2. Fusion Engineering Research Project

The Fusion Engineering Research Project (FERP) started in FY2010 at NIFS. Along with the conceptual design studies for the helical fusion reactor FFHR, the FERP has been developing technologies of key components, such as the superconducting magnet, blanket, and divertor. The research is also focused on materials used for blankets and divertors, the interaction between the plasma and the first wall including atomic processes, handling of tritium, plasma control, heating, and diagnostics. The FERP is composed of 13 tasks and 44 sub-tasks with domestic and international collaborations. Cooperation with the Large Helical Device Project, the Numerical Simulation Reactor Research Project, and the Task Force for Next Research Project have been promoted.

(T. Muroga)

Design Studies on Helical Fusion Reactor

Regarding the fusion reactor design study, the step-by-step strategy towards the helical fusion power plant FFHR-d1 has been proposed, as shown in Fig. 1. In the last fiscal year, three intermediate step devices have been considered before the construction of FFHR-d1: FFHR-c1 (experimental/prototype reactor for the demonstration of steady-state operation of the power plant system, FFHR-b2 (volumetric neutron source for the early utilization of fusion fast neutrons) and FFHR-a1 (non-nuclear system for the examination of improved magnetic configuration and advanced engineering concepts). In this fiscal year, a new design called FFHR-b3 which can operate more than 5 years with a net electric output of 100 MW at twice the size of LHD has been proposed based on the progress in physics and engineering research, such as the improvement of plasma performance by optimizing the helical coil shape and the development of compact and high-efficiency engineering components. This new design has led to the proposal of a new development strategy to demonstrate steady-state electric power generation at an early stage and to accelerate the realization of a commercial power plant.

Various innovative ideas are being proposed in the design of a helical fusion reactor. The divertor, which is directly exposed to the plasma in order to exhaust impurities in the plasma, has the problem of how to remove the ultra-intense heat from the plasma. To solve this problem, the pebble divertor concept, a method of dropping small ceramic pebbles near the plasma boundary, is being investigated (Fig. 2). Such a concept had been studied in the past, but at that time, damage of pebbles due to drop impact had become an issue. To presently address this issue, the cooperative system of a pebble divertor and a liquid metal blanket has been proposed. In this cooperative system, the liquid metal, which is circulated to cool the blanket module, is pooled at the bottom of the vacuum vessel. The pebbles drop into the liquid metal pool for alleviating the impact and for efficient use of the heat flowing into the divertor.

(T. Goto)

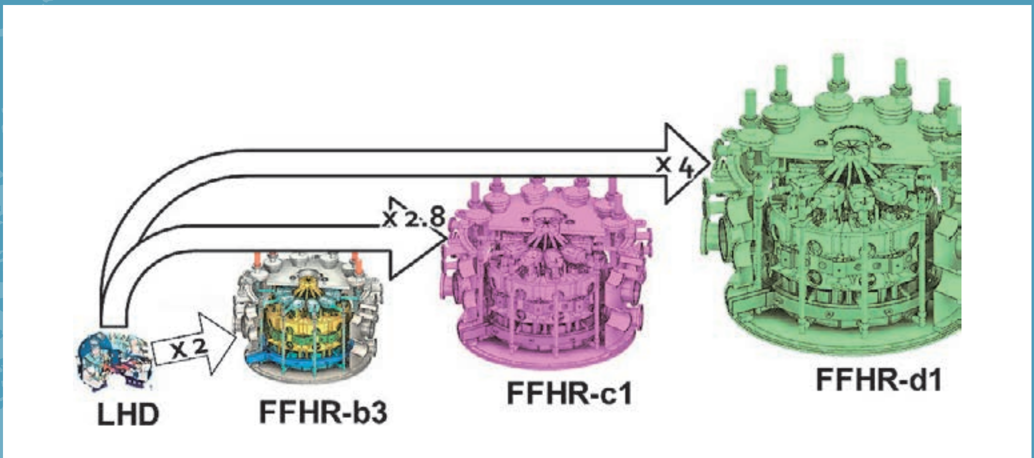


Fig. 1 Comparison of the size of helical fusion reactors. The latest design, FFHR-b3, is twice the size of the LHD and aims to generate net electricity of 100 MW.

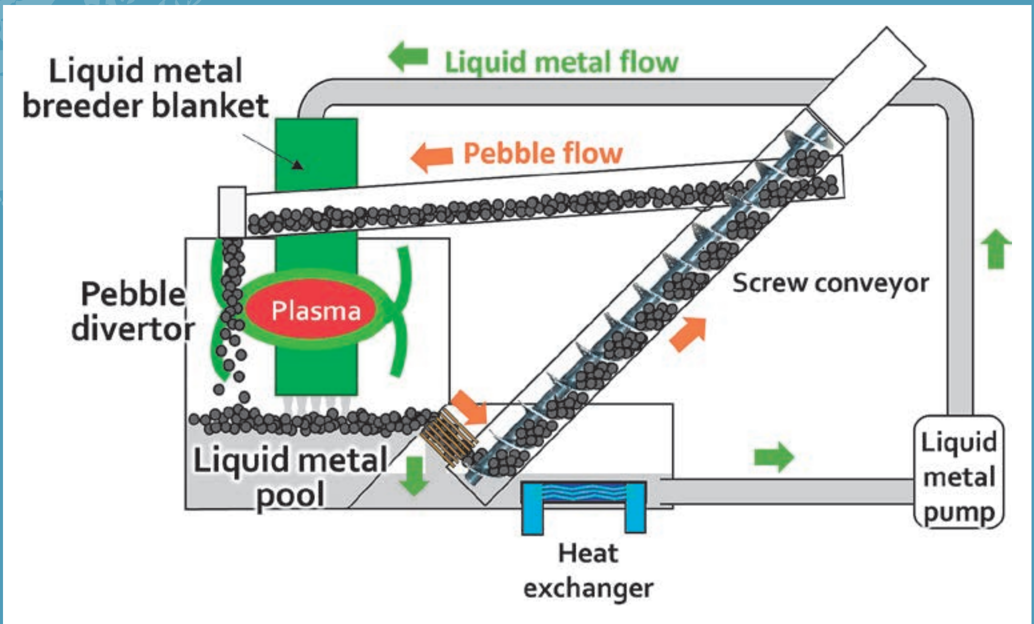


Fig. 2 2 Schematic view of the cooperative system of the pebble divertor and the liquid metal blanket.

Research and Development on Blanket

The FLiNaK/LiPb twin loop system Oroshhi-2 (Operational Recovery Of Separated Hydrogen and Heat Inquiry-2) is operated at NIFS as a collaboration platform for integrated experiments on the liquid blanket technologies. In FY2019, two new testing sections have been installed in Oroshhi-2: (i) a test chamber for demonstration of continuous and high efficiency hydrogen isotope recovery from LiPb by the vacuum sieve tray (VST) technique (Fig. 3 (a)) and (ii) a test section for evaluation of heat removal performance of a FLiNaK flow under an intense magnetic field (Fig. 3 (b)). In FY2020, experiments using these sections have been delayed significantly due to the severe influence of the COVID-19 pandemic, but preparation works have continued despite this situation. Regarding the test section of the VST technique, it was confirmed that dissolution of deuterium gas into a circulated LiPb flow and measurement of the deuterium concentration in the flow were performed steadily. The VST test section is ready for the demonstration of hydrogen isotope recovery. The temperature control of the FLiNaK heat removal test section was checked, and it is also ready for starting the experiments in FY2021.

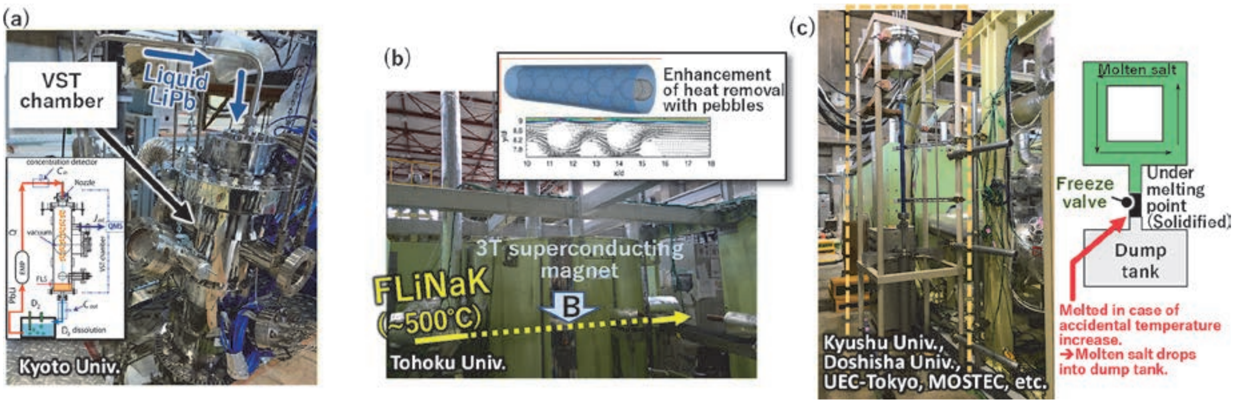


Fig. 3 Photos of test sections installed in Oroshhi-2. (a) LiPb vacuum sieve tray (VST) chamber, (b) FLiNaK heat removal test section penetrating the 3-Tesla superconducting magnet, and (c) FLiNaK freeze valve test section.

(T. Tanaka and Y. Hamaji)

Development of advanced structural materials for fusion reactor blanket

Low-activation vanadium (V) alloys are promising candidate structural materials for the first wall/blanket applications in advanced fusion reactor systems. One of the recent studies on the V-Cr-Ti system alloys is to further reduce the radioactive characteristic after use in fusion reactors. Effects of titanium (Ti) and chromium (Cr) concentrations on microstructure and tensile properties of high purity vanadium alloys, which contains less than 300 mass ppm interstitial impurities (e.g., carbon, nitrogen, oxygen), have been investigated. It has been found that Ti can be reduced from 4wt% to 1wt% for scavenger and precipitation. On the other hand, Ti reduction results in the strength degradation of V-Cr-Ti system alloy for both room temperature and high temperatures. Furthermore, it reveals that tensile strength gradually increases with increasing Cr concentration, indicating that a higher level of Cr can compensate for strength degradation by lowering Ti concentration. Comparing with the yield stress of V-4Cr-4Ti alloy, low-Ti and high-Cr candidates are V-6Cr-3Ti, V-8Cr-3Ti, V-10Cr-1Ti, V-12Cr-0.5Ti, and V-12Cr-1Ti alloys. The Ti and Cr concentrations will be further optimized according to irradiation damage resistance and ductile-to-brittle transition temperature investigations in the future.

(J. Shen)

Highlight

Corrosion of fusion structural materials in a molten salt with a hydrogen fluoride solution

NIFS has collaborated with universities and institutions worldwide to study lithium-containing molten salt as a candidate coolant for the fusion reactor blanket. These salts are well-known to be chemically stable and less affected by magnetic field. Hence, molten salts can be used for self-cooled liquid breeders. However, one of the difficulties with these salts is corrosion. Although these salts themselves hardly corrode with structural materials, the corrosive oxidants, such as water, hydrogen fluoride, and isotopes of these, are easily mixed or produced, corrode with structural materials. Nevertheless, the unknown was the quantitative relationship between the concentration of these corrosive oxidants and the corrosion mechanisms of the structural materials. Therefore, in this study, the effect of hydrogen fluoride (HF) in the molten fluoride salt (LiF-NaF-KF; FliNaK in short) on the corrosion mechanism of the reduced-activation ferritic steel (JLF-1 steel) is investigated.

Fig. 4 (a) shows the apparatus developed and used to measure the corrosion in the salt under the controlled HF concentration similar to the blanket condition. A long-time weight-change measurement and a short-time electrochemical measurement determine the corrosion rate, reaction, and form. Fig. 4 (b) shows the current-potential relationship of JLF-1 steel and pure iron in FLiNaK at 773 K. It is easily seen that the current density of pure iron, which is equivalent to the corrosion rate, increases with the increase in the HF concentration in FLiNaK. Furthermore, the HF concentration is proportional to the cathodic limited current density, which means that the rate-determining step of the corrosion is the HF diffusion occurring at the liquid-solid boundary. These tendencies are also matched for JLF-1 steel. In addition, surface and cross-sectional observations have gradually exhibited that HF prefers to attack the specific microstructure (lath boundary).

NIFS possesses the large circulation loop of a molten salt equipped with a 3-Tesla superconducting magnet (the “Oroshhi-2 loop”). Therefore, corrosion tests under flowing and magnetic field conditions are prospective.

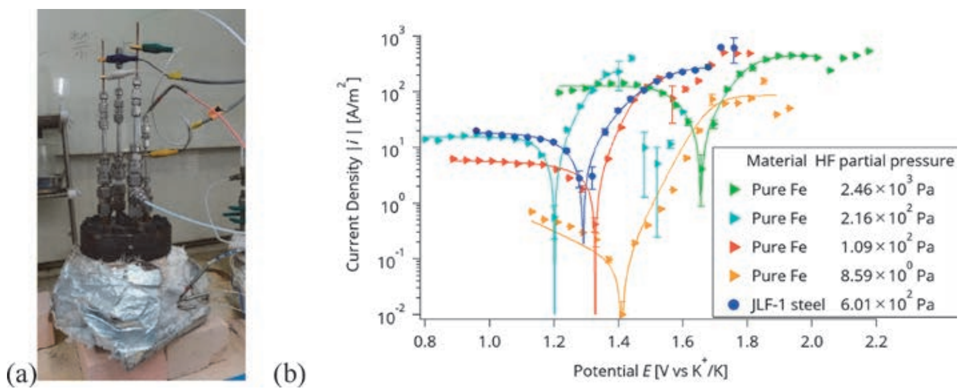


Fig. 4 (a) Apparatus for the corrosion tests. (b) Current-potential relationship of JLF-1 steel and pure iron in the molten salt (FLiNaK), depending on the HF concentration. Points are experimental results whereas lines are the fit results.

(G. Yamazaki and T. Nagasaka)

Research and Development on Divertor

Superior fabrication technique of the Advanced Multi-Step Brazing (AMSB) for fabricating the plasma facing component has been developed. The AMSB is based on the repetitive application of the Advanced Brazing Technique (ABT), in which the ABT was originally developed for joining tungsten (W) and oxide dispersion strengthened copper alloy (ODS-Cu; GlidCop®) in our previous work. The AMSB has large advantages for fabricating the high heat flux component with rectangular-shaped cooling flow path channel and V-shaped staggered rib structure because pre-processed cooling flow path channel can be tightly sealed via AMSB process with leak-tight condition. The new AMSB-type divertor heat removal component with rectangular-shaped cooling flow path channel and V-shaped staggered rib structure was developed, in which a pre-processed cooling flow path of the GlidCop® heat sink was sealed with stainless steel lid structure and then, W armor tiles were jointed on the GlidCop® heat sink. The component showed excellent heat removal capability under reactor-relevant conditions with a heat flux of ~30 MW/m^2. The AMSB type divertor component was inserted into the divertor strike point of the Large Helical Device (LHD) in the FY2020 plasma campaign, and has demonstrated the excellent heat removal capability and structural reliability by being exposed 1,180 plasma shots.

The AMSB is expected to be developed not only for a divertor heat removal component but also for a first-wall component with a W armor. For developing the first-wall component, very thin W armor (W sheet) should be jointed on the base metal such as stainless steel (SS) or ferritic/martensitic steel. The prototype first-wall component with W sheet armor and SS substrate was developed using the AMSB process. In this fabrication procedure, intermediate material of GlidCop® was inserted between the W sheet and SS. Fig. 5 shows a schematic drawing of the AMSB joining scheme and photograph of the fabricated AMSB first- wall component. The interface between GlidCop® and SS (GlidCop®/SS) and W and GlidCop® (W/GlidCop®) were jointed by the ABT with BNi-6 filler material. The very high joint area ratio was confirmed even in the 40 mm × 40 mm area by the ultra-sonic testing (UT). The AMSB technique is expected to be applied not only to the fusion engineering but also to other industrial fields.

Using a combined process of Mechanical Alloying (MA) and Hot Isostatic Pressing (HIP), new dispersion strengthened tungsten (DS-W) alloys containing TiO_2 particles on grain boundaries is being studied for the improvement of Plasma Facing Materials (PFM) on divertor components. The initial materials were mixed and mechanically alloyed in a planetary-type ball mill using tungsten carbide MA balls and pot. The mechanically alloyed powders were compressed by cold isostatic pressing (CIP) and then pre-sintered in hydrogen gas

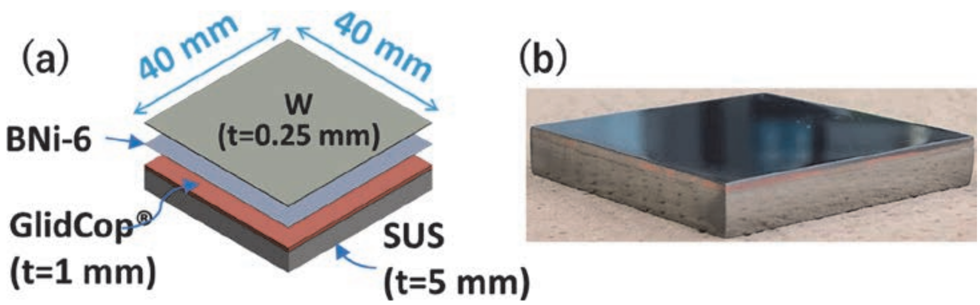


Fig. 5 (a) Fabrication scheme of the W/GlidCop®/SS first-wall component using the AMSB process. (b) Photograph of the prototype W/GlidCop®/SS first-wall component by AMSB.

atmosphere at high temperature (preliminary sintering). The pre-sintered W alloys were then sintered by HIP at 1750°C for 1.5 hours with a pressure of 186 MPa. For comparison, a pure tungsten supplied by the A.L.M.T. Corp. (hereafter named ITER-Pure W), was prepared. The HIPed DS-W and ITER-Pure W were annealed at 1600–1800°C for 1.5 hours in vacuum. The annealed tungsten materials were characterized by the four-point bending tests and the microstructure analysis. Fig. 6(a) shows EBSD (Electron Back-Scatter Diffraction) map images before and after annealing. The ITER-Pure W and the DS-W before annealing exhibited textures of fine grains by rolling and the equiaxed-fine grains of non-texture by HIP treatment, respectively. Grains of ITER-Pure W after annealing at 1600 and 1800°C coarsened drastically, indicating completion of the secondary recrystal-

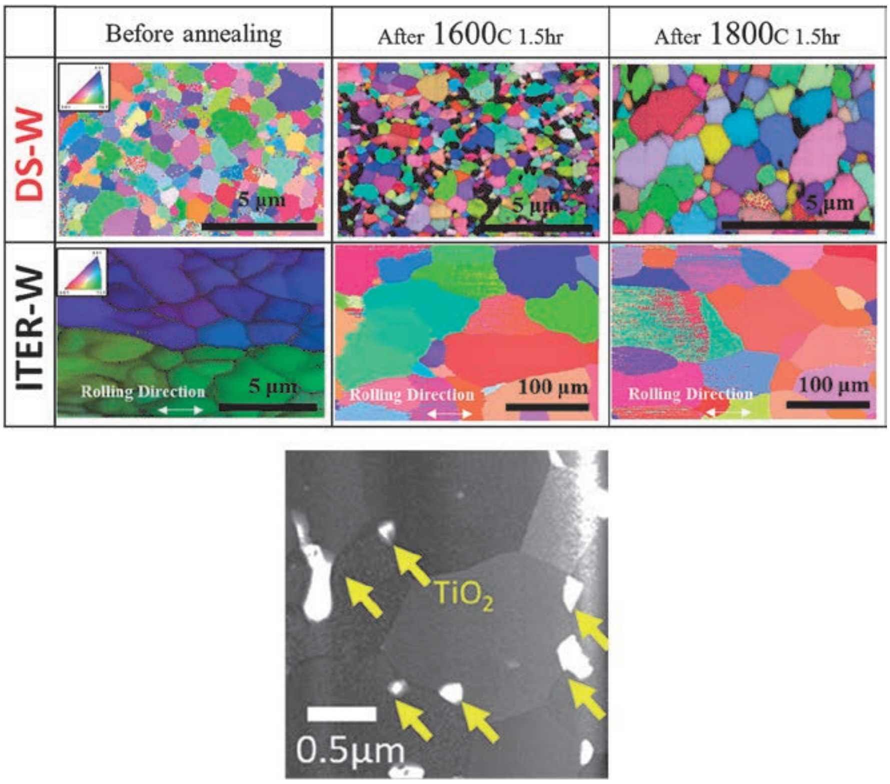


Fig. 6 (a) EBSD map images of the ITER-Pure W and DS-Ws before and after annealing. (b) Transmission Electron Microscope (TEM) image of nano-titanium oxide at grain boundaries after annealing at 1800°C.

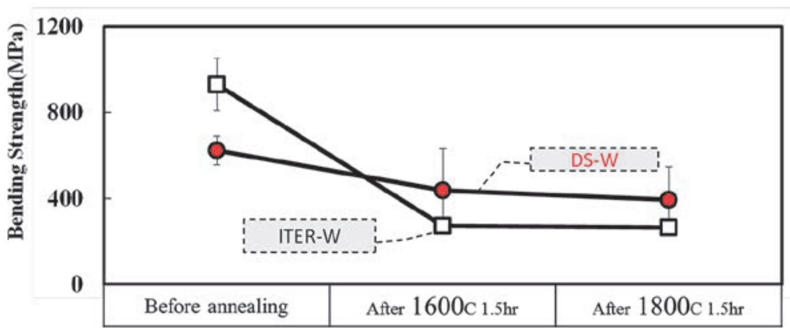
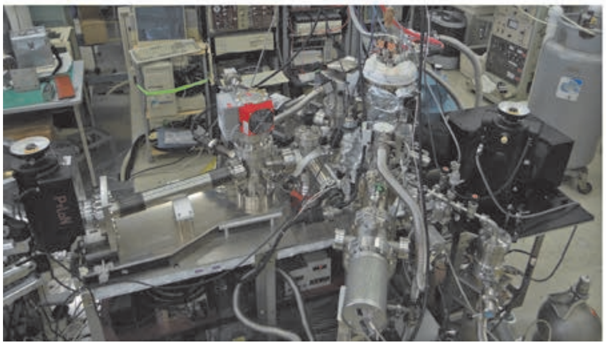


Fig. 7 Bending strength of the ITER-Pure W and DS-Ws before and after annealing.

lization. On the other hand, grains of DS-W after annealing at 1600 and 1800°C showed limited coarsening. As seen in Fig. 6(b), TiO_2 particles in DS-W were finely dispersed after annealing. The present result suggested that the dispersed particles of TiO_2 is stable up to 1800°C, maintaining the pinning effect on grain boundaries. Fig. 7 shows the fracture strength derived by four-point bending tests before and after annealing. Fracture strength of ITER-Pure W decreased drastically with grain coarsening after annealing. On the other hand, the fracture strength of DS-W after annealing decreased slightly, but was still higher than that of ITER-PURE W. This was attributed to the effect of fine dispersion of TiO_2 in DS-W.

We have been acquiring highly-charged tungsten (W) ion spectroscopy data for spectral identification of fusion plasmas using the electron beam ion trap, "CoBIT" (Fig. 8(a)). In FY2019, for the first time in the world, we successfully observed a very strong forbidden transition line called "electric octupole transition (E3)" of "W^{27+}4f-5s". In FY2020, the atomic number dependence of this E3 transition was successfully measured in Re (rhenium), Os (osmium), and Ir (iridium) ions (Fig. 8(b)).



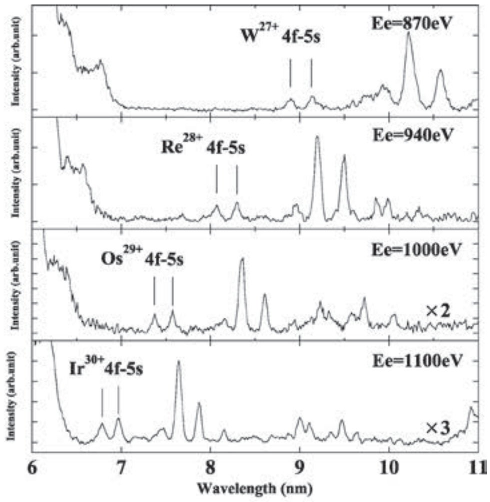


Fig. 8 (a) Photo of the electron beam ion trap "CoBIT" with installed spectrometers, (b) the E3 transition lines spectra of the highly-charged W, Re, Os, and Ir ions.

(M. Tokitani, H. Noto and H. Sakaue)

Research and Development on the Superconducting Magnet

In the recent designs of the helical fusion reactor series FFHR, the High-Temperature Superconducting (HTS) magnet is considered and a 100-kA-class HTS conductor has been developed. It is noted that large-current capacity HTS conductors are being developed also in the world to be applied to a variety of designs of fusion reactors. As a prior phase to fusion reactors, it is now being explored to apply HTS magnets to the next-generation helical experimental device. For this purpose, a relatively smaller conductor is required, and presently, the target is found at 6–18 kA current in the magnetic field of ~10 Tesla at temperature of 20 K. Three types of HTS conductors with different internal structures are being developed with all using REBCO tapes, as shown in Fig. 10. For these

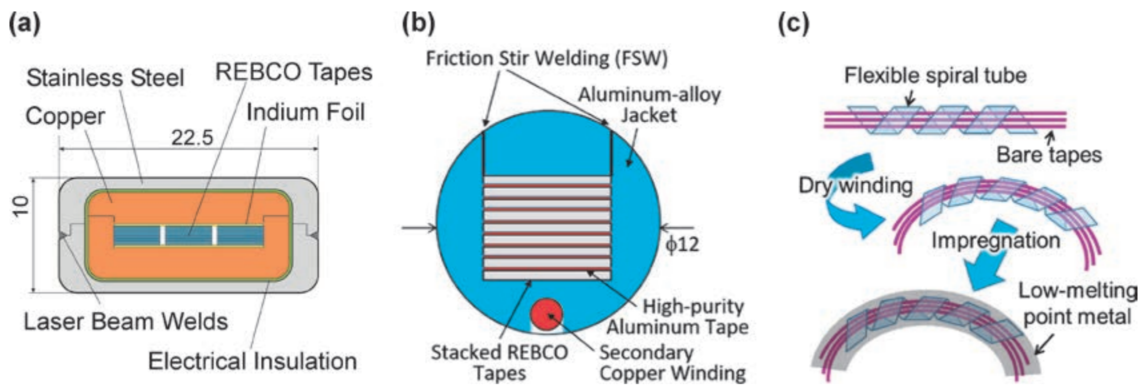


Fig. 10 Schematic drawings of the three types of large-current HTS conductors being developed to apply to the next-generation helical device: (a) STARS, (b) FAIR, and (c) WISE conductor.

Fig. 11 Experimental setup of the STARS conductor sample to be installed into the NIFS superconductor testing facility.

conductor types, 1–3 m-long samples have been fabricated and tested in liquid nitrogen (77 K) with no external magnetic field, and the fabrication method has been improved. Then, the conductors are being tested in high magnetic field (<8.5 Tesla) and low temperature (20–50 K) using the large-superconductor testing facility by installing 2-m-long conductor samples, such as shown in Fig. 11.

Development of mechanically reinforced Nb_3Sn multifilamentary wires has progressed for the Low-Temperature Superconducting (LTS) magnet option, as an extension of the ITER technology. We employed an internally reinforced matrix method using ternary bronze alloys (Cu-Sn-X), where X is a solute element such as Zn or In, as shown in Fig. 12(a). After the formation of the superconducting region of Nb_3Sn, the ternary alloy “hishimatrix” is transformed to a Cu-based solid solution such as (Cu, Zn) or (Cu, In), which contributes to the improvement of mechanical strength of the whole wire. When a Cu-Sn-In ternary alloy is used as the matrix material, the critical current (Ic) degradation under the compressive stress is restricted compared with the conventional bronze processed and CuNb-reinforced Nb_3Sn wire, as shown in Fig. 12(b).

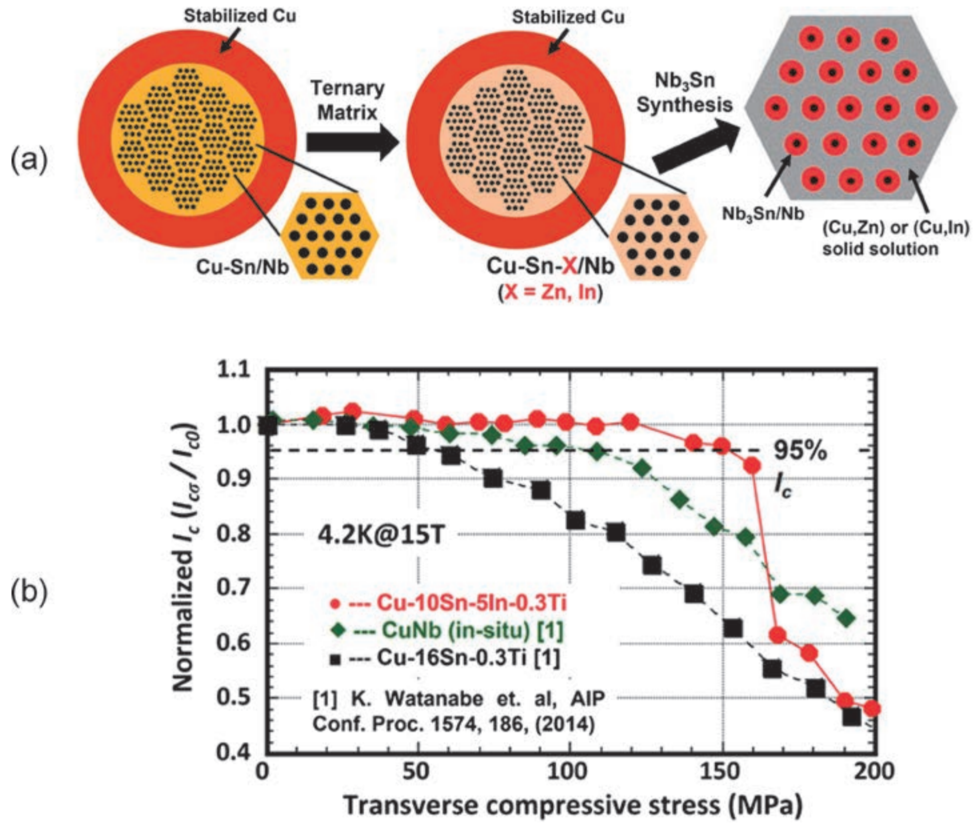


Fig. 12 Typical cross-sectional image of the internally reinforced Nb_3Sn multifilamentary wire using several ternary alloy matrices. (b) The comparison of I_c degradation under the compressive stress between several Nb_3Sn wires.

(N. Yanagi and Y. Hishinuma)

LHD-Project Research Collaboration

The LHD Project Research Collaboration program has been contributing to enhancing both the scientific and technological foundations for the research related to the LHD project as well as to the future helical fusion reactors. The feature of this collaboration program is that each research is performed at universities and/or institutions outside NIFS. For the fusion engineering, the following ten subjects were conducted in FY2020:

1. Engineering study on lithium isotope enrichment by ion exchange
2. Tritium behavior in a secondary cooling system in a fusion DEMO reactor
3. Field estimation for improvement of environmental tritium behavior model
4. Development of highly ductile tungsten composite systems
5. Fundamental engineering of tritium recovery process for liquid blanket of helical reactor
6. Development of new rapid-heating and quench processed Nb_3Al large-scaled cables for the helical winding due to the react and winding method
7. Evaluation of heat-transfer-enhanced channel under high magnetic field for liquid molten salt blanket development
8. Evaluation of multi hydrogen isotope transfer behavior on plasma driven permeation for plasma facing

materials

9. Studies on liquid hydrogen cooled HTC superconducting magnet
10. The analysis of biological effects elicited by organically bound tritium using life science techniques
11. Technological development of FeCrAl-ODS alloys coexisting with liquid metal cooling system of helical fusion reactor

From the above ten collaborative research items, the research 4 is briefly described below:

4. Development of highly ductile tungsten composite systems

A highly-deformed tungsten (W) foil shows excellent low-temperature ductility, even at room temperature. A pure W foil laminated composite (PWF-L) has been proposed for applying to the plasma-facing material of divertors. This is a composite material consisting of stacked pure W foils, and degradation of low-temperature ductility due to recrystallization is a concern for PWF-L. Potassium (K) doping is known as a method to suppress recrystallization, in addition to improving the low-temperature ductility. Therefore, a K-doped W foil laminated composite (KWF-L) was developed in the present study, which has a possibility to show improved mechanical properties. In FY2020, development of a fracture map to predict fracture behavior of KWF-L with various structures has been done by performing heat load tests of divertor target mockup in the ACT-2 (a 300 kW electron beam facility) at NIFS to clarify the performance of KWF-L in the actual fusion reactor environments. As shown in Fig. 13, heat conduction ability of the mockup was kept up to 20 MW/m^2 of steady-state heat load.

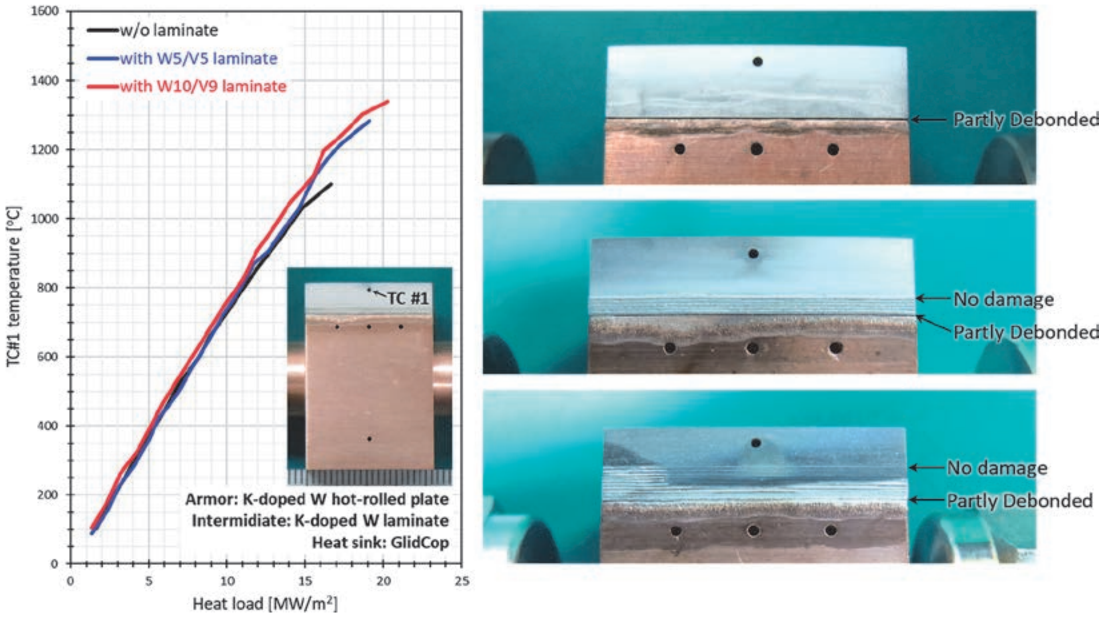


Fig. 13 Near-surface temperature during heat load tests of divertor target mockups using KWF-L (left) and their appearances after heat load tests (right).

(T. Nagasaka and S. Nogami (Tohoku University))