AP Oral | Large Scale Applications

Wed. Dec 3, 2025 3:00 PM - 5:00 PM JST | Wed. Dec 3, 2025 6:00 AM - 8:00 AM UTC Room C(103)

[AP5] Fusion

Chair:Michinaka Sugano(KEK)

3:00 PM - 3:30 PM JST | 6:00 AM - 6:30 AM UTC

[AP5-01-INV]

Status and Progress of Superconducting Magnet for Fusion in China

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3:30 PM - 4:00 PM JST | 6:30 AM - 7:00 AM UTC

[AP5-02-INV]

Progress on the HTS Magnet Design for the FAST Project at the Conceptual Design Stage

*Sadanori Iwai[1], Yoshikazu Takahashi[1], Kiyoshi Yoshida[1], Kenzo Ibano[2], Kenji Tobita[3], Takaaki Fujita[4], Miki Nishimura[1], Satoshi Konishi[1] (1. Kyoto Fusioneerin.Ltg (Japan), 2. The University of Osaka (Japan), 3. The University of Tokyo (Japan), 4. Nagoya University (Japan))

4:00 PM - 4:30 PM JST | 7:00 AM - 7:30 AM UTC

[AP5-03-INV]

Advances in Practical REBCO HTS Magnet Design: Modeling, Validation, and Reliability

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4:30 PM - 5:00 PM JST | 7:30 AM - 8:00 AM UTC

[AP5-04-INV]

Conceptual Design Study for Japanese Fusion DEMO Reactor

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Status and Progress of Superconducting Magnet for Fusion in China

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Abstract

Due to their higher critical current and lower cooling cost, HTS magnets are being increasingly adopted in fusion device. HTS magnets can realize 20 T level high magnetic field, a crucial factor for controlling the high temperature plasma and enhancing fusion power. Extensive researches on HTS fusion are underway at various institutions and companies, which are aiming for validating the performance of HTS magnet under fusion operating conditions.

Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) is now focusing on the large-scale hybrid superconducting central solenoid coil for the new-generation compact fusion device, capable of achieving a 19.6 T magnetic field in the inner HTS magnet, which is wound with HTS CICC conductor with an operating current of 46.5 kA. It can generate about 55 voltage-second to realize easier plasma discharge. To realize higher current capacity and improve preparation process of HTS conductor, ASIPP is currently developing the new TMCC conductor, which simplifies the preparation process by removing the need for soldering and twisting. TMCC conductor can realize better anisotropic properties and suppress the AC loss during fast current ramping rate, which is capable of achieving 50 kA@20T, 4.2 K. Based on TMCC conductor, ASIPP is working on the 21 T D-shape HTS fusion magnet, which is designed to be operated at 25 kA and total stored energy is about 94 MJ. It is aiming to be used on the next-generation high-field compact fusion reactors. Considering the critical performance degradation of HTS magnet under fusion D-T discharge irradiation conditions, ASIPP has carried out a series of pioneering research work on the damage formation mechanism and critical characteristics evolution process of HTS materials, and scaling-law of REBCO under stress and irradiation has been corrected. It has laid a theoretical foundation for the development and safe operation of large-scale HTS fusion magnets for fusion reactors.

In the research of small-scale HTS fusion devices, Energy Singularity Company in China has designed the world's first full HTS tokamak-HH70 (6.8 m height, 3.6 m outer diameter, 1 T toroidal magnetic field @ R=0.7m). It has successfully achieved its plasma discharge in 2024. The Startorus Fusion, based in Xi'an, China, is cooperating with Tsinghua University on the development of a full-HTS Spherical tokamak device. They achieved first plasma based on the SUNIST-2. Currently, they are working on the world's first Negative Magnetic Spherical Tokamak (NTST), characterized by a 1 T magnetic field and a plasma center radius of 0.4 m radius at plasma center. Meanwhile, the R&D work on 3 T D-shaped TF magnets is also in progress.

References

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Keywaords: Superconducting Magnet, Fusion reactor*, Conductor, HTS, TMCC*

Progress on the HTS Magnet Design for the FAST Project at the Conceptual Design Stage

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Abstract

The FAST (Fusion by Advanced Superconducting Tokamak) project[1)] was initiated in 2024 as a private sector fusion tokamak project in Japan with the collaborators from a variety of expertise fields. The primary objective of this project is to demonstrate the capability of fusion power generation, integrating not only a fusion reactor but also other power plant components, such as a heat extraction system comprising a blanket, a heat exchanger, a turbine, and a plasma exhaust system from D-T fusion power of ~50 MW with a major radius of 2-3 m. The conceptual design stage will be completed by the end of FY2025, and the radial build of the tokamak is on-going, taking into account all constraints relevant to plasma physics, neutronic requirements, and the engineering feasibility of the superconducting magnets and other components. As is customary, the superconducting magnets are composed of the center solenoid (CS), the toroidal field (TF) and the poloidal field (PF) coils. Meanwhile, the magnet cables take advantage of the high current capacity of the REBCO-coated conductors under high magnetic fields (14-16 T for the TF coil) thereby facilitating the compact construction of magnets within constrained spatial dimensions. The presentation will report on the latest progress of the HTS magnet design.

References

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Keywaords: Fusion, Tokamak, HTS, REBCO

Advances in Practical REBCO HTS Magnet Design: Modeling, Validation, and Reliability

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High Temperature Superconductors (HTS), and particularly REBCO (Rare-Earth Barium Copper Oxide) conductors, are enabling a new generation of compact, fast-ramping, and quench-safe magnets. While their potential for disruptive use across multiple sectors is widely acknowledged, challenges remain in accurately modelling their electromagnetic behaviour and ensuring that simulations reliably predict performance in real systems.

This work reviews the advances we have achieved in simulation and modelling of REBCO HTS magnets, focusing on methods to capture quench dynamics [2] and screening current effects [1]. Emphasis is placed on the matching between modelled and experimentally measured responses, with results showing close correlation under dynamic operating conditions.

Representative examples of magnet prototypes are presented to illustrate validation of simulation models against experimental data, including demonstrations of fast-ramping operation**,** stable high-field generation**,** and their intrinsic quench safety**.**

By systematically comparing modelling outputs with real world measurements, TEM demonstrates its ability to provide robust, predictive simulation frameworks for REBCO HTS magnets. This builds confidence in the use of our modelling and simulation tools as a foundation for designing reliable, application-ready magnet systems.

References

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Conceptual Design Study for Japanese Fusion DEMO Reactor

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Abstract

The conceptual design of the Japanese demonstration (DEMO) reactor is being carried out by the Joint Special Design Team for fusion DEMO to establish the Japanese DEMO concept, named "JA DEMO" [1]. The following values are set for the main design parameters of JADEMO to meet the requirements of the DEMO reactor [2]. The plasma major radius (R_p) is 8.5 m, fusion output (P_{fus}) is 1.5-2 GW, the net electric power (P_{net}) is 0.2-0.3 GW, and the magnetic field on the plasma axis (B_t) is 6 T. On the other hand, from the viewpoint of early power generation demonstration, the larger reactor in the conventional JADEMO concept leads to a more extended construction period and higher development risk. Therefore, based on ITER's experience in manufacturing toroidal field coils and its ability to foresee burning plasma (high energy multiplication), for the early power generation demonstration, a conceptual design study was carried out on a DEMO reactor downsized from JA DEMO (R_p = 8.5 m) to the ITER size (R_p = 6.2 m), with a step-by-step approach to demonstrate early power generation and tritium breeding, and to obtain the net electric power of 100 MW-class.

By improving the in-vessel components step by step in a single device, the DEMO rector concept was presented that could achieve a net electric power of more than 0 with ITER-like parameters in Phase I, demonstrate comprehensive tritium breeding for self-realization with JA DEMO-like parameters in Phase II, and achieve the net electric power of 100 MW-class with JT-60SA-like parameters in Phase III. As initial parameters, the TF coil was assumed to have the same dimensions and performance as the ITER-TF coil (superconducting wire: Nb_3Sn, conductor current: 68 kA, design stress: 667 MPa (yield stress: 1000 MPa)). On the other hand, assuming the use of high-strength cryogenic steel with a yield stress of 1200 MPa [3] and 83 kA conductors, which have been developed for JA DEMO, the magnetic field on the plasma axis B_t will increase by 0.35 T and P_{net} by about 15 MWe. These are very important to ensure net electrical output in Phases I to III, and since TF coils are difficult to upgrade step by step, it is essential to complete these R&Ds as soon as possible.

References

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Keywaords: Fusion DEMO, TF coil, large coil, LTS