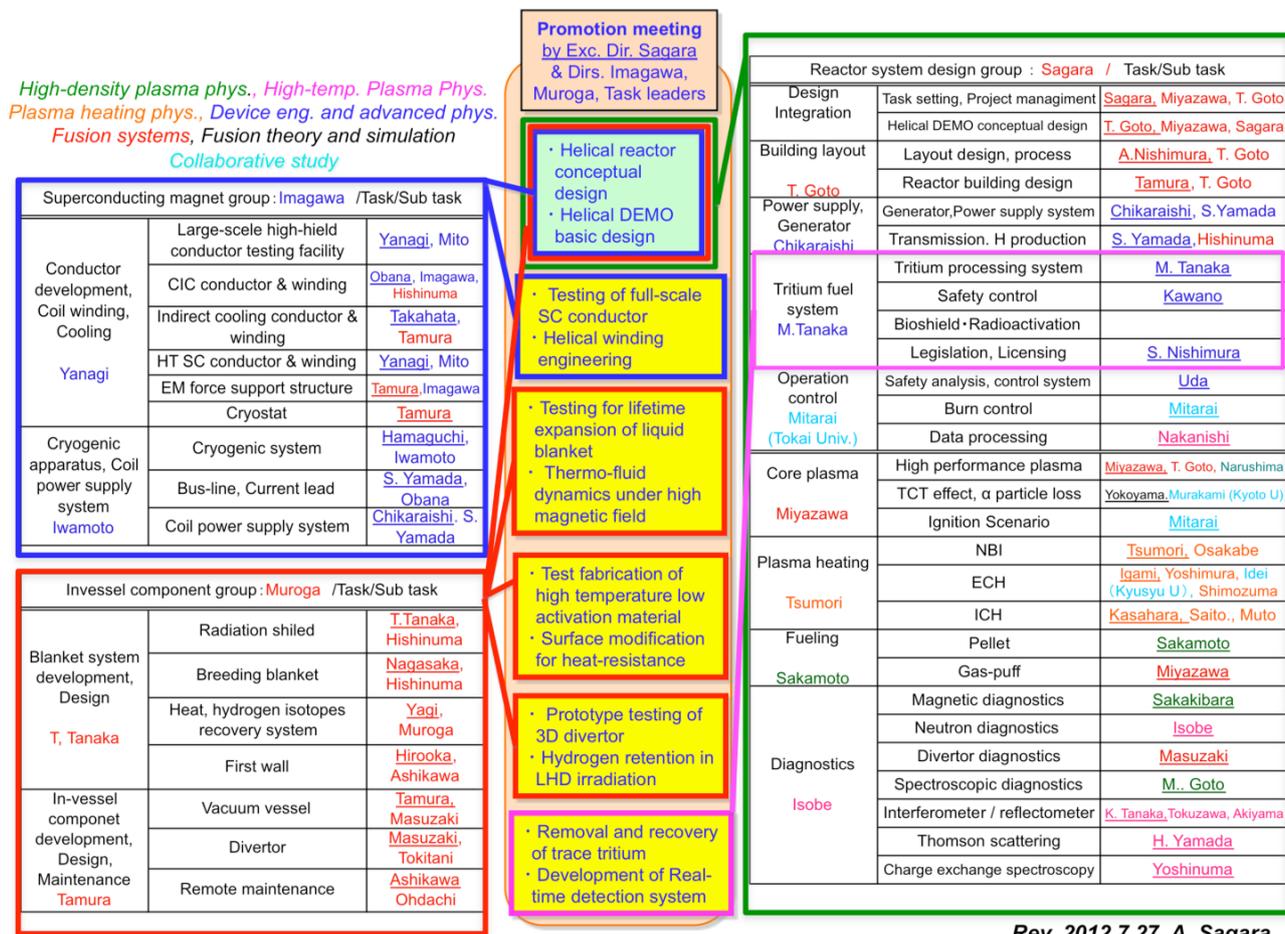
In the first three years, as the first round of design integration, primary design parameters of FFHR-d1 have been selected with a helical system code by introducing core plasma design with the Direct Profile Extrapolation (DPE) from LHD experimental data and by reducing blanket thickness with advanced shielding materials, resulting in reactor size optimization for blanket space and magnetic stored energy < 160GJ. The detailed 3-D design of in-vessel components, mechanical supporting structures, divertor pumping configurations and replacing scenarios are in progress as the second round.

Intensive R&D works are also in progress on

superconducting magnets, such as innovative superconducting current leads, basic study on High Temperature Superconductor (HTS), properties of superfluid HeII, remountable joint of YBCO conductor, HTS low porosity bulks, analysis of joints between Cable in Conduit (CIC) conductors, high efficiency cryocooler, analysis of a normal-zone propagation in the LHD helical coil, next generation power devices, properties of Cu addition MgB₂ wires, series compensated thyristor converters, inter-strand resistance in CIC conductor, temperature control with high-precision, transposed tape conductors, and so on.

As for in-vessel components, low activation vanadium alloys and ferritic steels were improved on high temperature mechanical properties and radiation resistance by compositional and microstructural control such as addition of nano-particle dispersion and reduction of grain sizes by mechanical alloying. Coating and surface modification methods were investigated for MHD pressure drop mitigation, tritium permeation reduction and corrosion protection for liquid breeder blankets.

Regarding tritium handling and safety technologies, many kinds of investigations have been successfully carried out as collaboration with many universities, research institutes and companies on hydrogen isotope separation and removal technology, tritium measurements, fueling in fusion reactor, safety in environment, and so on.



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Fig.2 NIFS FERF with the total 13 task and 44 sub-task groups.