2. Fusion Engineering Research Project

The Fusion Engineering Research Project (FERP), started in FY2010, accomplished its twelve-year mission in FY2022. In this report, the research activities of the FERP in its final year are briefly summarized with its highlights.
Along with conceptual design studies for the helical fusion reactor, the FERP has been developing technologies for key components in fusion reactors, such as the superconducting magnet, blanket, and divertor, as illustrated in Fig. 1. The research also focuses 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 was composed of 13 tasks and 44 sub-tasks with domestic and international collaborations. There has also been cooperation with the Large Helical Device Project and the Numerical Simulation Reactor Research Project. The research activities of the FERP have been succeeded by a couple of "Units" in the new structure of NIFS.

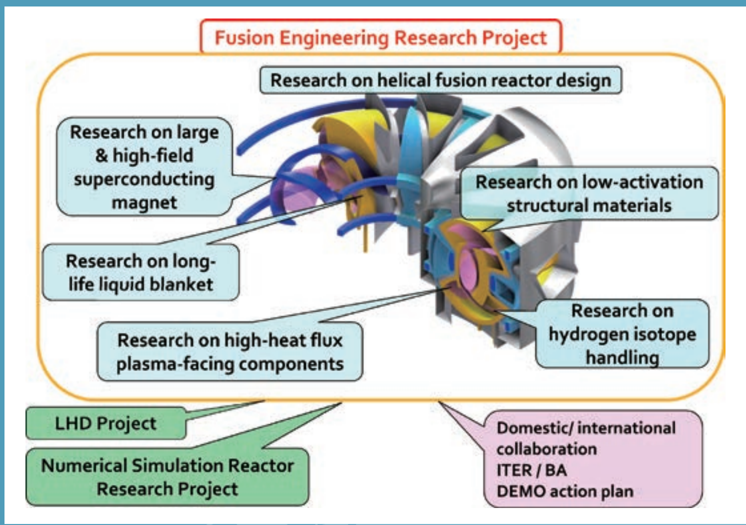


Fig. 1 Research areas of the Fusion Engineering Research Project with domestic and international collaborations.

(I. Murakami and N. Yanagi)

Design Studies on the Helical Fusion Reactor

Since the establishment of the Fusion Engineering Research Project (FERP) in FY2010, conceptual design studies for a steady-state helical fusion reactor have continued. Since FY2022 was the final year of the FERP, a summary of design studies has been conducted.
The conceptual design studies in the FERP have focused on the improvement of accuracy and feasibility of the design in cooperation with the Large Helical Device (LHD) Project and the Numerical Experimental Simulation Reactor Research Project, based on the achievements of previous design studies. While the conceptual design study has shown the feasibility of fusion reactor design utilizing the excellent features of the LHD-type heliotron device (steady-state operation capability, constructability, maintainability), it has also revealed various design issues. The two most significant ones identified are improvement of plasma performance (simultaneous achievement of MHD stability and good energy confinement) and ensuring sufficient blanket space.
To solve these problems, configuration optimization studies using the helical coil optimization code OPTHECS and engineering experiments on high-temperature superconductors and liquid blankets/divertors have been advanced. The R&D issues for improving the plant performance and public acceptance of fusion reactors have also been organized, leading to the systematization of fusion engineering as an academic field. The conceptual design study also led to collaborative research on three-dimensional neutron transport calculations, visualization of magnetic field lines, and an investigation of better magnetic field configurations using machine learning. It is expected that these research activities will be continued in the newly established "Units" as themes that lead to the solution of problems related to fusion reactors in general.
In the conceptual design of helical fusion reactors, structural analyses of the optimized helical magnet

system have been carried out. The basic geometry has been based on a similar expansion of the winding law of the helical coil (HC) as in the LHD. On the other hand, studies have been conducted to aim for better confinement performance, and optimization by changing the parameters of the winding law and geometry optimization by flexibly deforming the coil trajectory with spline curves. The effects of such changes in the geometry of the HC on the electromagnetic (EM) forces and the mechanical behavior of the coils were verified from the viewpoint of the integrity of the equipment. In the helical winding law, α is the HC pitch modulation parameter which determines the poloidal position of the coil, relative to the toroidal angle. The effect of the winding law optimization with α of 0.1, 0, and with the spline-based one were investigated. Although the changes in coil position due to these optimizations appear to be slight, they can result in relatively large changes in the generated magnetic field and electromagnetic force distribution. For the structural analysis, a major radius of 7.8 m and a plasma center field of 6.6 T were assumed. To compare stress distribution over a wide area, stress analysis was performed by applying the same support structure. The coils are supported by a torus shell and there are no openings. Table 1 and Fig. 2 show the results of the electromagnetic field and stress analysis. Even if the magnetomotive force of the HC is the same, the stress and displacement generated in the support structure are different, depending on the coil trajectory or the magnitude of the magnetomotive force of the vertical field coil.

Table 1 Coil specifications and electromagnetic properties

			LHD-type α=0.1	LHD-type α=0	Spline HC
Magnetomotive force (MA)	HC		25.74	25.74	25.97
	IV		15.03	15.37	19.46
	OV		−14.59	−16.91	−14.43
Magnetic stored energy (GJ)			85	89	93
Maximum magnetic field (T)			18.57	18.91	19.51
EM hoop force (MN/m)	HC	innermost	70.41	80.73	76.67
		outermost	53.73	44.36	50.10
	IV		77.36	75.59	113.94
	OV		22.65	28.61	22.82
Max. Mises stress (MPa)			506	549	553
Max. displacement (mm)			7.1	7.3	8.1

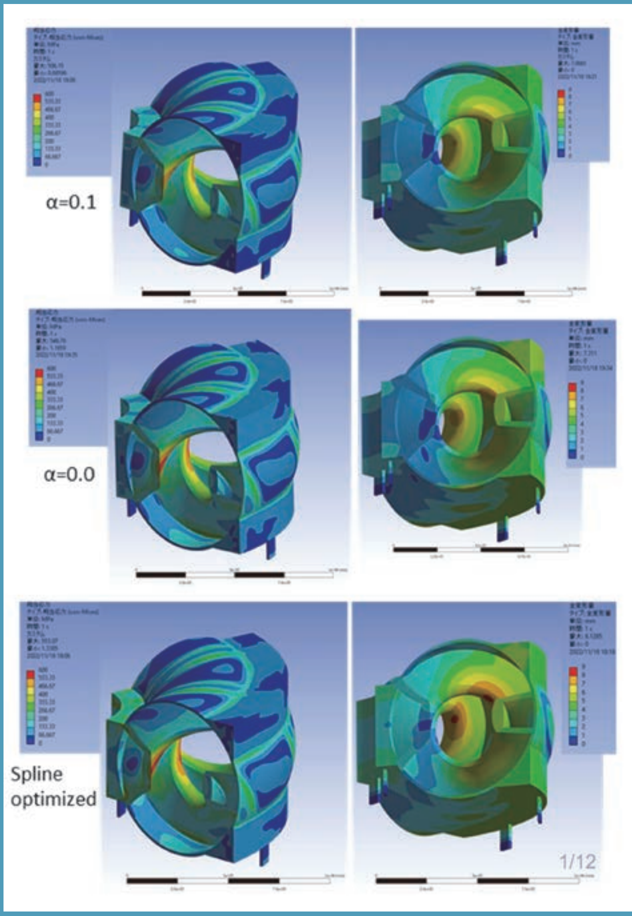


Fig. 2 Stress and displacement distributions of the magnet system of the helical fusion reactor with the helical coil configuration defined as (top) α = 0.1, (middle) α = 0.0, and (bottom) spline-optimized.

(T. Goto, H. Tamura and H. Yamaguchi)

Highlight

Research and Development on Divertor

A new type of divertor heat removal component with tungsten (W) armor and an oxide-dispersion-strengthened copper (GlidCop®) has been developed by original fabrication technology, so called Advanced Multi-Step Brazing (AMSB). The component has a rectangular-shaped cooling flow path channel with a V-shaped staggered-rib structure in the GlidCop® heat sink. This W/GlidCop® component showed an extremely high heat removal capability during a ~30 MW/m^2 steady-state heat loading condition in our previous work. In this work, to investigate the durability and plasma wall interaction (PWI) phenomena of the new component against simultaneous loading of high-heat and high-particle flux under a large-sized plasma confinement device, the component was installed in the divertor strike position of the Large Helical Device (LHD), as shown in Fig. 3, and exposed to neutral beam heated plasma discharges with 1180 shots (~8000 s) in total. After the exposure, several microstructural analyses were conducted by using a dual-beam type of focused-ion beam/scanning electron microscopy (FIB-SEM) device, nanoDUE'T® NB5000 (Hitachi High-Technologies Corp.) on the W armor plate. An extremely high heat removal response in the component was demonstrated through the quick increase and decrease of the temperature of the W plate, located on the divertor strike point during discharges. Though submillimeter-scale damages such as unipolar arc trails and microscale cracks were identified on the W surface, such a high heat removal response did not show any sign of degradation during the 1180 shots (~8000 s) of plasma discharges. If these submillimeter-scale damages are evolved beyond a millimeter scale, the heat removal performance will be degraded. The improved component, which has tolerance against repetitive heat loading, will be required for a reactor environment.

From a viewpoint of microstructural damage on the surface of the W armor, remarkable sputtering erosion and redeposition phenomena, due to the strong influx of the divertor plasma, was confirmed on the side surface wall of the W armor. This phenomenon could be caused by the synergistic effects of high flux and long-time particle loading. Such a high fluence could also be easily reached in future fusion devices. This work has demonstrated one of the critical erosion phenomena of W armor for future fusion devices.

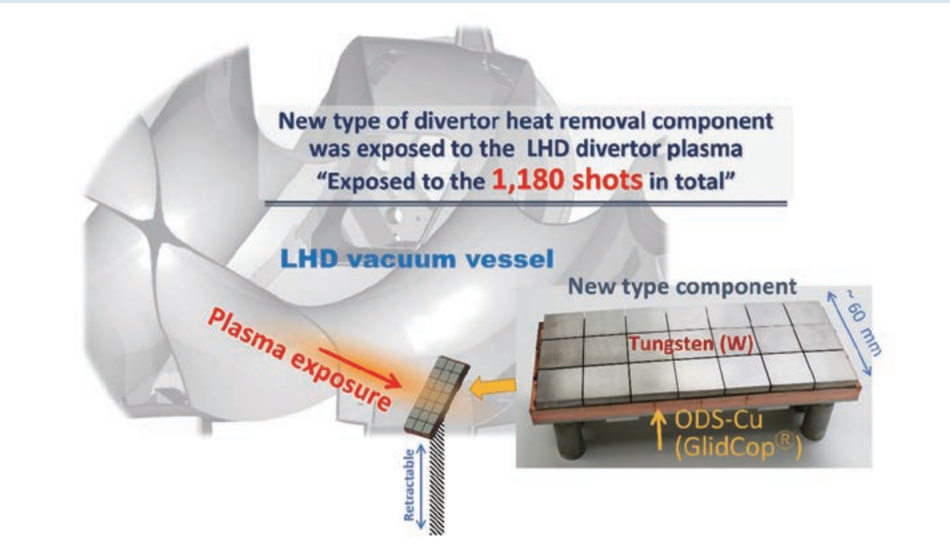


Fig. 3 Schematic figure for an exposure experiment of new type of divertor heat removal component to an LHD divertor plasma in the vicinity of the joint interfaces for W/Cu (b) and Cu/RAFM steel joint.

(M. Tokitani)

Development of oxide dispersion strengthened tungsten (ODS-W) including titanium oxide

For improvement of plasma facing tungsten on a fusion divertor, we have developed a new oxide dispersion strengthened tungsten (ODS-W), including titanium oxide as strengthening particles (Fig. 4), fabricated by mechanical alloying (MA)-hot isostatic pressing (HIP), which can inhibit the decrease of a mechanical property at high temperature, even after recrystallization occurs. In terms of its subsequent evaluation, it is therefore important to assess the material's soundness, whether it can withstand intense irradiation and high heat loads. We have researched the irradiation resistance of new oxide dispersion strengthened tungsten with a grain size of 1–2 μm and dispersed small titanium or titanium oxide nanoparticles. For the evaluation of irradiation resistance, tungsten ion irradiation experiments were carried out at 500 °C to about 0.66 dpa at the damage peak by 2.9 MeV W^{2+} ions from the Tandetron accelerator at the High Fluence Irradiation Facility, the University of Tokyo (HIT). A nano-hardness measurement was carried out using a Shimadzu nano-indenter with a Berkovich-type indenter. A nano-hardness of ODS-W was 6.4±1.2 GPa and 6.4±1.4 GPa before and after irradiation, respectively, and no irradiation hardening was observed. On the ODS-W, no void formation was observed after ion irradiation, as shown in Fig. 5. Therefore, according to the above results of a TEM observation and the hardness tests, this material is considered to have a high performance against irradiation.

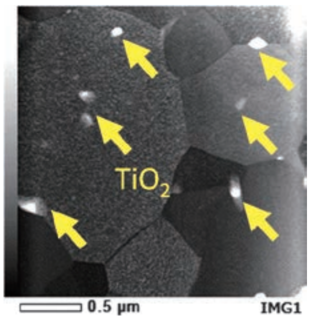


Fig. 4 Transmission Electron Microscope (TEM) image of nano-titanium oxide at grain boundaries after annealing at 1800 °C.

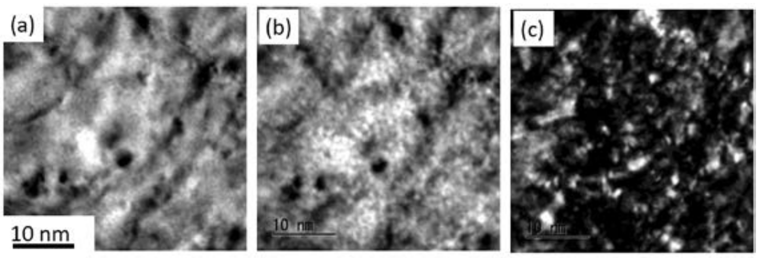


Fig. 5 (a) Bright-field image (in focus), (b) bright-field image (out of focus), (C) weak-beam dark field image in the irradiated W-TiC composite at 500 °C to 0.66 dpa. Copied from [1].

[1] E. Wakai, H. Noto *et al.*, "Titanium/titanium oxide particle dispersed W-TiC composites for high irradiation applications", Research & Development in Material Science **16**, 1879–1885 (2022).

Tungsten is a strong candidate material for plasma-facing walls such as divertors in next-generation fusion devices. However, although the highly charged ion spectrum of tungsten is very important for plasma diagnostics, the identification of emission lines has not yet been established, due to the complexity of the excitation process. Therefore, we have been conducting a spectroscopic study of the highly charged tungsten ions using an Electron Beam Ion Trap (EBIT), in order to obtain atomic data on highly charged tungsten ions.

In highly-charged tungsten ion spectroscopy to date, quasi-continuous spectra (Unresolved Transition Array, UTA), which are formed by a large number of emission lines peculiar to highly charged ions, have been measured. These UTAs appear not only in tungsten but also in the emission lines of highly charged heavy ions and are widely observed in extreme ultraviolet light sources, fusion plasmas, and astronomical plasmas. However, there is still no quantitative diagnostic method for UTA spectra. Our research group has been focusing on UTA and analyzing its structure. We obtained UTA emission spectra from tungsten pellet injection at the LHD and compared them with the monoenergetic spectra at EBIT. The upper panel of Fig. 6 shows the spectrum at tungsten pellet injection. UTA appears at around 5 nm. The lower panel shows the electron energy dependence of the highly charged tungsten ion spectrum in the Tokyo-EBIT. Third-order light from diffraction grating (the wavelength is tripled) was used to obtain a high-resolution spectrum. In the future, we will analyze the spectral data using statistical theory to develop a new quantitative diagnostic method.

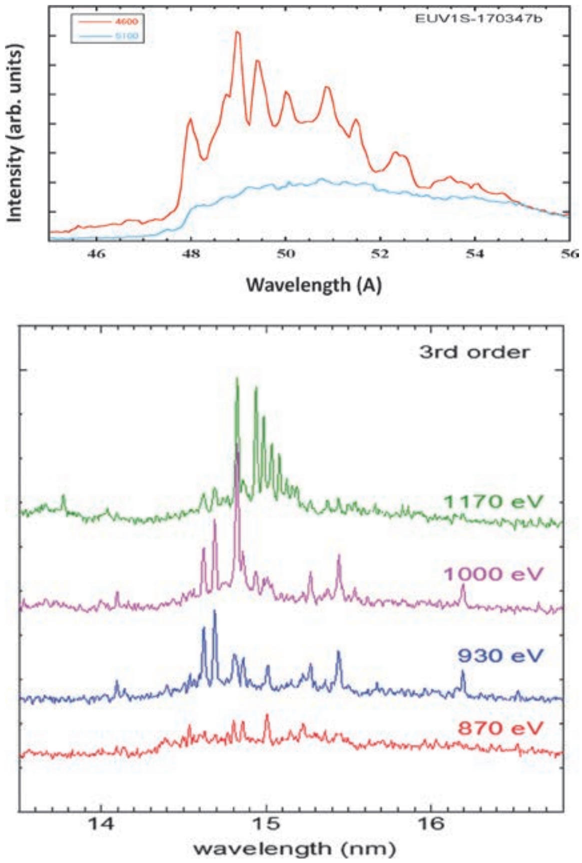
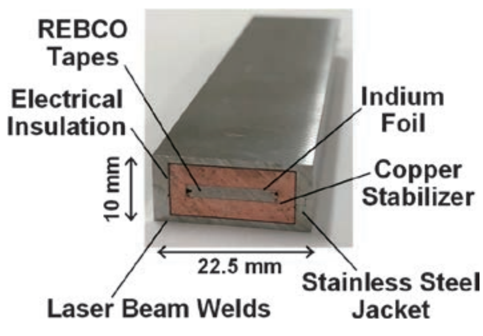


Fig. 6 (Upper) W pellet injection spectrum at LHD. (Lower) Energy dependence of highly charged W ion spectra in Tokyo-EBIT.

(H. Noto and H. Sakaue)

Research and Development on Superconducting Magnet

The NIFS-FERP has been developing three kinds of High-Temperature Superconducting (HTS) large-current conductors to apply to the next-generation fusion experimental devices, namely, STARS, FAIR, and WISE conductors. Here, the progress of the STARS conductor is briefly reported. The STARS conductor is characterized by its high current density (current amplitude divided by the cross-sectional area of the conductor). The target current density is 80 A/mm^2, which is about twice that of an LTS conductor of the same size. The STARS conductor consists of 15 layers of REBCO tapes, encased in a stabilized copper jacket and strengthened by an outer stainless steel jacket (Fig. 7(a)). A coil-shaped sample having three turns of a diameter of 600 mm and a 6-m-long conductor (Fig. 7(b)) was fabricated by Metal Technology Co. Ltd. using the state-of-the-art EuBCO tapes manufactured by Fujikura Ltd. It was tested in the NIFS large-diameter high-field conductor testing facility and stably energized up to its rated current of 18 kA at a temperature of 20 K (–253 °C) and magnetic field strength of 8 T. This means that the target current density of 80 A/mm^2 was successfully achieved. In addition, it was confirmed that the coil was energized stably, even when the current was raised and lowered at a high speed of 1 kA/s and repeated a total of more than 200 times with no change, indicating that the STARS conductor is stable and robust.



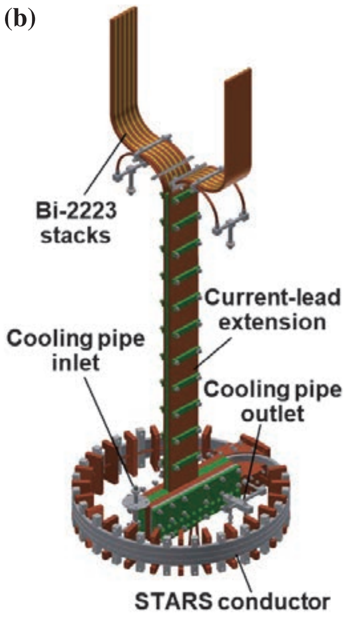


Fig. 7 (a) Photograph of a mock-up sample of the 20-kA-class STARS conductor with descriptions of internal layout, having a stack of 15 REBCO tapes (width: 12 mm) and internal electrical insulation between the copper stabilizer casing and stainless-steel jacket. (b) A three-dimensional drawing of the 6-m solenoid-coil sample without showing the stainless-steel supporting ring.

Magnesium diboride (MgB_2) wire is a promising alternative material to NbTi wire under a neutron irradiation environment such as fusion or accelerator applications. This is because MgB_2 has higher critical temperature (39 K) properties, lower cost, and low activation compared with NbTi wire. These factors will contribute to realizing a liquid hydrogen-cooled low-activation superconducting cable for future fusion magnets. We contemplated reconsidering the raw materials of in-situ processed MgB_2 wire in order to improve the low activation property. In the case of the in-situ process, generally, the MgB_2 phase is formed by the thermal diffusion reaction between magnesium (Mg) and amorphous boron (B). In particular, B material has two kinds of stable isotopes, i.e., ^{10}B

and ^{11}B, and the ^{11}B isotope is well known to have no nuclear transmutation by neutron irradiation. We thought that use of ^{11}B isotope as the raw B material in the in-situ processed MgB_2 wire could suppress the degradation of critical current density characteristics by nuclear transmutation, based on neutron irradiation.

Several in-situ processed MgB_2 wires using different ^{11}B isotope powders (crystallized or amorphous and coarse or fine) were fabricated, and their microstructures and critical current density (J_c) characteristics under a magnetic field were evaluated. A typical image obtained by the High-Angle Annular Dark Field Scanning Transmission Electron Microscope (HAADF-STEM) and the electron diffraction pattern after heat treatment are shown in Fig. 8(a). Formation of rectangular plate-like finer crystals was confirmed. The electron diffraction patterns indicate that plate-like crystals are an MgB_2 ($Mg^{11}B_2$) phase originated by the ^{11}B isotope. This tendency was also confirmed in in-situ wire samples using the other ^{11}B isotopic powder. The J_c-B performances of in-situ processed MgB_2 wires using a different ^{11}B isotope powder are shown in Fig. 8(b). In terms of the J_c-B properties, the amorphous 11B powder was more suitable than the crystallized ^{11}B powder as the B raw material. Although the present J_c-B property of $Mg^{11}B_2$ wires is lower than those of NbTi wires, we believe that the J_c-B properties will be improved by optimizing various conditions, such as heat treatment and in-situ nominal composition in the future.

(a) **(b)**

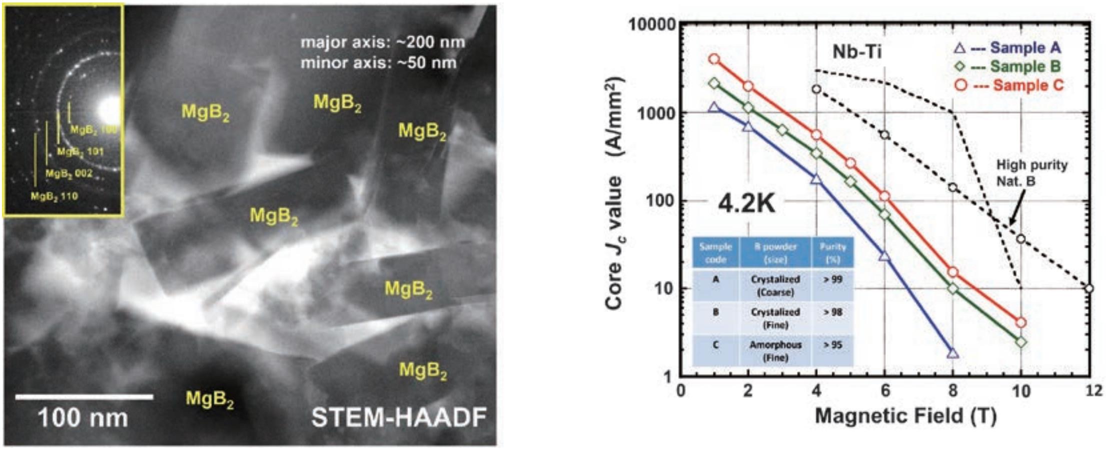


Fig. 8 (a) HAADF-STEM image and electron diffraction pattern after heat treatment and (b) J_c-B performances of in-situ processed MgB_2 wires using different ^{11}B isotope powder.

(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 of research related to the LHD project as well as to the future helical fusion reactors. A feature of this collaboration program is that all research is performed at universities and/or institutions outside NIFS. For the fusion engineering area, the following seven subjects were conducted in FY2022:

1. Evaluation of multi hydrogen isotope transfer behavior on plasma driven permeation for plasma facing materials
2. Studies on liquid hydrogen cooled HTC superconducting magnet
3. The analysis of biological effects elicited by organically bound tritium using life science techniques

4. Technological development of FeCrAl-ODS alloys coexisting with liquid metal cooling system of helical fusion reactor
5. Development of REBCO high Tc coated conductor with conductive micro-path
6. Fabrication technology development toward practical use of functional coating for helical reactor liquid blanket
7. Improvement of environmental tritium transfer model based on the observation near tritium released facilities

From the above seven collaborative research items, subject 5 is briefly reported below:

5. Development of REBCO high Tc coated conductor with conductive micro-path

In developing High-Temperature Superconducting (HTS) magnets for fusion and other applications, a drawback of REBCO tapes is that the buffer layer prevents current sharing between tapes, causing reduced conductor stability. We propose conductive micro-paths to improve conductor stability where the current is shared between the REBCO tapes. In this report, we fabricated the conductive micro-paths in REBCO tapes to investigate the current sharing in the tapes. Blind holes were made on REBCO tapes by Fujikura and SuperOx Japan, as non-conductive micro-paths, by using Nd:YAG lasers. Additional Ag films were deposited on the tapes by a sputtering method to make the micro-paths conductive. We observed the microscopic structure of the blind holes and conductive micro-paths by SEM microscopy. As a result, the blind holes reached the substrates of the REBCO tapes and the holes were filled by the deposited Ag films. Two and three REBCO tapes, one with degradation intentionally introduced, were prepared and stacked to investigate current sharing between them. Partial voltage in the tapes was measured with some voltage taps by sweeping the current at a temperature of 77 K. As a result, the current was shared between the REBCO tapes and successfully bypassed the damaged part. This phenomenon was confirmed with good reproducibility regardless of the sample type.

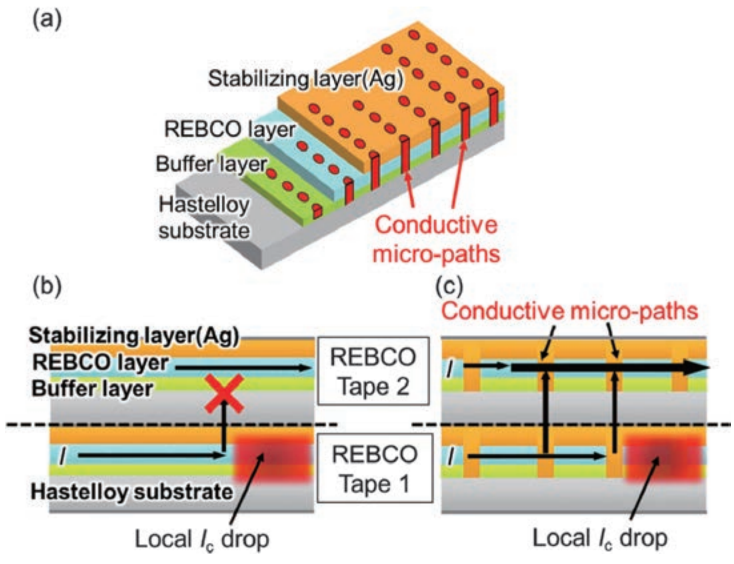


Fig. 9 (a) Schematic drawing of a REBCO tape with conductive micro-paths. Illustrative images of current sharing between the REBCO tapes including local defects with (b) and without (c) conductive micro-paths. Reproduced from [2].

[2] H. Yamada, Y. Tsuchiya, Y. Yoshida *et al.*, "Conductive micro-paths for current sharing between REBCO tapes in high-Tc superconducting conductors to improve stability", IEEE Transactions on Applied Superconductivity **32**, 6602204 (2022).

(Y. Yoshida (Nagoya Univ.) and Y. Onodera)