

TOPICAL REVIEW

Superconductors for fusion: a roadmap

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Topical Review

Superconductors for fusion: a roadmap

Neil Mitchell¹ , Jinxing Zheng² , Christian Vorpahl³ , Valentina Corato⁴ ,
Charlie Sanabria⁵ , Michael Segal⁵, Brandon Sorbom⁵ , Robert Slade⁶ , Greg Brittles⁶,
Rod Bateman⁶, Yasuyuki Miyoshi¹, Nobuya Banno⁷ , Kazuyoshi Saito⁸, Anna Kario⁹ ,
Herman Ten Kate⁹ , Pierluigi Bruzzone¹⁰ , Rainer Wesche¹⁰, Thierry Schild¹,
Nikolay Bykovskiy¹¹ , Alexey Dudarev¹¹, Matthias Mentink¹¹ ,
Franco Julio Mangiarotti¹¹ , Kamil Sedlak¹⁰ , David Evans¹²,
Danko C Van Der Laan^{13,14} , Jeremy D Weiss^{13,14}, Min Liao¹  and Gen Liu¹

¹ ITER Organization, 13067 St. Paul-lez-Durance, France

² Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP), Hefei, People's Republic of China

³ EUROfusion, PMU PPPT, Boltzmann Strasse 2, D-85748 Garching, Germany

⁴ ENEA, 00044 Frascati, Italy

⁵ Commonwealth Fusion Systems, Cambridge, MA 02139, United States of America

⁶ Tokamak Energy Ltd, Oxfordshire OX14 4SD, United Kingdom

⁷ National Institute for Materials Science (NIMS), Tsukuba 305-0047, Japan

⁸ Japan Superconductor Technology Inc. (JASTEC), 651-2271 Kobe, Japan

⁹ Faculty of Science and Technology, Energy Materials and Systems, University of Twente, 7500 AE Enschede, The Netherlands

¹⁰ Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-5232 Villigen PSI, Switzerland

¹¹ CERN Technology Department, Geneva, Switzerland

¹² Advanced Cryogenic Materials Ltd, Oxfordshire OX14 2HQ, United Kingdom

¹³ Advanced Conductor Technologies LLC, Boulder, CO 80301, United States of America

¹⁴ University of Colorado, Boulder, CO 80309, United States of America

E-mail: neil.mitchell@iter.org

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Abstract

With the first tokamak designed for full nuclear operation now well into final assembly (ITER), and a major new research tokamak starting commissioning (JT60SA), nuclear fusion is becoming a mainstream potential energy source for the future. A critical part of the viability of magnetic confinement for fusion is superconductor technology. The experience gained and lessons learned in the application of this technology to ITER and JT60SA, together with new and improved superconducting materials, is opening multiple routes to commercial fusion reactors. The objective of this roadmap is, through a series of short articles, to outline some of these routes and the materials/technologies that go with them.

Keywords: fusion, superconductors, fusion power, superconductor technology, superconducting materials, superconducting magnets, energy

(Some figures may appear in colour only in the online journal)

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1. Background and introduction to the roadmap

Introduction

Superconductor technology has always been a part of the dream of achieving nuclear fusion by magnetic confinement, pre-dating even the discovery of the best-performing magnetic confinement device, the tokamak, in the 1960s [1]. Throughout the last 50 years, steady progress has been made in the use of superconductors in a range of superconducting magnetic confinement devices, starting with T-7 in 1979 [2] shown in figure 1, and culminating in the construction of the huge ITER tokamak, the first device designed with full nuclear operation in mind [3].

The construction of ITER has also been associated with steady progress in what may be called ‘fusion plasma engineering’, where success in improving energy confinement in experimental machines gives confidence in plasma performance predictions for ITER. This increased confidence is giving rise to a series of proposals for post-ITER machines, capable of producing net nuclear power. These fall into two categories, the first of which may be named ‘big machines’, which are DEMO-type fusion reactors based on the ITER design concept and technologies [4, 5], supported by the plasma performance of the large tokamaks that operated in the 1990s. The second category is the ‘compact machines’ based on more advanced technology (especially in superconductivity) [6, 7].

Within this process and conceptual design activity, there is a lack of focus on the targets for the superconducting community to achieve to develop superconductors that will provide the most useful technology for these future nuclear devices. ITER itself was designed, during its operating life, to allow testing of nuclear/plasma technology in the areas of irradiation performance of materials, first wall, blanket and divertor technologies and plasma control. However, there is no such matching program for the superconducting magnets, which in ITER remain substantially based on the magnet technologies from the 1980s and 90s.

The objective of this roadmap article is to provide a series of snapshots by superconducting experts of their personal views of the most important avenues of development for superconducting technology to follow, when applied to a nuclear magnetic confinement device. These articles include of course the obvious avenues of superconducting material, including the high-temperature superconductor (HTS), but also some on the supporting technologies (and demonstrators for them) that are needed to improve the applicability of these materials. These technologies, among them quench protection, operational voltages, reliability and maintainability, are tightly integrated into the materials and the manufacturing routes for magnets, but also merit development of their own since they play a key role in the ultimate usability of the superconductor materials.

Magnetic confinement

Figure 2 shows the three forms of magnetic confinement devices that appear in this article. The most common (and that

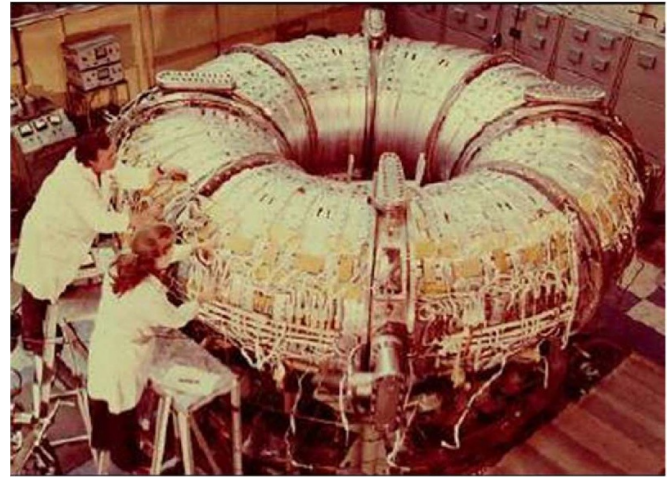


Figure 1. Where it started: T-7 tokamak, Kurchatov Institute, Moscow, 1979, with NbTi TF coils (picture from <https://alltheworldstokamaks.wordpress.com/gallery-of-external-views/t7/>).

with the best plasma confinement performance) is the tokamak, with its three basic sets of coils: the toroidal field (TF) coils, the central solenoid (CS) coils and the poloidal field (PF) coils. Then, the stellarator appears in two forms, one with discrete but three-dimensional coils and the other with largely toroidally wound spiral coils. The tokamak has a plasma current, but the stellarators do not. The stellarator Large Helical Device (LHD type) is the basic design considered in the article ‘Overview of HTS joint technology for segmented coils’. A specific type of tokamak, the spherical tokamak, appears in ‘REBCO magnet technology—a key enabling technology for compact fusion devices’.

Superconductivity and fusion: a brief history

The first generation of magnetic confinement devices to use superconductivity was based on NbTi and Nb₃Sn strand technology.

- T-7 Kurchatov (1979) had NbTi TF coils (first with superconductors).
- The Mirror Fusion Test Facility (MFTF) used NbTi and Nb₃Sn, fully superconducting, completed in 1984, but not operated.
- TRIAM-1M Kyushu University (1986) used Nb₃Sn superconductor in its 16 D-shaped TF coils, cooled by pool boiling liquid helium (first to operate with Nb₃Sn).

In parallel with these confinement machines, the superconductor community recognized the need for a large-scale development and test of ‘magnets for fusion’. This led to the IAEA Large Coil Task (LCT), a collaboration between the US, Japan and the European Union in the late 70s and 80s (figures 3 and 4). This was conceived as a stimulus to provide integrated

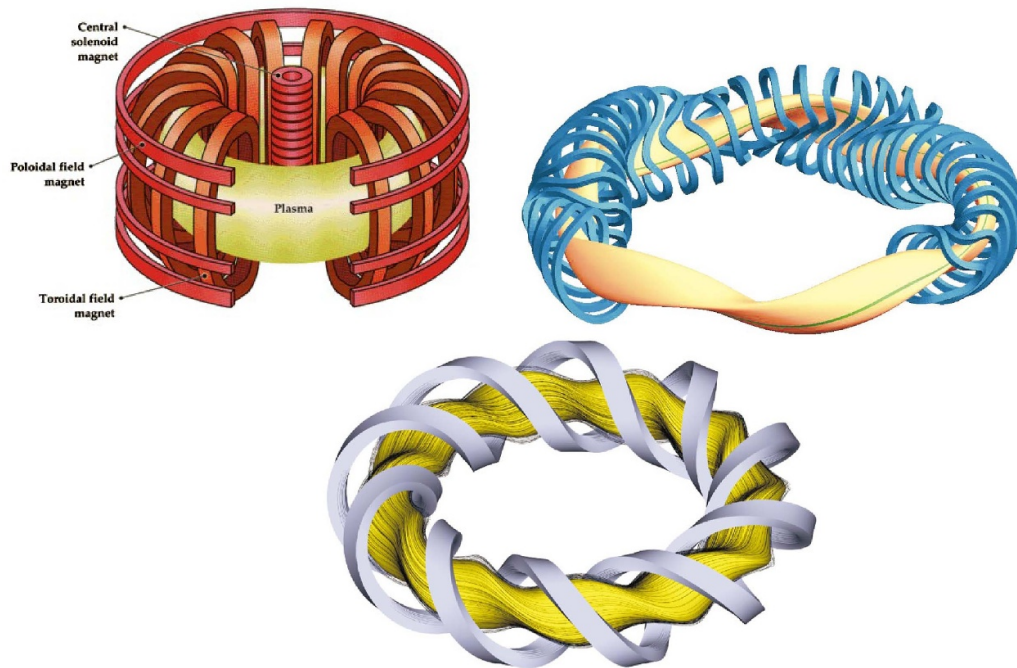


Figure 2. Diagrams of the coil systems needed for magnetic confinement. Top left: general tokamak such as JET, EAST, KSTAR and JT60. Top right: a stellarator like W7X. Below: another form of stellarator like LHD.

conductor and coil solutions to be tested in a shared environment [8]. The ancestor of the cable-in-conduit Nb_3Sn conductors used in the ITER project can be clearly recognized in one of the coils (the Westinghouse one).

Almost at the same time, a collaboration between the US and Japan manufactured and tested model pulsed coils in a facility in Japan [9]. This showed some unforeseen behavior regarding current distribution, which prevented several coils from reaching the intended performance but also thereby influenced the design of later conductors for ITER.

Following the start of the ITER project in 1988, one of the activities initiated during the engineering design activity (EDA) phase in the 1990s was that of the ITER model coils. Two of these were produced, a CS model coil (MC) and a TF MC, and successfully tested, demonstrating many of the critical technologies later used in ITER coils [10, 11].

Subsequent superconducting magnetic confinement devices have not substantially changed the technology, with cable-in-conduit NbTi and Nb_3Sn used exclusively since the 1990s. After the 1980s, the increasing magnetic energy (and hence the need for high voltage during fast discharge to reduce the copper needed for protection) and higher magnetic loads drove the selection of solid insulation systems. Internally cooled conductors were used, with glass-kapton-resin becoming a general standard.

- Tore Supra, CEA Cadarache, France (1988), NbTi TF coils run at a temperature of 1.8 K
- T15, Kurchatov (1988), largest Nb_3Sn TF coils
- LHD, Toki, Japan, NbTi (1998)
- Wendelstein 7X Stellarator 2015 NbTi

- KSTAR, Daejeon, S. Korea, Nb_3Sn and NbTi (2008)
- EAST, first fully superconducting tokamak, NbTi (2006)
- JT60SA, Nb_3Sn and NbTi (due 2021).

Future fusion reactors and superconducting technology

The main driver of magnetic confinement for fusion is of course the plasma physics and the confinement. Modeling of the plasma core suggests that there are windows for ‘compact’ machines, which can exploit the high field or very low aspect ratio (or both) to achieve a higher power density than the larger medium field and large machines characterized by ITER and JT60SA. However, the outer plasma boundary and first wall/divertor interaction (and control) with ‘hot’ plasmas is still relatively uncertain. The heat loads on the plasma-facing components (PFCs) are proportional to the surface area, and so, for the same plasma power, large tokamaks have lower PFC heat loads. Minimizing these loads has for ITER been one of the main design drivers for the size since the PFCs are one of the most technically challenging and uncertain areas for future fusion power plants. In the future, we see compact machines being proposed that aim to exploit the plasma confinement regime; this idea is becoming more attractive due to developments in superconducting technology and materials. By using less of these materials they can reasonably claim to be able to exploit novel superconducting materials with a weaker industrial base. These appear alongside ‘traditional’ machines, which are built on the ITER technologies and which tend to extrapolate the plasmas obtained in the tokamaks of the

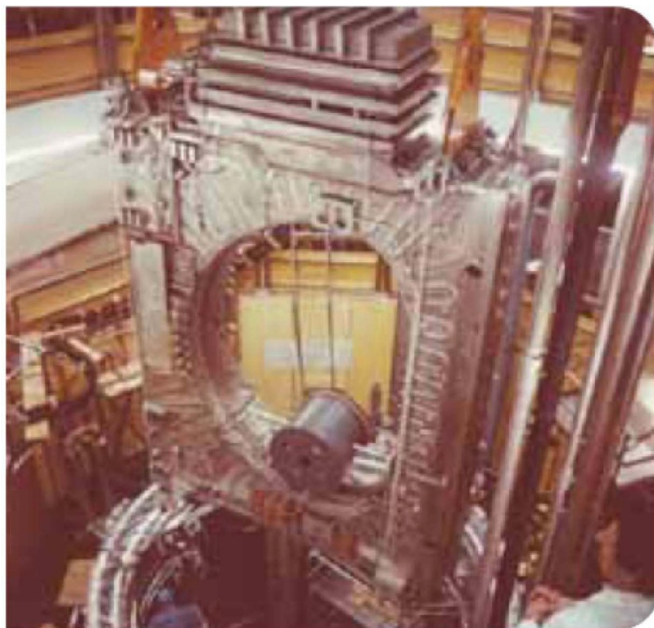


Figure 3. TF model coil and pulse coil in LCT.

1990s in terms of the aspect ratio, dimensions, TF and PFC performance.

What could the key superconductor technologies be for the future? Of course, the answer depends also on the plasma operational requirements on the magnets to achieve good confinement and also on the capabilities of the components facing the plasma: a compact high-field machine ought to produce a high fusion power but also imposes very high thermal and nuclear fluxes on the PFCs. But a key technology for fusion is applied superconductivity, and the fusion field cannot wait 15 years for plasma performance data to be confirmed by machines like JT60, ITER and the new spherical tokamaks in order to decide on the technology that is needed and then spend another 15 years developing it. The point of this roadmap article is to look at directions in which to encourage the development of superconductivity in order to be in a position to deliver what is needed for a fusion reactor.

Superconducting magnets can be broken down into four components

- (a) Conductor
- (b) Structures
- (c) Insulation
- (d) Instrumentation

and the key factor to use them is integration. The overall usefulness of any integrated solution using these components is also strongly influenced by the quality (and therefore reliability) and repairability.

The integration of the four components depends on the final machine parameters but also on the technologies selected for them. In a research environment, the last two items are sometimes not given the weight that they will receive in a nuclear environment. To draw attention to the likely expectations of a



Figure 4. Six LCT coils in a vacuum tank (October 1985).

nuclear regulator, we have included several articles relating to these last two bullets.

The roadmap articles each span some parts of the list above, but not all. To provide an overview of the roadmap content we have selected five areas for use as a content index (see table 1):

- (a) Superconductor material
- (b) Conductor design and manufacturing
- (c) Conductor integration in magnet (including instrumentation, insulation, selection of conductor current and voltage levels)
- (d) Systems engineering (quality and repairability, which cover items like demountable coils, demonstration coils and proof of reliability)
- (e) Magnet and machine design.

Economics of superconductors in fusion

Absolute cost estimates of the magnets within a fusion reactor, even if available for a detailed design, are difficult to work with, being subject to many assumptions and often considerable optimism. To put the magnet cost within the context of the full plant cost, two sources have been selected, one the ITER Final Design Report (FDR) from 2001 [12] and the other a study of the costs of a full electricity-generating plant [13]. Figure 5 shows the relative contributions. Neither of these include the engineering costs, which in the case of such first-of-a-kind machines could approach the direct costs [13]. In the case of ITER, experience suggests that the relative contribution of the plant costs (including buildings) was significantly underestimated. Figure 5 indicates that in a future fusion plant, with higher building and balance of plant costs due to power production, we can expect the magnet costs to lie in the range of 10%–15% of the total direct cost.

This roadmap is focused on the technology use issues and does not include much discussion on the costs of

Table 1. Distribution of roadmap content.

	SC material	Conductor design and manufacturing	Conductor integration in magnet	Systems engineering	Magnet and machine design
Low-price and high-field hybrid superconducting magnet for future fusion devices	×	×	×	—	×
The DEMO magnet system: unifying mature technology and innovation	×	—	—	×	×
Commercial fusion power: a killer app for HTS	×	—	×	—	×
REBCO magnet technology—a key enabling technology for compact fusion devices	—	—	×	—	×
Nb ₃ Sn wire development for fusion beyond ITER	×	—	—	—	—
REBCO for fusion and fusion for REBCO	×	—	—	—	—
Very high current conductors for reduced operating voltage	—	×	×	—	—
Fusion superconducting cables: design criteria	—	×	—	—	—
Superconducting switches for energy extraction in fusion magnets	—	—	×	×	—
Overview of HTS joint technology for segmented coils	—	—	×	×	—
Advances in react-and-wind	—	×	×	—	—
Nb ₃ Sn coils for fusion	—	—	×	—	—
Plastic resins and composites at low temperatures	—	—	×	—	—
CORC [®] cable and magnet R&D in preparation for the next generation of fusion facilities	×	—	—	×	—
Quality of superconductors in a nuclear environment	—	—	—	×	×

superconductors and superconducting technology. Cost assessments for superconducting magnetic confinement devices often focus on a costing based on the superconducting materials because they are simple to obtain—the iterations on ITER during the period 1997–2001 [14] provide an example—but neglect the real costs of an integrated system as these are much more design-specific and cannot simply be taken as proportional to the superconductor material costs.

ITER experience in the industrialization of Nb₃Sn is quite relevant [15]. In the early stages, Nb₃Sn costs (in terms of Euro/kg of material) were high, and the quality unpredictable due to the limited industrial base, and led to attempts to design the tokamak to minimize them. In the end, with ITER, adroit exploitation of multiple suppliers enabled material costs to be kept under control (even if not as low as originally hoped) despite the natural political desires of the partners to keep a stake in strategic technologies. However, the absolute magnet system costs, particularly from contributions in such basic technologies as large steel forgings, and from the complexity

caused by an excessively compact design, were far above the original estimates.

From this, and considering the relatively low fraction of the total cost driven by the magnets (<20%), we suggest that if costs have to be considered, the trap of basing them on very early estimates of the superconducting materials must be avoided. This can lead to decisions that actually enhance the overall cost because good systems engineering is sacrificed in the interests of minimizing the use of superconducting materials.

Roadmaps

To provide an overview of the content range of the individual roadmaps, table 1 has an index. To provide some structure, the articles have been ordered starting with those that give substantially ‘machine overviews’ (i.e. the rightmost column). Following this, they are selected on the basis of first appearance, from columns 1–4.

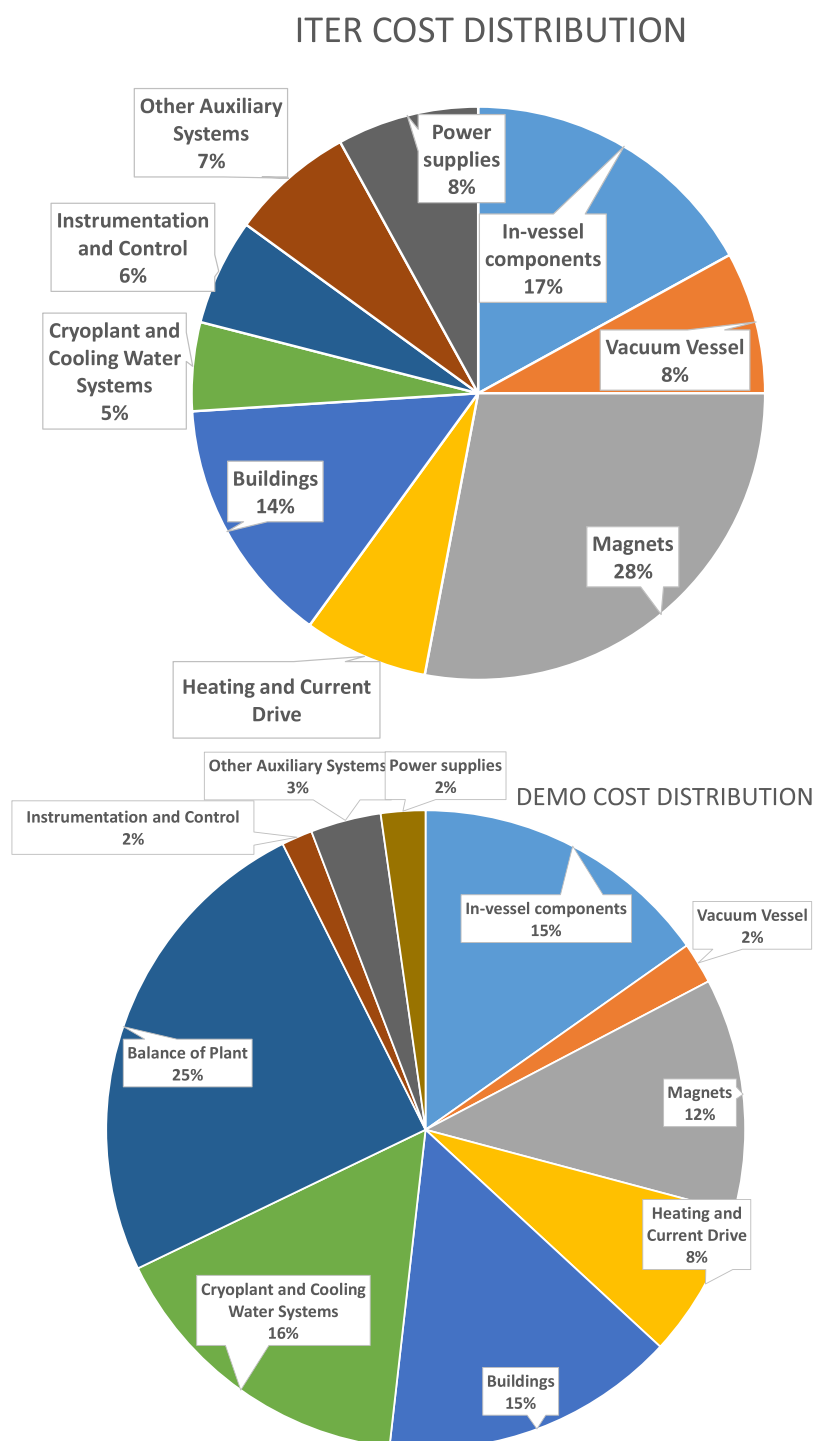


Figure 5. Relative predicted cost contributions for ITER (top) and DEMO (bottom).

Conclusions

It is clear from the range and subject matter of the 14 articles that have contributed, that the superconducting magnet community has multiple opinions on the route to follow to provide proven superconducting magnet technology for a

future power-producing magnetic fusion device. We can see the three drivers behind this in the form of

- (a) Progress in magnetic fusion plasma confinement, to the extent that tokamaks presently under construction will be achieving fusion-grade plasmas. By doing so, they confirm

much of the modeling basis for plasma design and support; in particular, the high-field compact tokamak configurations.

- (b) Progress in HTSs, which has made available a new material with improved operational ranges in temperature and field.
- (c) Experience in nuclear tokamak construction, where repairability and reliability as well as improved overall system integration, are recognized as major considerations in a future power plant design.

All of the 14 contributions pick up one or two of these drivers and look at specific ways in which the progress could be further exploited to eventually provide a conceptual power plant. There is not, however, a common theme that can be obviously picked out so that the multiple specific technologies within the contributions can be integrated to provide a single roadmap for the superconducting field. What we have are several roads on which traffic can proceed rather independently, all with their own difficulties. We have a range of proposals within the articles for assistance to overcome these difficulties with auxiliary technology such as insulation or joints, or design features like high current or quench protection. Our problem in the superconductors-for-fusion area is to identify common themes so that these roads can be linked and difficulties addressed with a common approach, to avoid dissipation of resources into too many possible solutions, none of which end up being brought to completion. We can contrast with the internal (nuclear) components, where ITER itself is designed as a test bed, which can test future reactor concepts for components like the blankets and first wall.

The 14 contributions substantially reflect two fields of activity in nuclear fusion magnets. On one side we have the various contributions related to the DEMO projects, e.g. in China, Europe, Korea and Japan, which are the natural follow-up of ITER. The focus of design and R&D for these projects is on the improvement of the reliability, the reduction of risk in operation, the quality assurance procedures and the effectiveness (i.e. the cost) of the design and manufacturing approach. The lessons learned in the design and construction phase of ITER drive the effort. Typical examples are the reduction of the operating voltage, triggered by the various Paschen test failures during the ITER and W7-X coil manufacture, the react-and-wind revival for very high current Nb₃Sn graded windings, triggered by the low effectiveness of the wind-and-react cable-in-conduit conductor (CICC), the fatigue in the CS, which becomes a central issue for a power plant with a lifetime much longer than that of ITER, the tolerance requirement on large structural items, e.g. the TF cases, which eventually dominates the manufacturing cost and time for very large tokamaks, the quench management, aiming at a robust and resilient approach, passing the scrutiny of the repairability, accessibility, maintenance and inspectability (RAMI) analysis and certification for nuclear-grade components.

In most cases, the issues must be dealt with and solved at design level, the R&D being the necessary and powerful support of specific conceptual approaches. The roadmap (toward a large pulsed tokamak with electrical power output larger than 500 MW) consists of a number of technology improvements, which are presently addressed without real coordination by the various countries with their own national DEMO project. A better exchange of information on design and R&D at international level is desirable and beneficial for an effective use of resources.

The second field of activity, addressed in several chapters of this article, is focused on the use of HTS magnets for fusion devices, which are definitely not in the footprint of ITER. In contrast to the activities described above, here the main objective is to achieve proofs-of-principle and demonstrations for novel technology approaches, ranging from demountable magnets and segmented windings with thousands of demountable joints, non-insulated winding packs, very high field magnets, and indirect cooling. The challenging character of these activities is surely stimulating and attractive for the young generation of scientists, eager to achieve a technological breakthrough within a reasonably short delay. The proposed R&D plans and demonstrations can presently hardly fit into a well-defined roadmap with the many different strategic approaches. An ambitious proof-of-principle, for example the successful test of a full-scale demountable joint for HTS winding segments, is surely a necessary condition for the feasibility of a large torsatron magnet, but may not be a sufficient argument to move ahead on that path.

Similar methods to those already used in the early days of ITER [16] could stimulate DEMO engineering in the magnet area and link these two activities. There are interesting new proposals regarding compact high-field tokamaks as pilot fusion plants, which would not rely on ITER superconducting technology, but the lessons learned from ITER show how improvements can be made in the engineering layout of a tokamak fusion reactor. These proposals bring in new sources of funding, and the superconducting community needs to encourage these developments. From the roadmap articles, we can identify two objectives for such a stimulus and focus, which we could call a Superconductor Engineering Concepts Demonstrator (SECD):

- (a) Testbed for new technology, to engage industry and develop maturity
- (b) Testbed for basic engineering improvements (repair, reliability, effectiveness).

To launch such a stimulus requires funding and, above all, imaginative (and light) technical coordination. The LCT project in Oak Ridge in the 1970s, SULTAN from the 1980s and ITER CSMC in the 1990s are examples of how resources could be combined, engaging many industries by having a modular form, allowing nearly-independent in-kind contributions while requiring a common time schedule. Such a combined facility could be designed to allow the different contributions to work under conditions that

match their test/demonstration requirements. The LCT project provided a test bed for six similar TF coil concepts, but the ITER CS MC facility required an inner coil, an outer coil and several insert coils. We could, as an example, imagine a toroidal test facility with perhaps 10–12 TF coils being contributed, including advanced insulation, demountable coils and innovative quench detection/protection. The focus would be on the integration of the basic

machine (magnets with feeders and instrumentation, vacuum vessel, cryostat and integration and repairability) rather than a conductor test facility as created by SULTAN and the CSMC. An SECD coordination group could also provide a forum for the magnet engineering of the very different individual ‘next-step’ tokamak concepts to be discussed with some engineering exposure and expert review for the different magnet concepts.

2. Lowprice and highfield hybrid superconducting magnet for future fusion devices

Jinxing Zheng

Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP), Hefei, People's Republic of China

Status

The need for superconducting magnets for fusion reactors has existed for over 40 years. The advanced plasma experiments and future fusion reactors call for a long confinement time and high magnetic fields, which can be reasonably maintained only by superconducting coils. A dozen fusion devices have been built or are under construction using superconducting magnets; for example, EAST, WEST, KSTAR, JT-60SA and ITER with parameters summarised in table 2. The superconducting magnet systems for these fusion devices were manufactured using NbTi or Nb₃Sn CICC.

The magnetic field is a key parameter for the fusion device, and the plasma confinement time is increased with the magnetic field increasing. HTSs have been found to be a good solution in the current leads for high-current fusion magnets, and the HTS fusion magnet has been put forward by MIT in SPARC [23]. An aluminum alloy-jacketed Nb₃Sn superconductor and indirect cooling using cooling panels within the coil was proposed as a candidate magnet system for the LHD-type reactor FFHR [24]. However, the basic engineering issues of HTS technology and the high price are the main reasons that make HTS impractical in the winding of fusion magnets. So, a hybrid magnet design scheme is the optimal solution for high-field magnet manufacturing. As examples, hybrid HTS-Sn₃Sn-NbTi was designed for the EU DEMO CS coil [25], and Nb₃Sn and NbTi hybrids were designed for the Chinese Fusion Engineering Test Reactor (CFETR) CS modular coil [26]. The benefits of a hybrid CS coil are two-fold: to reduce the outer radius of the magnet and to increase the generated magnetic flux [27]. Therefore, hybrid magnets will become the trend for a high magnetic field in pulsed tokamaks.

Current and future challenges

In general, the future fusion reactor magnets will need to be operated in a strong magnetic field environment. From the economic point of view, the hybrid magnet is an ideal choice. For superconducting magnets, there are two categories of hybrid. For a magnetic field less than 16 T, the hybrid is a high Jc Nb₃Sn and NbTi superconductor. The maximum magnetic field of the CFETR TF magnet is about 15 T, which adopts the hybrid of the high Jc Nb₃Sn, ITER grade Nb₃Sn and NbTi, figure 6. The lengths of each conductor are 2000 m, 3200 m and 1700 m, respectively. So, each TF coil can save about 30 million dollars, and the total 16 TF coils can save 480 million dollars. However, high Jc Nb₃Sn has a large hysteresis loss, so it is not suitable for CS and PF magnets. Because the operation mode of CS and PF magnets is pulsed, the current and

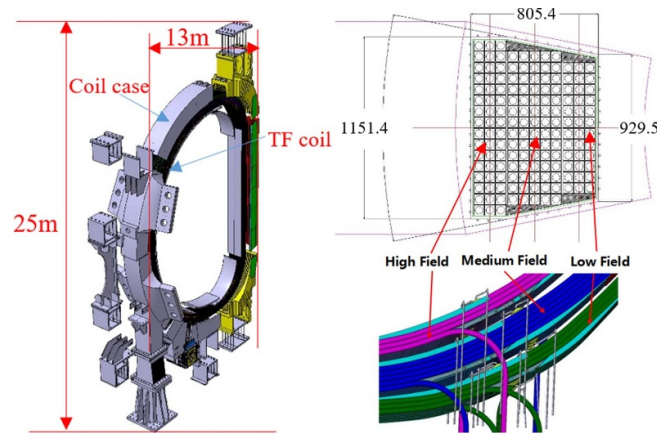
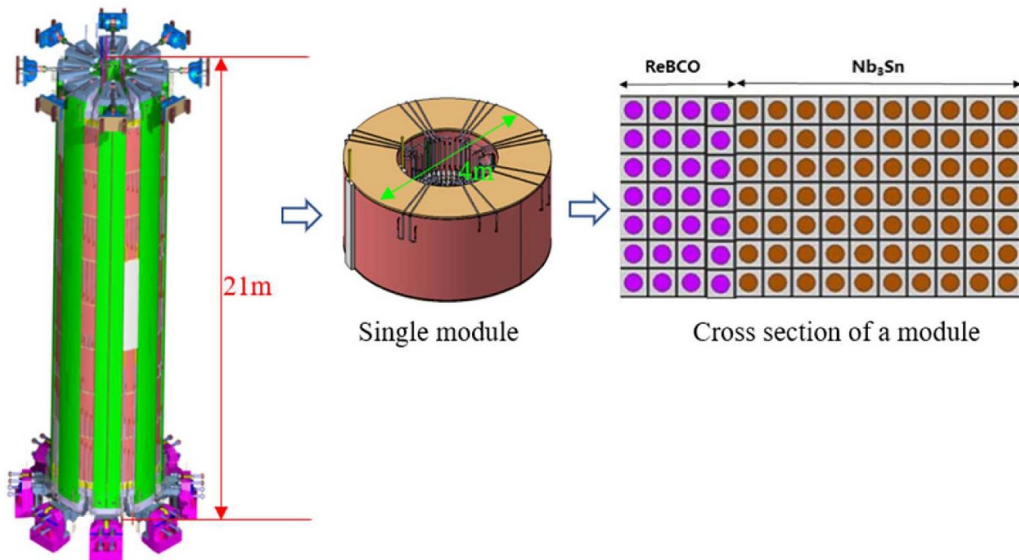
magnetic field will change all the time, so AC losses will be correspondingly large. Due to the intrinsically large hysteresis for high Jc Nb₃Sn wire (2500 mJ/cc for ± 5 T cycles), it is not suitable for CS and PF magnets at the present stage. For a magnetic field higher than 16 T, the hybrid is usually an HTS (Bi2212, REBCO) and low-temperature superconductor (LTS) (Nb₃Sn, NbTi), such as the conceptual design of the CS magnet for CFETR shown in figure 7. Similar to the LTS, Bi2212 composite can be manufactured in circular strands. Its electromagnetic properties are very similar to those of the Nb₃Sn superconductor, so the design method of the CICC conductor based on cryogenic superconducting material is fully compatible with Bi2212 material. But the increase of critical current for the Bi2212 strand should be obtained by heat treatment under about 100 bar high pressure. In addition, the Ag matrix of Bi2212 has the potential to activate under an irradiation environment in fusion devices. The other HTS material, the rare-earth barium copper oxide (REBCO) conductor, has so far been provided only in the form of tapes to ensure grain boundary alignment. The superconducting properties of REBCO tapes are strongly anisotropic. The critical field current J_{c11} arranged along the tape width direction is 4–5 times higher than that in the vertical direction (J_c⊥). Due to the weak ‘adhesion’ between the substrate and copper, there is a high mechanical anisotropy in REBCO. The critical transverse tensile stress and peel stress are 10 times and 100 times lower than the axial tensile stress and peel stress [28]. At present, in addition to the relatively high price of the HTS-CICC conductor, the preparation technology of the CICC conductor based on REBCO tape is still in an immature stage of development. Bi2223 is more suitable for current leads than superconducting magnets. On the one hand, Bi2223 cannot be made into wire like Bi2212 for easy winding, and on the other hand, Bi2223 cannot perform like YBCO, which has a large current-carrying capacity even if there is a high vertical magnetic field.

Advances in science and technology to meet challenges

In view of the technical challenges of hybrid magnets for fusion reactors, a series of new technologies must be developed. The first task is the improvement of material properties. Heat treatment of Bi2212 strands in large coils is a technical challenge. Compared with Nb₃Sn, the peak temperature of Nb₃Sn is higher, up to 878 °C. The development of heat treatment technology directly affects the properties of Bi2212 strands. For large coil heat treatment, the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP), Hefei, China, suggests using medium-pressure argon in a furnace and high-pressure oxygen (up to 100 bar) inside the CICC to improve the Bi2212 properties. The pressure of argon could be evaluated by the jacket thickness and oxygen pressure. But the overpressure heat treatment makes the Bi2212 wires shrink, so the void fraction will be increased. The cable will be loose and may be move under electromagnetic force. The stability of the conductor is decreased. So, the

Table 2. Main parameters of magnet system for EAST, WEST, KSTAR, JT-60SA and ITER.

Item	EAST [17]	WEST [18]	KSTAR [19]	JT-60SA [20]	ITER [21, 22]
Major radius (m)	1.7	2.4	1.8	2.96	6.2
Minor radius (m)	0.4	0.7	0.5	1.18	2
Central magnetic field (T)	3.5	3.9	3.5	2.25	5.3
Maximum magnetic field (T)	5.8	8.1	7.2 T	8.9	13
Maximum operation current (KA)	14.5	1.255	35.2	25.7	68
Superconductor	NbTi	NbTi	Nb ₃ Sn (TF, PF1-5), NbTi (PF6-7)	Nb ₃ Sn (CS), NbTi (PF, TF)	Nb ₃ Sn (TF, CS), NbTi (PF)
TF system energy	300 MJ	480 MJ	470 MJ	1.5 GJ	41 GJ

**Figure 6.** The hybrid magnet of CFETR TF coil.**Figure 7.** The hybrid magnet of CFETR CS coil.

cable structure of Bi2212 CICC is a critical technology. ASIPP has carried out lots of R&D activities on Bi2212 strands for CFETR magnets [29]. The short twist pitch design solved the problem of performance degradation due to strain. For

REBCO tape, great efforts are continuing to improve the performance of the tape by increasing the critical current and its uniformity along the tape length, reducing the substrate thickness and overall product price. At present, the maximum

length of a single tape with fairly uniform transmission is about 1 km. As mentioned above, the current superconducting CICC technology based on wire is not compatible with the tape structure, so a new HTS-CICC structure must be developed. At present, the three main HTS-CICC structures are Roebel, conductor-on-round-core (CORC[®]) and TSTC. ASIPP has developed a nine-turn solenoid magnet wound around a 4 m long CORC[®] cable and tested at 4.2 K in a background magnetic field up to 19 T to check the stability of the current-carrying performance of REBCO cable under the combined action of thermal load and electromagnetic load [30]. No performance degradation was observed after ten warm-up and cool-down cycles of operation at 90% I_c , which built the basis for future fusion hybrid magnets reaching 20 T. On the other hand, the Lorentz load generated by a 20 T magnetic field needs high-strength structural materials to support the magnet, which requires new metals or composite structures with very high yield strength, ultimate strength, ductility and elastic modulus. In addition, insulation materials should be developed to achieve excellent radiation resistance for fusion hybrid magnets.

Concluding remarks

High-field hybrid superconducting magnets may be a good choice for future fusion devices. However, there are some

issues and challenges. High J_c Nb₃Sn has a large hysteresis loss and is not suitable for CS and PF magnets. The hybrid is usually HTS (Bi2212, REBCO) and LTS (Nb₃Sn, NbTi) when the magnetic field is higher than 16 T, as the Ag matrix of Bi2212 has the potential for activation under irradiation, and the superconducting properties of REBCO tapes are strongly anisotropic. In addition to the high price of the HTS-CICC conductor, the preparation technology of the CICC conductor based on REBCO tape is still in an immature stage of development. New technologies must be developed to meet the technical challenges, which include superconducting magnet material properties, heat treatment technology, a new HTS-CICC structure, new metals or composite structures with a high yield strength and ultimate strength, high ductility and high elastic modulus, and insulation materials with a high radiation resistance. Bi2223 is suitable for current leads rather than superconducting magnets.

Acknowledgments

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3. The DEMO magnet system: unifying mature technology and innovation

Christian Vorpahl¹ and Valentina Corato²

¹ EUROfusion, PMU PPPT, Boltzmann Strasse 2, D-85748 Garching, Germany

² ENEA, 00044 Frascati, Italy

Status

The European DEMO project aims for a tritium self-sufficient plant capable of producing hundreds of MW of electric power. The overall strategy is to systematically increase the technical, manufacturing and integration readiness of all systems while complying with nuclear licensing and in view of construction, projected to start in 2040. A key pillar of the strategy is to build on ITER technology and physics, keeping the required development steps¹⁵ between ITER, DEMO and a future commercial power plant credible. In the current DEMO Conceptual Design Phase (CDP, 2021–27), co-existing variants will be further developed to allow down-selection to a unique DEMO baseline design with possibly one back-up concept (for all plant systems, including magnets). At the same time, R&D on attractive technologies with high risk or low maturity (e.g. liquid metal divertor targets) is pursued in a parallel effort. These options are aimed at improving the commercial viability of future fusion reactors and are not required for DEMO to succeed, but may enter its main program if deemed useful. The subsequent engineering design phase (EDP) is intended to de-risk the selected baseline design.

With regard to magnets, the recent DEMO pre-conceptual gate review in 2020 confirmed the feasibility of several concepts per subsystem (TF, CS and PF). The innovative coil options offer an improved performance and cost-saving potential. The more ‘conventional’ options are mostly ITER-like, where extensive expertise exists and major difficulties, including industrial production, have already been identified and addressed.

DEMO is a large tokamak with large coils. Its design point at $R_{\text{major}} \approx 9$ m and $B_{\text{T (on axis)}} \approx 6$ T defines a machine roughly 1.5 times the size of ITER in linear dimension. It is, however, important to note that the available design space is rather narrow. Following for instance the 0 D modeling in [31] for 2 GW fusion power, systematic limits are due to (a) the impurity concentration needed for divertor detachment $\hat{c}_{z,\text{det}} < 1$, (b) H-mode access $f_{\text{LH}} > 1$ and (c) the allowable divertor power load $P_{\text{sep}} \times B_{\text{T}}/q \times A \times R < 9 \text{ MW} \times \text{T m}^{-1}$. Only small variations are possible without violating these strict limits: the extremal values for R_{major} and B_{T} are

8.3 m (more compact) and 6.4 T (higher power possible), respectively. However, R&D progress or breakthroughs *in several fields*, e.g. target loads and physics or materials, would have a big impact and open up the available design space¹⁶.

Carrying the previous example a bit further, the options imply either ‘slimmer’ coils or a higher field. In these cases, structural materials with higher strength would be required for the TF inner leg, which is tokamak size-driving and hence dimensioned with minimum margins. Compared to ITER’s 316LN, it may seem justified to assume that improved materials will be available in 2040. However, this may also have been the general belief in the 1980s, when a systematic screening for ITER was performed with much effort, see e.g. [32]. From the 89 candidate high-strength cryogenic steels investigated, all promising on the lab scale, only one (JJ-1) was validated for final application in ITER, and with a relaxed specification. Typically, issues arise during process industrialization, e.g. controlling bulk properties of large material quantities or large-scale forging and welding of materials that are unusual for suppliers [33]. It is unlikely that R&D on advanced steels can contribute to DEMO, at least at reasonable effort, which is why this is not part of the CDP program. The return of experience from ITER is, however, of key importance. As a general conclusion, the extrapolation of small-scale properties to large components bears risks not to be underestimated.

The 16-fold segmentation of the DEMO machine is a compromise between vertical port space and breeding blanket segmentation to allow efficient remote maintenance on the one hand, and to limit the size and total current to be carried per TF coil on the other. In addition, limiting fast particle losses from the plasma requires the TF variation (‘ripple’) to be less than 0.3%, defined at the equatorial outer mid-plane plasma surface. As DEMO has ‘only’ 16 TF coils, the relative toroidal distances between them are large, resulting in large ripples. In order to flatten the field, the outer TF limb is moved away from the plasma¹⁷. This results in wider TF coils, as visible in figure 8, also moving the PF coils further from the plasma. Another major design parameter is the *minimum* time constant τ of fast TF coil discharges, e.g. triggered by the detection of a quench. A faster discharge causes higher stresses in the vacuum vessel (due to $j \times B$ forces) as well as higher peak voltage to be withstood by the insulation. DEMO is using a time constant as high as $\tau = 35$ s (ITER $\tau = 11$ s) as the design requirement¹⁸. This calls for a considerable Cu fraction in the conductors and thus enlarges the winding pack. In the same

¹⁵ These steps are not small. As a physics example, note that the ITER (and DEMO) design relies in terms of confinement time on an empirical scaling, because no first-principles physics model is available. ITER itself is expected to validate predictions for burning plasmas for the first time and will perform D-T experiments during DEMO’s Engineering Design Phase. The findings may alter the DEMO design significantly.

¹⁶ In practice, the DEMO design point of course depends on more than a few parameters, and so-called system codes [180] take into account many more input variables, such as the required thicknesses of components for neutron shielding, breeding etc. In the above model, these are implicit. A general conclusion is valid for both: the more challenging (i.e. risky) the set of input assumptions, the more attractive the power plant concept. Therefore, fusion concepts benefit from well-defined assumptions (i.e. a high *overall* level of confidence), which are continuously improved through technological development and modeling.

¹⁷ Ferritic inserts in the vacuum vessel also help to reduce the TF variation.

¹⁸ The previous vessel design only marginally fulfilled the above stress criterion, but a recent design update with dedicated changes alleviated this issue.

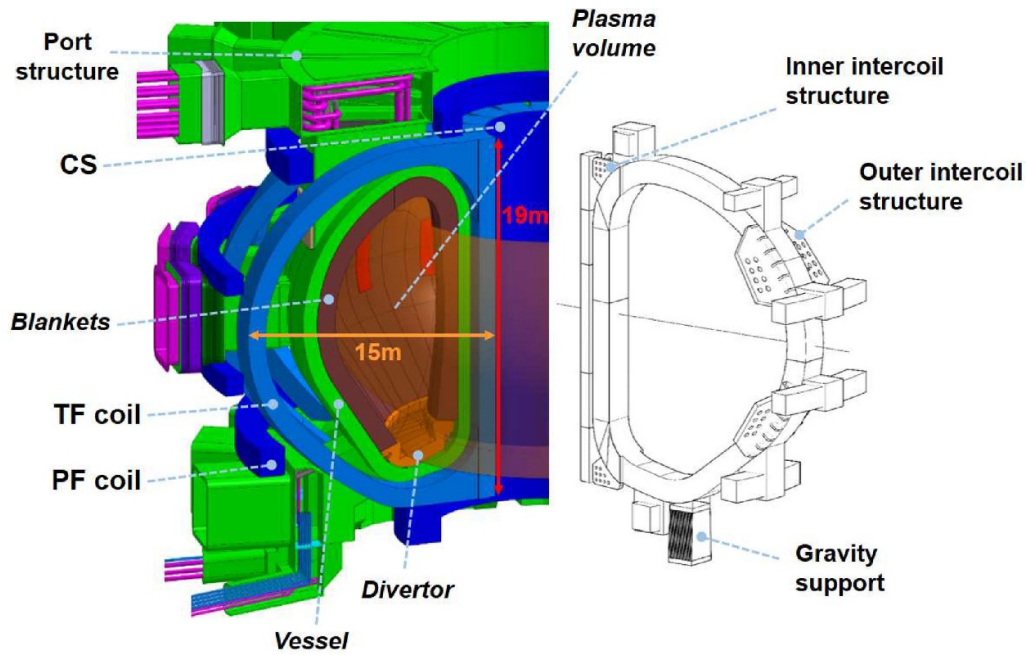


Figure 8. Magnet system integrated in DEMO. Cut-away view of configuration model (low detail, feeders not shown, left); TF coil 22.5° model (right).

context, the TF coil shape was optimized for minimum magnetic stored energy, deviating from a bending-free¹⁹ contour (see the high-curvature region on the bottom), which complicates the mechanical design. The magnet system represents a large part of the mass and cost of the tokamak and may seem to govern its architecture. However, as outlined above, requirements from the plant level and other systems affect the magnet design, sometimes in complex interactions. To produce the 2027 DEMO conceptual design, these will be further quantified and balanced.

An overview of the integrated DEMO magnet system is given in figure 8. Table 3 presents the main parameters of the system specification.

The system architecture is ITER-like and consists of 16 TF coils, 6 PF coils and a CS to enable 2 h pulses. The TF and PF coils are mechanically connected and supported by 16 gravity supports located underneath the TF coils. Inter-coil structures between the TF coils, together with the central wedges, bear out-of-plane forces on the coils when subjected to a vertical field during operation. Cryogenic coolant and electric power are supplied via magnet feeders.

Two options for every coil system are considered in the CDP as presented in table 4, together with their main design characteristics. Several options overfulfil their specification (for instance, TF option #1 is capable of achieving a ~20% higher field). The two TF winding pack options included here were down-selected in 2020 from four previous options.

¹⁹ Bending-free in self-field, not during operation.

Table 3. Main parameters of DEMO and ITER (for comparison).

Item	DEMO	ITER
Fusion power	2 GW	0.5 GW
Plasma volume	2580 m ³	816 m ³
Major radius	9.1 m	6.2 m
Minor radius	2.9 m	2 m
TF on axis	5.3 T	5.3 T
Max. TF	12 T	11.8 T
Number of TF coils	16	18
TF overall height	~19 m	~13.5 m
TF system stored energy	150 GJ	41 GJ
Fast discharge time constant	35 s	11 s
Centering force per TF	850 MN	400 MN
Vertical force on 'half-TF'	520 MN	200 MN

Current and future challenges

Although ITER-like concepts are mostly transferrable to the European DEMO magnet system, the following critical issues need to be addressed before the final design selection.

(a) Higher mechanical loads

Mechanical loads in DEMO are much higher than in ITER. To give an example, the centering force per DEMO TF coil is 850 MN vs. 400 MN in ITER, and the vertical force on half a TF coil is 520 MN vs. 200 MN [34]. This requires robust concepts for coils, TF cases and inter-coil structures

Table 4. Pre-CDP coil options to be further developed in the CDP.

Sub-system	Option #1	Option #2 (ITER-like)
TF	RW Nb ₃ Sn, layer winding, grading both SS & SC	WR Nb ₃ Sn, double pancakes w. radial plates (no grading)
CS	Hybrid: HTS-RW Nb ₃ Sn-NbTi, layer winding (cost-effective, slightly higher magnetic flux)	WR Nb ₃ Sn, pancake winding
PF	PF1 & 6 made of RW Nb ₃ Sn (peak field close to 7 T); PF2-PF5 in NbTi. All coils pancake winding	PF1-PF6 in NbTi, pancake winding

R&W = react-and-wind, SS = stainless steel, SC = superconductor, W&R = wind-and-react, HTS = high-temperature superconductor.

in order to sustain high forces under a fatigue load over ~30 000 plasma pulses.

(b) Higher long-term availability of magnets at full performance

Since DEMO aims to demonstrate the commercial viability of fusion power, the availability of the plant (i.e. the fraction of time it is operational and produces energy over its lifetime) is a key driver. The strategy to maximize the power production and reduce the maintenance durations is to have a robust design that foresees minimal levels of planned maintenance and guarantees full performance over several decades. A challenging aspect of magnet design is to eliminate the degradation of both the conductor and the insulation due to mechanical and thermal cycles. Efficient quench detection and protection systems are needed to guarantee availability of the coils in the long term.

(c) Scalability

DEMO magnets cannot be built by simply upscaling the ITER design, because industrial and logistic limits are already being stretched for the ITER coils. Examples are forging and welding of the TF coil cases due to very large thicknesses, handling a ~900 ton TF coil, compared to 360 tons of the ITER TF coil, as well as the transportation of a TF coil of 19 m height and 15 m width.

(d) Cost

From the long-term perspective, fusion has to be economically competitive with other sources of energy such as solar and wind. This requires reducing the total cost and increasing the net electricity production through improved reactor performance. To this end, innovative designs and manufacturing approaches will be validated in the CDP of DEMO, including but not limited to layer-wound coils, react-and-wind Nb₃Sn technology and HTS modules. Cross-system dependencies, e.g. with the Cryoplant, must be closely monitored as well.

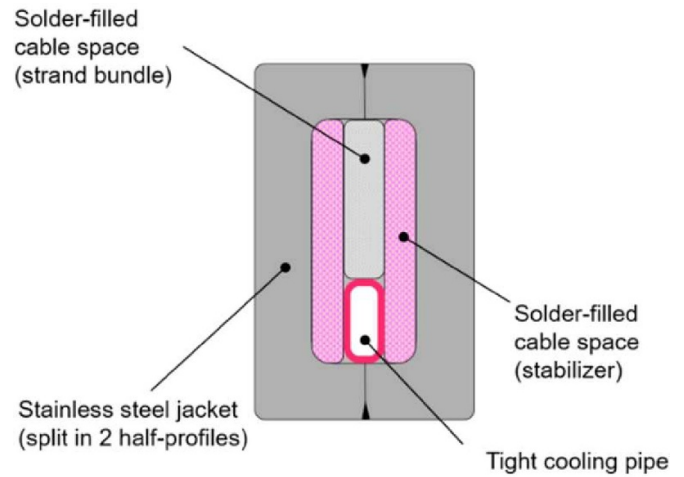


Figure 9. Proposal for an alternative CS conductor design decoupling the two functions of the CICC jacket (structural support and helium containment).

Advances in science and technology to meet challenges

Innovative design and technology solutions are required to meet the challenges of the DEMO magnet system. The strategy concerning magnets adopted pre-CDP until 2020 is presented in [35] and [36]. Demonstration at industrial level and validation of advanced solutions have been included in the CDP work-plan (2021–27). The main topics are reported below.

Structures to sustain the enormous forces acting on the coils and the fatigue load will be considered at different levels. At conductor level, the jacket in standard CICC acts both as a structural support for the operational loads and as helium containment. In a magnet subjected to cyclic mechanical loads like the CS, the hydraulic containment function imposes the most stringent constraint from a mechanical point of view, since no local cracks through the thickness of a single conduit can be tolerated. For this reason, the jackets in the standard CICC as assumed in the reference DEMO CS design [27] are significantly oversized compared to the static load case, which reduces the current density in the coil and thereby the ability of the CS to generate high flux. In future developments, alternative CS conductor designs will be tested, in which the two main functions of the CICC jacket (structural support and helium containment) are decoupled [25]. Figure 9 shows a sketch of one such proposal. The cable space is solder-filled and indirectly cooled by a separate copper pipe. In this case, penetration of a crack through the steel jacket wall is acceptable, because the helium coolant remains contained inside a separate conduit, made of a more plastic metal.

At coil level, due to the higher mechanical forces, the thickness of the DEMO TF casing is up to 510 mm, as compared to ~200 mm in ITER. A dedicated study of the TF case manufacturing and the casing procedure (winding pack insertion, case closing, etc) was carried out by the company SIMIC [37]. It evidenced that simply extrapolating the ITER manufacturing

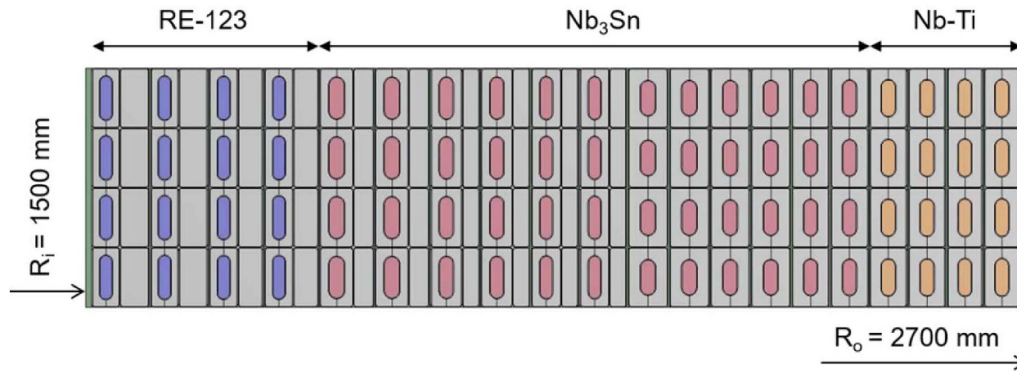


Figure 10. DEMO CS advanced hybrid option (HTS-Nb₃Sn-NbTi). Radial slice showing four rows of conductors of the CS1 winding pack (tokamak center is left). Superconductor and stainless steel (SC + SS) graded design for maximum flux-generating capability.

process would not be possible, one reason being the depth of the closing weld for the case, which is too large for narrow-gap TIG welding. This calls for an entirely different casing procedure, and for splitting the case assembly into a larger number of units because of the limitation of the forging volume. The overall duration for TF case manufacture and casing has been estimated to be 30 years²⁰, at a cost of €1.5 B. Under these premises, which do not align with the DEMO schedule and cost, a large effort will be devoted to identifying alternative solutions for the TF case and casing procedure.

At the magnet system level, the coil structures and supports will be designed in detail during the CDP, and feasibility studies will be assigned to industrial companies to assess all relevant issues, e.g. in terms of manufacturing tolerances and assembly procedures.

Full performance of conductors and electrical insulation are required throughout the lifetime of the power plant. The degradation of conductor performance under mechanical and thermal cycles, as observed in several ITER TF conductors [38], is a major issue for a machine expected to operate efficiently for tens of years with high availability. In the DEMO conductor design, the issue of degradation was successfully addressed using rectangular samples with a low void fraction and an increased stiffness of the cable. Two sets of samples with an Nb₃Sn superconductor have been manufactured and tested: one with react-and-wind technology [39], the other using the wind-and-react approach [40]. For both types, no degradation was observed after 1000 electromagnetic cycles and various thermal cycles.

The long-term integrity of the electrical insulation is of paramount importance as well, because magnets need to withstand high voltages of up to 20 kV across the coil. Developing robust insulation concepts and techniques for all critical areas, in particular insulation discontinuities and penetrations²¹, is a topical subject of the R&D program of DEMO

in the CDP. In order to mitigate the risk of coil degradation due to the high voltage, the feasibility of very high current conductors (~ 100 kA) is being investigated. As fewer turns are required, these allow a reduction of the coil inductance and thereby the maximum operating voltage. More detail is provided in section 8 of the present article. However, due to higher mechanical loads, demonstrating that conductor cyclic performance degradation remains negligible will be challenging.

R&D on innovative conductor and winding pack designs is a core activity of magnet development in the DEMO CDP. The main aim for advanced concepts for DEMO TF and CS coils is to improve their efficiency, i.e. to increase the achievable field or flux, while minimizing the cost and/or required space. The strategy adopted relies on winding the conductor in layers²² instead of pancakes like in ITER. In fact, this approach allows *grading* of both superconductor and steel cross-sections in the different layers based on the local conditions. For the CS design, a hybrid solution has been proposed, with the high-field module made of REBCO conductors, the intermediate-field section with react-and-wind Nb₃Sn conductors, and the low-field module made of NbTi conductors, as shown in figure 10. The advantages and drawbacks of using REBCO and react-and-wind Nb₃Sn conductors are summarized in sections 7 and 12 of this article. A major challenge for the EU DEMO project is demonstrating the feasibility of such conductors at industrial level, including non-destructive tests and quality assurance activities. The second step will be to check the electrical performance of short samples under operational cycles and driven quench events. Finally, the manufacture of 50 m long conductors, subsequently wound to insert coils and tested, will validate the innovative technologies in view of the final design selection. It is worth noting that these activities also provide the *building blocks* for HTS coils other than the CS. Independent of the final choice, the 7 year work plan of the EU DEMO project will drive the maturation of innovative technologies that will certainly be useful for future power plants.

²⁰ This is a worst case assuming the TF winding packs are available at time zero, but with a single supplier and no duplication of equipment. Assuming two suppliers with full duplication of equipment and manpower (higher cost), the duration is reduced to 12 years.

²¹ For example, for instrumentation, helium piping or joints.

²² Extended in the vertical (CS) or toroidal direction (TF), respectively.

Concluding remarks

In the conceptual design phase of DEMO, two alternative concepts, one advanced and one ITER-like, are proposed for each coil system (TF, PF and CS). These will further be developed based on previous achievements and supported by an extensive and targeted R&D program. Prior to the end of this phase in 2027, one concept per coil system will be selected to enter the plant baseline design. The major targets are very high reliability and overall cost reduction as well as tackling specific challenges for magnets in DEMO. These mostly result from the required machine size (see above) and the high plant availability²³. The development plan outlined above addresses the resulting engineering implications, while a certain conservatism lies in the fact that although R&D progress is of course

required, no *breakthroughs* in any field are relied on. Nevertheless, if unforeseen leaps in technology or physics are made, they may be integrated and lead to a higher performance of DEMO or reduced cost or both.

Acknowledgments

The EU DEMO magnets team consists of more than 50 members from research units all over Europe. The design presented here illustrates only a small part of the great progress achieved by this team, which is highly acknowledged.

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²³ The assumed availability leads to an accumulated operational time of ~ 7 full-power years (plasma at full rated power).

4. Commercial fusion power: a killer app for HTS

Charlie Sanabria, Michael Segal and Brandon Sorbom

Commonwealth Fusion Systems, Cambridge, MA 02139,
United States of America

Status

From the 1960s to the 1990s, the triple product of tokamaks increased by four orders of magnitude—faster than the transistor density scaling famously described by Moore's law. This improvement in fusion's key figure of merit was significantly driven by advances in conductor technology like Bitter plates, multifilamentary superconductors and CICC's [41]. By the 1990s, improvements in the triple product stalled, as copper magnets ran into resistive power-loss limits, and LTS critical field limits necessitated ever-larger and more expensive devices [5, 42, 43]. This was true for both tokamak and non-tokamak devices [44].

The advent of the commercially available REBCO HTS capable of maintaining large critical currents at high magnetic fields is breaking this stalemate. REBCO is particularly impactful because the triple product scales with the cube of the on-axis field, allowing for energy break-even in compact devices [45]. Beyond the high-field capabilities, REBCO enables:

- (a) Higher operating temperatures, allowing for greater temperature margins and cryostability [46, 47].
- (b) Ease of manufacturability of cables and joints, with no high-temperature activation heat treatment required [48–51].
- (c) Predictable cable and joint performance, without the variability seen in most LTS conductors. It was recently shown [47] that the critical current of a twisted-stack cable can be predicted from the individual characteristics of commercially purchased REBCO tape reels.
- (d) Excellent cyclic performance, for example through monolithic conductor designs [47].
- (e) Demountable joints that greatly improve the RAMI of devices, including tokamaks [7, 51].

Today, at least three private fusion companies are leveraging these benefits across multiple device classes. They include standard aspect ratio tokamaks [52], low aspect ratio, or spherical, tokamaks [53] and stellarators [54, 55]. To date, these companies have attracted over \$300 M in private capital investment, allowing them to aim for the aggressive timelines required by the climate crisis [56].

This private sector support is in turn defining a new set of public–private roles with governmental organizations (including the US National Laboratories and the United Kingdom Atomic Energy Authority (UKAEA)) focusing on the basic science of plasmas and materials, and private companies are taking on commercial fusion technology challenges including

the design and manufacture of HTS magnets. These complementary roles are being reinforced by new government programs that directly support private fusion companies [57–60] or encourage innovative fusion concepts supported by private companies. For example, the axisymmetric mirror experiment at the University of Wisconsin is enabled by a REBCO magnet produced by a commercial partner [61].

Current and future challenges

In 2017, Commonwealth Fusion Systems (CFS) presented REBCO manufacturers with an aggressive specification of $J_e > 700 \text{ A mm}^{-2}$ (20 K, 20 T, worst angle), and having piece-length and uniformity that satisfied all the present magnet designs proposed by CFS. By 2018, several suppliers had improved their performance to meet this spec, and by mid-2019 a large order totaling $\sim 500 \text{ km}$ was placed with several suppliers. Over the next year, this order was delivered on time while exceeding the required J_e specification. CFS is now placing even larger orders for REBCO tape to be used in the SPARC break-even tokamak (described below).

This exercise proves two important points: first, that present-day REBCO performance is sufficient for next-generation fusion devices, and that REBCO manufacturers are capable of scaling up the performance and volume in the face of commercial demand. There are, however, several critical challenges remaining for REBCO tape to make fusion its killer app:

Tape cost. Today, REBCO production is reaching volumes comparable to the production of NbTi in the 1980s, shortly before it became a commodity [62]. The impact on cost is starting to materialize [63]. However, significant further reductions in tape cost will be required before widespread application to commercial fusion can be achieved.

The HTS field has long hypothesized that large volumes will be able to drive tape prices down through economies of scale. This claim has been historically proven in technologically adjacent industries such as thin-film solar [64], but has yet to be substantiated for HTS. The next challenge the industry faces is to scale volumes up further to the 1000s of km yr^{-1} , where the cost is expected to decrease to the levels necessary for economical fusion power plants and other applications. This scale-up will likely involve a shifted R&D emphasis away from price improvement through further I_c optimization, which is already offering diminishing returns, and toward novel volume production technologies.

Quench detection. The high electromagnetic noise and complex coupled inductances among magnets in tokamaks complicate the reliable detection of voltage rises indicative of a magnet quench. The use of special voltage tap routing and complex signal processing has yielded reasonable quench detection times in LTS tokamaks [65, 66]. However, quench detection in REBCO-based tokamaks will likely have to be twice as fast as in their LTS counterparts—which may

Table 5. Risk retirement approach used in the ARC roadmap.

Risk → Risk retirement device ↓	Tape cost	AC losses	Demountability	Quench detection	High-field magnet integration and production	Structures and cryogenics
TF model coil				×		×
CS model coil		×		×		×
SPARC	×			×	×	
ARC demonstration plant	×		×		×	

render voltage taps ineffective unless complemented by new technologies such as fiber-optics [67–69] and acoustic sensors [70, 71]. The development of these technologies is therefore a priority for both the fusion community and other communities that rely on high-field magnets, such as high-energy physics.

AC losses. REBCO magnets operating at 20 K have a higher AC loss tolerance thanks to the improved heat capacity that most materials exhibit at this temperature. Nonetheless, AC losses still represent a significant challenge for REBCO coils on three levels: the tape level [72, 73], the stabilizer level [47, 74] and the structure level. At the tape level, encouraging efforts are being made towards filamentization of REBCO [75–77]; however, these processes have yet to prove economically viable and scalable. At the cable stabilizer level, it is likely that conductor geometries will require optimization in order to enable fast ramping magnets such as CS and PF coils in tokamaks. Novel concepts for low-loss, high-current cables have been proposed for tape-based cables [78–80]. However, the engineering critical current and cooling capacity required for compact pulsed fusion devices have not been met. At the structural level, special manufacturing processes and composite materials may help reduce eddy current loops within and between supporting structures, and therefore heat loads. These and other solutions will require magnet system integration work to truly be tested.

Demountability. Demountability has been pursued by MIT since the early 2000s [46, 51]. Demountability promises to improve power plant operational costs through improved RAMI [7]. REBCO conductors are well suited for demountability because they have high thermal stability and do not require post-winding activation heat treatment. Reproducible, low-resistance, demountable REBCO joints that are mechanically stable at high fields have already been demonstrated [47]. However, basic engineering design questions on issues such as coolant routing, cyclic endurance and remote handling remain open.

Highfield magnet integration and production. To realize its potential impact on fusion devices, HTS magnets need to be scaled to high fields (>20 T on-coil) and large bores (>1 m). This poses several structural and integration challenges, including the managing of large $I \times B$ forces, tape QA, cryogenics and manufacturing. To realize their potential impact on climate change, these innovations then need to be

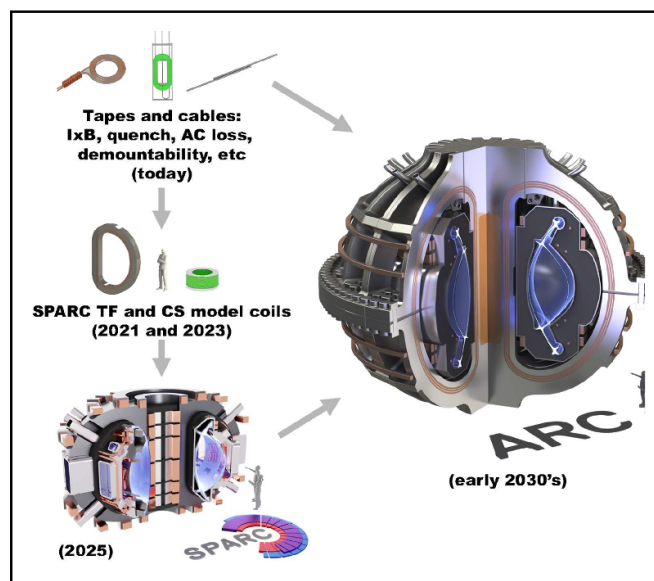


Figure 11. The CFS roadmap to commercial fusion power. Image adapted from renderings by T. Henderson under CC BY-SA 4.0 license and A. Creely (CFS/MIT-PSFC). Image under CC BY-SA 4.0 license.

scaled to production volumes exceeding thousands of magnets per year, with a correspondingly accelerated innovation cycle. This mass production has only been observed in the magnet community at low fields in the MRI industry, and at moderate fields in large research projects like the Large Hadron Collider (LHC)—where temporary mass production capabilities wind down once a project enters its operation phase. The continual, high-volume mass production of high-field magnets is a new challenge that will require new materials, fabrication, integration and testing methods.

Advances in science and technology to meet challenges

CFS and MIT are aggressively pursuing solutions to each of the challenges identified above. The two organizations are jointly developing a high-field compact tokamak called SPARC, which is due to begin construction in 2021 and aims to achieve fusion energy break-even. Using the same conservative plasma physics basis used to design ITER, SPARC will use high-field HTS magnets built with today's commercially available REBCO tape, and achieve an on-axis magnetic field

of 12.2 T with a major radius of 1.85 m [23]. CFS will follow SPARC with ARC, a commercial fusion pilot plant, to be commissioned in the early 2030s.

In order to meet this aggressive timeline, CFS is employing a risk-retirement approach, according to which the identification and addressing of system-wide risks is prioritized over comprehensive scientific understanding or complete sequential subsystem builds. Table 5 shows a sample of significant CFS milestones together with the risks each will retire.

The TF model coil will be a high-field, DC, large-bore magnet, while the CS model coil will be a high-field, fast-ramping magnet. SPARC, for which tape orders are already being placed, is challenging the industry to improve costs and volumes, and ARC will implement demountable magnets so that its vacuum vessel can be periodically replaced. The design, construction and operation of SPARC itself will be a means to retire certain outstanding risks of an ARC demonstration power plant (figure 11). Additional risk retirement tests address the cyclic loading of cables, quench technology and AC loss characterization, and are being carried out both in-house and in collaboration with the US National Laboratories and universities and private companies around the world. These collaborations benefit from growing domestic and international governmental support, and both CFS and MIT continue to actively pursue new collaborative work.

Concluding remarks

The HTS industry is experiencing a rapid scale-up in volume because of demand from fusion applications. The performance of HTS conductors is already sufficient to enable a break-even fusion device, SPARC, which is due to begin construction in 2021. The major challenges related to HTS conductors on the pathway to widespread commercial fusion are cost, quench, AC losses, demountability and magnet integration/production, approximately in order from most to least critical. If these and other challenges are solved, fusion will become the killer app for REBCO, much like MRI is for NbTi, but at the considerably larger scales involved in the \$2 tr global energy market. This will enable a virtuous cycle of new applications and new science for magnet technology and plasma physics. In the near term, milestones on the high-field fusion pathway will include significant new magnet builds in the private sector aided by government and educational institutions. These and future innovations will draw on accelerating interest in commercial fusion from both the private and public spheres, and leverage public—private partnerships that combine the experience of National Laboratories, educational institutions and international research organizations, with the speed and industrial capability of private companies.

5. REBCO magnet technology—a key enabling technology for compact fusion devices

Robert Slade, Greg Brittles and Rod Bateman

Tokamak Energy Ltd, Oxfordshire OX14 4SD, United Kingdom

Status of HTS tokamaks

Spherical tokamaks (STs) offer significant advantages for commercial fusion power plants: higher thermal power per unit plasma volume and significant bootstrap current [81]. These benefits enable smaller, more efficient machines to be developed, accelerating development timescales and reducing recycled power. Progress in understanding the physics of STs is continuing around the world on experimental devices such as MAST [82], NSTX [83] and ST40 [84], which all use pulsed resistive magnets.

A commercial power plant requires superconducting magnets for long-pulse or continuous operation and to maximize net electrical power generation. This has represented a roadblock for STs because the slim center column of the TF magnet results in a peak field on the conductor beyond the capability of the conventional LTS. The recent commercial availability of high-performance REBCO-coated conductors ('tapes') from multiple suppliers makes a high-field ST, with a mission to demonstrate net power gain ($Q > 1$) using D-T fuel, feasible at smaller scale than a conventional aspect ratio tokamak using LTS. A 1.4 m major radius HTS ST with 4 T field on axis can achieve this mission if an adequately thick neutron shield (>25 cm) can be implemented.

REBCO cable studies for fusion applications generally assume CICC construction, with twisted and/or transposed strands [85, 86]. This typically results in a winding pack current density (J_{wp}) less than 100 A mm^{-2} . Figure 12 depicts a segment of the center column in the cross-section for the exemplar 1.4 m ST, comparing two cable schemes: (a) CIC with $J_{wp} \sim 75 \text{ A mm}^{-2}$, and (b) a simple stack of pancake coils wound with multiple tapes-in-hand, with $J_{wp} \sim 350 \text{ A mm}^{-2}$. Figure 13 compares the cable construction in more detail.

The thinner neutron shield of (a) leads to substantially higher in-pulse nuclear heating to the HTS. The thicker shield of (b) means that conduction cooling to a supercritical He flow annulus is feasible and the risk of neutron-induced degradation of critical current is also substantially alleviated [87, 88].

Current and future challenges

Tokamak energy (TE) is developing magnet technology using pancake coils wound from simple stacked HTS tapes, without twisting or transposition (figure 13(b)) and solder-impregnated. Indirect cooling is provided by supercritical helium flowing in channels outside the coil pack. Stress management is simplified by the absence of internal cooling channels and the requirement for a high-strength jacket, with poor

thermal conductivity, around each turn is eliminated. This coil structure minimizes the use of materials with poor thermal conductivity, such as insulation and epoxy potting.

The quench protection benefits of no-insulation (NI) and partial insulation (PI) coils have been widely reported [e.g. 89]. A small solder-impregnated NI pancake stack tested by TE in 2019 achieved stable quench-safe operation above 24 T peak field on coil at 21 K, with an average J_{wp} over 700 A mm^{-2} [90]. It is believed that this technology can be scaled up using a novel turn-turn insulation combining high thermal conductivity with the ability to choose the resistance between turns, thus retaining a high J_{wp} without compromising quench protection. This would satisfy the TF center column conductor requirements for a net fusion gain compact ST.

The use of cables without twisting or transposition in fusion-scale magnets is perhaps controversial. These features have been carried over from LTS fusion cables, nominally to minimize AC loss and ensure equal current sharing between tapes [85, 86]. However, the large size of coated REBCO conductors means that twist pitches are long, and loss reduction is minimal in practice. Conversely, the increased thermal stability provided by operation at higher temperatures suggests that stable operation of large coils without twisting or transposition is feasible [91]. The simple stacked tape design choice also enables 3–5 times higher critical current to be achieved by better aligning the REBCO *ab*-plane with the local magnetic field vector; this is possible in the TF center post. Combined with conduction cooling, the elimination of additional structural material in the cable, and significant reduction in the quantity of normal metal stabilizer for quench protection (as explained below), the target J_{wp} of 350 A mm^{-2} for the TF magnet can be achieved.

Advances in magnet technology to meet fusion scale-up challenges

The critical current of a REBCO tape can be degraded in several ways: (a) during tape and coil manufacturing, (b) by operational damage (e.g. strain/fatigue), or (c) by radiation damage. NI and PI coils are damage-tolerant because current can bypass defects in the superconductor by sharing both between tapes in the affected turn, and between adjacent turns. In contrast, a fully insulated coil with the same normal metal fraction could quickly burn out. For this reason, an additional normal metal stabilizer, typically copper, is added to the cable to slow the rate of temperature rise. NI and PI technology allows a significant reduction, or even complete elimination, of this additional stabilizer. In the event of a local hotspot developing into a thermal runaway, such that all the tapes in the cable (turn) become normal, current can rapidly transfer to adjacent turns before the temperature of the hotspot becomes too high. The ensuing propagation of the normal zone superficially resembles the quench behavior of LTS magnets. Working initially in collaboration with researchers at CERN [92], TE

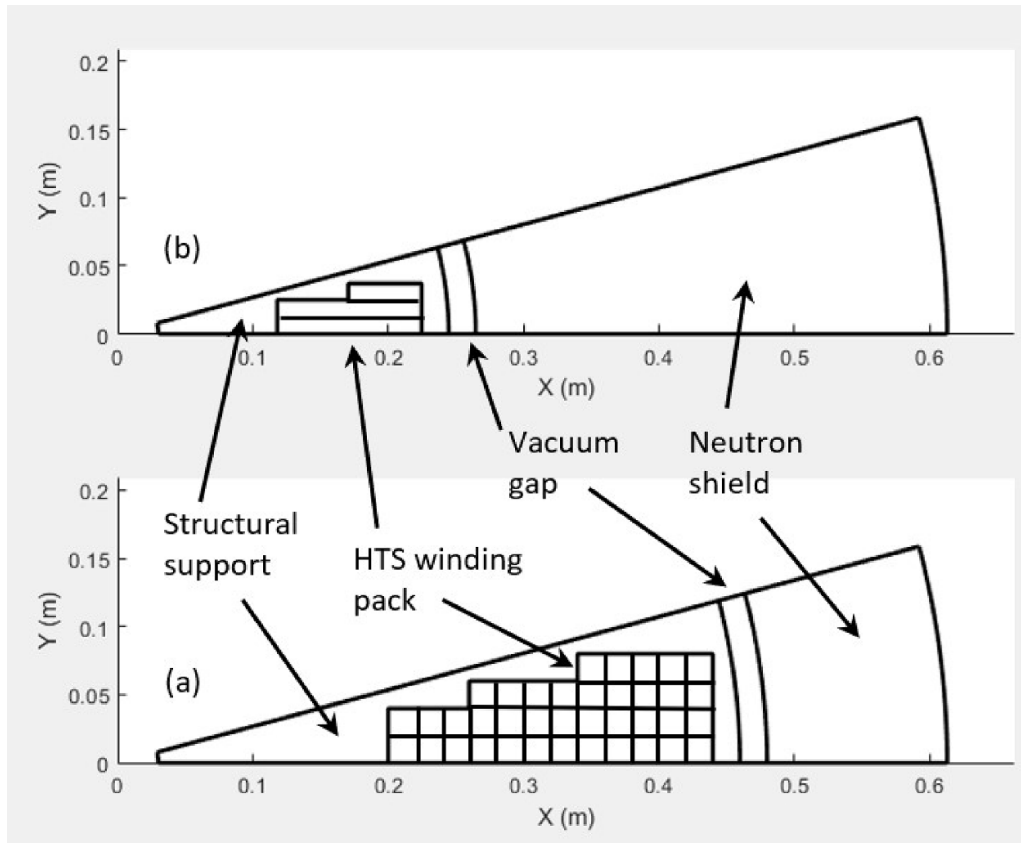


Figure 12. Centre column cross-section showing one half of one TF limb of a 1.4 m major radius 12-limb, 4 T ST with a net energy gain mission. Bottom figure (a) assumes CIC and top figure (b) assumes simple pancake coils. The cable details are shown in figure 13. The approximate areas allocated to the neutron shield, cryogenic insulation, structure and the TF magnet inboard coil packs are shown.

have developed detailed 3D thermal-electric network modeling tools to understand quench in NI and PI magnets, including the effect of screening (magnetization), coupling and induced eddy currents in all superconducting and resistive structures.

The frequently cited disadvantage arising from allowing conduction between turns in a large magnet is the increased charging time. An NI or PI coil can simplistically be modeled as a superconducting inductor (the spiral REBCO path) in parallel with a resistor (the normal metal radial path between turns). The resultant charging time scales approximately as the fourth power of the linear dimension of the coil, and would become impractically long if the TF magnet of a

fusion-scale coil were uninsulated. However, TE is developing a novel partial insulation technology that allows the resistance between turns to be chosen over a wide range, providing a compromise between charging delay, quench robustness and coil dump characteristics that satisfies the requirements of fusion-scale magnet protection. Depending on the chosen stabilizer content and turn-turn resistance, it is possible to make the magnet extremely resilient to transient energy inputs (e.g. caused by a disruption) and to tune the fraction of stored magnetic energy that is dumped into the cold mass and coupled structures, or to an external dump varistor.

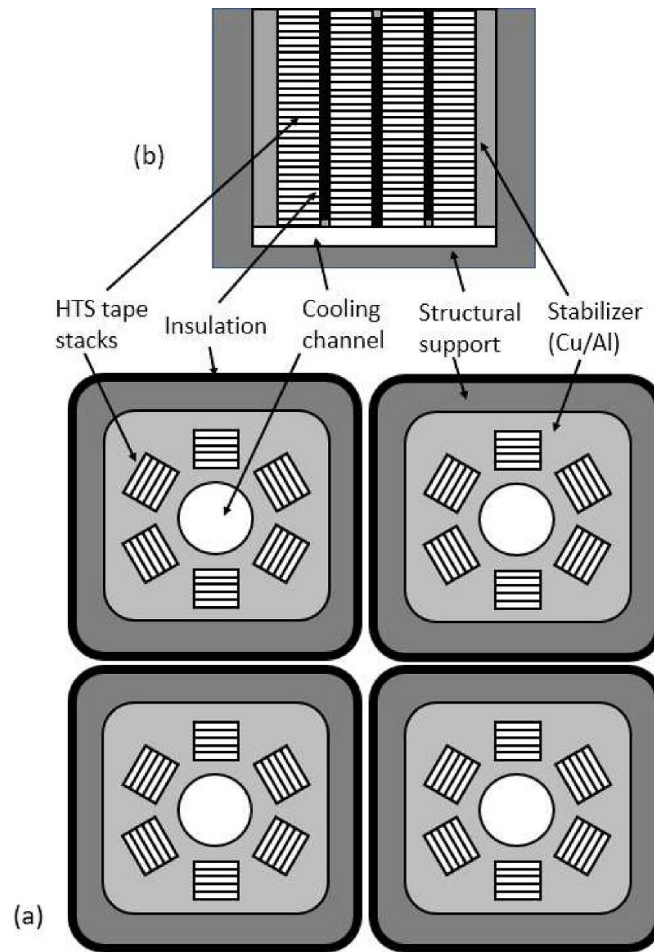


Figure 13. Detail of the REBCO coil packs for the center column cross-section layouts shown in figure 12; bottom (a) is for typical CIC (twisted but NOT transposed), and top (b) is simple pancake coil construction. Both layouts contain the same number of HTS tapes.

Concluding remarks

Significant progress has been made developing high-field magnet technology that plays to the strengths of the REBCO HTS. A change of mindset is necessary when thinking about superconductivity at temperatures above the conventional liquid helium regime. An increased heat capacity means that, with an appropriate design, and in contrast to LTS, stable operation is possible even when the current is shared between the normal and superconducting regions. This behavior is

exploited using simple cable and coil structures that combine high thermal conductivity and tailored, anisotropic electrical resistivity. Thorough analysis of the novel coil structures under all operating conditions (energization, steady-state operation, pulsed nuclear heating, AC loss and plasma interactions, and quench protection scenarios) has been carried out using a novel 3D thermal-electric element network model. This patent-pending technology will be tested in a medium-scale high-field REBCO toroid, including fast-ramping REBCO PF coils, currently under construction at TE.

6. Nb₃Sn wire development for fusion beyond ITER

Yasuyuki Miyoshi¹, Nobuya Banno² and Kazuyoshi Saito³

¹ ITER Organization, 13067 St. Paul-lez-Durance, France

² National Institute for Materials Science (NIMS), Tsukuba 305-0047, Japan

³ Japan Superconductor Technology Inc. (JASTEC), 651-2271 Kobe, Japan

Status and future challenges of Nb₃Sn wires for fusion

ITER has entered into its construction phase and gigantic Nb₃Sn magnets are traveling across the globe. The successors of ITER towards fusion power generation are DEMO reactors, for which conceptual designs are currently being developed [5, 93]. Faced with limits such as the power management of diverters and stress tolerance of stainless steel, their design is likely to be an extrapolation of ITER and with Nb₃Sn conductors. In short, the DEMO magnets will increase in size and the conductors will carry a higher current under a higher electromagnetic load (cf DEMO TF conductor design current 83 kA in 13.7 T under 800 MPa stress compared to ITER TF conductor current 68 kA in 11.8 T under 670 MPa [93]).

A review of the Nb₃Sn CICC shows a possible route to a DEMO conductor by strain management [94]. What is clear is that the compressive strain applied to the wires by the conductor jacket degrades the performance. For the ITER CICC, the wires are under thermal strain in the range -0.7% to -0.5% for TF conductors. By employing the react-and-wind method, i.e. reacting the wires to form a superconducting phase before jacketing and winding, this thermal strain could be reduced to -0.3% . One conductor design study for EU DEMO utilized the react-and-wind method, and showed that the reduced thermal strain as a result corresponded to the doubling of Nb₃Sn performance [95]. The choice of jacket material also influences the strand thermal strain [96].

Another route is improving the performance of Nb₃Sn wire itself, but there has been a lapse of a decade since the last R&D for ITER. For the bronze process, the ITER bronze wire is close to the Sn content limit of a workable bronze. For the internal-tin process, restacked-rod-process (RRP[®]) wires, developed for high-energy physics, hold the record J_c - B performance, non-Cu J_c 3000 A mm⁻² at 12 T, 4.2 K. The values are three and two times those of bronze and internal-tin wires for ITER, respectively. A key feature of RRP wire is the greater quantity of Nb₃Sn achieved by sacrificing the Cu fraction. However, it comes with a cost of extensive filament bridging over ~ 50 μ m. This would be unsuitable for a tokamak with CS-based operations that requires low hysteresis loss. Consequently, no wire architecture at present can satisfy both the high J_c and low hysteresis loss required for DEMO. The need for R&D to cover the lack of performance is illustrated in figure 14. Here, we identify some of the recent developments in materials science that are relevant for the development of next-generation Nb₃Sn wire for fusion.

Advances in science and technology to meet challenges

Firstly, on the mechanical stress aspect, the wires themselves can be reinforced. It is common practice to have inclusions of reinforcing materials in wires of commercial magnets that operate under high-stress conditions, e.g. high-field NMR magnets. Such wires typically have a Ta core in the matrix, which increases the Young's modulus of a bronze process wire from ~ 100 GPa up to ~ 170 GPa, and the 0.2% yield strength from ~ 17 MPa to ~ 300 MPa. In CICC, the wires are also subject to multiple bending stress. For reinforcement against bending, a ring of multifilamentary Cu-Nb composite just outside the barrier around the Nb₃Sn filamentary region has been developed [97]. Such reinforcement increases the Young's modulus up to 190 GPa and the 0.2% yield strength up to 350 MPa.

The Cu-Nb reinforcement exerts strain on Nb₃Sn filaments, similarly to the effect of jackets in CICC, which suppresses I_c . Some of this strain can be released by a process of bending a wire or a cable repeatedly at room temperature after the heat treatment—the so-called pre-bending process. Pre-bending changes the strain state of Nb₃Sn filaments, and can enhance I_c as well as shift the peak of I_c strain characteristics that can be used to optimize the conductor performance under bending strain. The method has been successfully applied with the react-and-wind Nb₃Sn coils wound from Cu-Nb reinforced wires for the 25 T cryogen-free magnet at Tohoku University. Such reinforcement and strain management of the wire could be relevant for the highstress tolerance required for DEMO conductors.

Alloying the wire matrix material with an element is another way of improving the wire mechanical strength. One candidate element is Zn. It comes with an advantage that added Zn, homogeneously present in the matrix, enhances Sn activity during the reaction to form Nb₃Sn. The alloying can be done for both the bronze process and internal-tin process, i.e. as an enhancement of the already well-established Nb₃Sn wire. For the bronze process, it will be an optimization to balance the concentrations of Zn and Sn. Recently, Tachikawa *et al* proposed the brass process, which is an internal-tin process with brass as the matrix material instead of copper. A detailed report of the brass process and its most recent developments can be found in [98].

Key findings of the brass process are in Sn diffusion during a three-step heat treatment. In the initial reaction at 215 °C, there is a notable absence of the porous ε phase as Sn starts to diffuse into the matrix. Then, at 400 °C, a dense β -CuZn phase forms at the diffusion front that suppresses the voids before the final reaction at 550 °C. In other words, Zn enhances Sn diffusion to Nb filaments by suppressing the formation of the porous phase and voids at the diffusion front. This would be a new tool for enhancing the J_c - B performance. Since Zn does not dissolve in Nb₃Sn, it is compatible with other types of performance enhancements, such as Ti doping through the matrix and filaments that increase the amount of fine-grain Nb₃Sn [98].

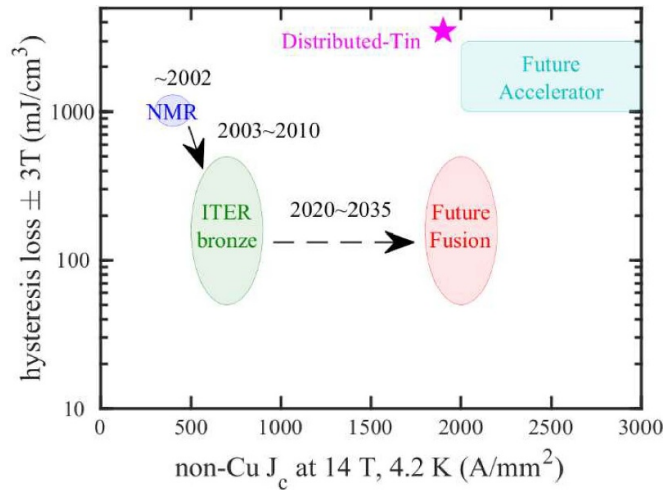


Figure 14. Illustration of wire hysteresis loss with applied field ± 3 T plotted against non-Cu J_c at 14 T, 4.2 K. The accelerator applications drive for higher J_c , but ITER-like fusion applications require low hysteresis loss.

Another relevant development in the wire architecture is an adaptation of the internal-tin process, called the distributed-tin (DT) process. In brief, the wire consists of modules with a single Sn core in copper matrix that are surrounded by several modules of multifilamentary Nb in copper matrix. In this way, the filamentary area is increased while the filament linkages are limited within a module. The architecture has existed since the 80s and it has recently received renewed interest for use in high-field NMR magnets. Meanwhile, CERN is currently leading a global effort for the development of Nb₃Sn wires for the High-Energy LHC or Future Circular Collider. These wires have an internal-tin architecture with a distributed barrier or distributed tin, and are developed with a strong emphasis on J_c . The prototype wires have achieved a non-Cu $J_c \sim 1800$ A mm⁻² at 14 T, 4.2 K and ~ 1000 A mm⁻² at 16 T, 4.2 K [99], which are comparable to the High-Luminosity LHC specification.

In a DT wire, a key parameter is the Sn diffusion distance. There is a gradient of Sn concentration that results in Nb₃Sn grain size variation from fine grains to coarse grains and unreacted cores from the periphery to the module center. A simple reduction of dimension by drawing down the same wire showed clear improvement of the J_c - B performance [100] due to the shortened Sn diffusion distance. Importantly, this indicates that the module size reduction is not limited by the barriers as in the distributed barrier wires. So far, prototype wires with modules ~ 30 μ m in size have been manufactured. The characteristics of these wires are compared with the relevant wires in figure 15. The highest non-Cu J_c of ~ 1100 A mm⁻² has been achieved. As for the industrial

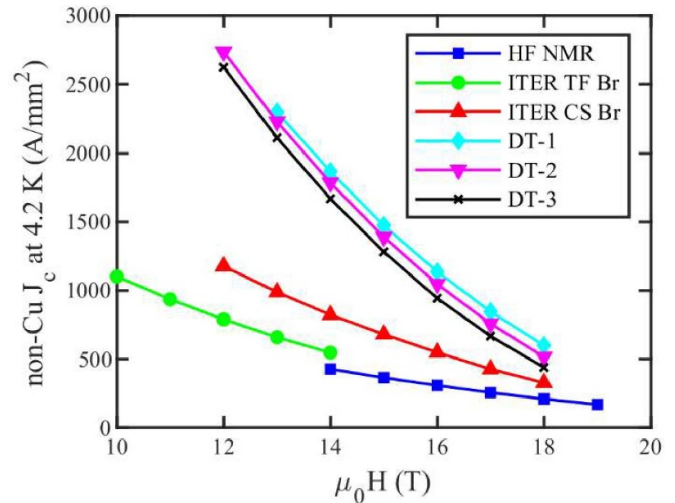


Figure 15. Non-Cu J_c at 4.2 K as a function of applied magnetic field for three prototype DT wires (DT-1, -2, -3) compared against high-field NMR bronze wire and ITER bronze wires for TF and CS coils.

scalability, a piece length of 900 m has been successfully manufactured. For a first trial, the brass process has been applied to a DT wire, which showed enhanced Sn diffusion and suppressed void formation [101].

The recent high J_c wire development is focused on increasing the Nb₃Sn filamentary area, but the extent of filament bridging means the hysteresis loss is much greater than that of ITER wires (figure 14). Since J_c is now approaching a DEMO-relevant level, a directed research effort is needed to reduce the hysteresis loss. While balancing J_c and the hysteresis loss, we may take advantage of a recent finding that the grain size is reduced by alloying individual Nb filaments with Ta and Hf [102]. As the method does not restrict itself to any specific wire architecture, it can be used to enhance the performance of candidate Nb₃Sn wires for fusion.

Concluding remarks

We have reviewed above what can be the starting materials for Nb₃Sn wire development for fusion, which requires enhancement in mechanical properties and J_c - B characteristics, while keeping the hysteresis loss small. With the DEMO engineering design phase starting in 15 years' time, it is time to start in earnest the development of Nb₃Sn wires for the next fusion.

Disclaimer

The views and opinions expressed herein do not necessarily reflect those of the ITER organization.

7. REBCO for fusion and fusion for REBCO

Anna Kario and Herman ten Kate

Faculty of Science and Technology, Energy Materials and Systems, University of Twente, 7500 AE Enschede, The Netherlands

Status

Nuclear fusion needs a high magnetic field for plasma confinement only achievable with superconductors. We have observed that ITER, the world's largest tokamak under construction, cannot exist without a NbTi and Nb₃Sn-based magnet system. Parallel to ITER construction, the next-generation DEMO reactor is in the research and development phase, for which the use of a high-temperature REBCO superconductor is naturally considered. The REBCO-coated conductor, with its remarkably high stability and critical current density, is certainly THE dream conductor for fusion magnets for very good reasons, when ignoring cost for the moment. Is this true for all sizes and flavors of fusion magnets? Fusion magnet developments are happening nowadays on two different scales.

Firstly, there is huge-scale technology at low magnetic field (5–6 T in plasma, 12–13 T in coils), where enormous and complex reactors like ITER and the future DEMO are being constructed and developed. In the case of DEMO, REBCO is not the only choice of superconducting material, and unquestionably not for all types of magnets [35]. In the present DEMO program flow, alternative designs using REBCO are ongoing. The application of REBCO would eventually allow a future machine to operate at the 30 K level, taking advantage of the quasi-infinite stability and thus profitable operational reliability and availability for power production; it also avoids the use of relatively costly helium as a coolant at the 4.5 K level and all the associated disadvantages.

Secondly, a compact technology, using the highest possible magnetic field (12–15 T in plasma and 25–30 T in coil) would enable much smaller size machines. In addition, to accommodate the higher nuclear load on the magnets, an operational temperature of 20–30 K may be beneficial [6]. Such compact machines are pursued by the UK-based company TE [103], followed by CFS [104] in the US. Obviously, they cannot be realized without REBCO technology. Figure 16 shows, as an example, a sense of the coil development route at TE, from simple pancake coils stepping up to sizable fusion magnets within a few years.

The crucial question is whether REBCO tape conductors are mature enough, or can become so in the near future to fulfill the requirements. Assuredly, substantial progress is visible in the production of unit lengths and production volume as well as the diversity and competition among manufacturers.

Current and future challenges

The companies SuperPower and AMSC in the US, Bruker and THEVA in Germany, Fujikura in Japan, SuperOx in Russia, Shanghai Superconducting Technology (SST) in China and

SuNAM in Korea are manufacturing REBCO tapes with good-quality unit lengths of 50–300 m [105]. The shortcomings at this stage of tape development toward all-REBCO magnets for fusion are primarily (still) the high cost, quench protection for high-energy coils, high AC loss, mechanical tape weakness in delamination strength at the REBCO–substrate interface, low production yield and short unit lengths, as well as the delivery time. Most of the issues can be solved or engineered around, but the most severe and potential showstoppers in the long term are cost, delamination strength and quench protection.

REBCO-coated conductor development is too focused on increasing the critical current in field and temperature by using different technologies. Less attention is on the mechanical properties, stabilization and uniformity in general. Coated conductors are modestly resistive to transverse tensile stress of 10–100 MPa, but fragile to local delamination; they can take some 20 MPa in shear, but are easy to bend, but only in the so-called easy-bending direction [105]. However, the uniformity of the properties, especially for delamination, is a concern. In addition, the response of REBCO to thermal and high numbers of electromagnetic cycles in magnets is still a field of research. Here, a joint effort of manufacturers and research labs would be beneficial. Some drawbacks of single tapes can be mitigated by using cables for magnets instead. Although the coated conductor is a tape with a high aspect ratio, ways have been found to assemble them in classical as well as novel cable structures based on creating round cables from stacks of straight tapes (stack cable) or by spinning multilayers of tape on a core (CORC® cable), and cutting strands from wide tape in a meandering shape (Roebel cables). Although some variants of such cables have been demonstrated, there is still a need for inventions, new cable concepts and testing of significant demonstrator coils made of these in order to deliver in a convincing way the proof of the technology.

Quench detection and protection are key issues, unsolved for meter-size high-energy REBCO magnets due to the low normal zone propagation in and high enthalpy of the coil windings. This issue may be solved at cable level by introducing embedded quench detection sensors [106–109]. Another route to be developed in full is to apply a controlled resistance between turns in the coil windings optimized for achieving an acceptable ramp time of the magnet although with sufficient quench protection by allowing current sharing among turns.

The development of the REBCO conductor not only occurs at tape level, but also through demonstrator magnets. Pancake and racetrack coils, NMR and MRI demonstrators as well as accelerator-type magnets are being developed. There is still a long way to go and the community would benefit by sharing at best, knowledge, ideas and results. The entire development chain, from material via tape to practical conductors and magnet design and construction, may enjoy a strong collaborative effort, since a successful magnet needs more than just a high critical current density in the tape.

Concerning the too high cost of REBCO tape, a cost reduction by a factor of 2 or 3 may be envisaged by optimists due to scaling up in production volume and improving yield, but the factor of 5 or 10 needed for allowing commercial large-scale applications is hard to imagine. Desperately needed

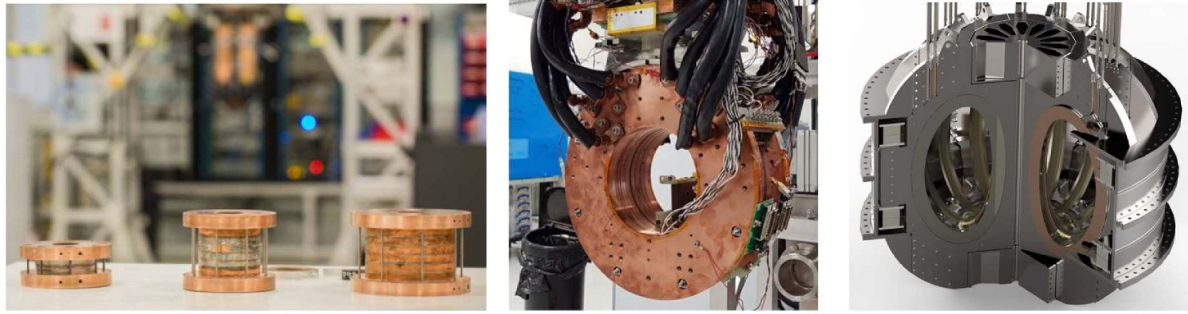


Figure 16. Series of REBCO demonstration coils from TE, demonstrating the onset to full size fusion-relevant REBCO coils; from 1 (left), via 3 (center) to 50 km (right) of tape use, increasing in volume and stored energy, nicely showing the quench protection issue when scaling up to larger coils and already 40 MJ in the DEMO 4 system. Pictures courtesy of TE.

is a much cheaper deposition process still achieving a useful current density. Note that fusion magnets in particular are stress-limited in construction materials rather than in the conductor cross-section. Thus, a much lower current density would be acceptable in tape that can be manufactured at a much lower cost. Many other techniques were already tried, but it would make sense to reopen the case and make a new joint effort to find and develop for production such a cost-efficient conductor. As mentioned previously, many problems relating to coated conductors can be engineered around, but the bottom line is that low-cost REBCO tape is mandatory for enabling large-scale application and true impact on society.

Advances in science and technology to meet challenges

To arrive at a world with reliable REBCO superconducting magnets, a large effort is still needed, even though a remarkable amount of magnets has already been built, as illustrated in figure 17. The tape unit length for larger systems needs to increase with the increasing production yield. The quality control of conductor properties needs another step-up, by reporting not only the critical current versus length, but also geometrical tolerances, mechanical properties and the delamination strength. Non-destructive methods for detecting delamination need to be developed, which will also contribute to a greater understanding and better control of this parameter [110]. Systematic research on cycling loading of REBCO in magnets is another key issue for demonstrating successful application in fusion magnets.

Quench detection and protection methods are also key for a fusion machine that needs to be reliable and available for power production, where long periods of service due to coil quench or even conductor burnout are disastrous. The research in this area raised already a few ideas such as: modification of the geometry of the coated conductor into so-called current flow diverter architecture to achieve a higher normal zone propagation velocity [109]; or using embedded quench detection sensors in the cables for monitoring normal zones along the entire length [106, 107], eventually using optical fiber sensing [108].

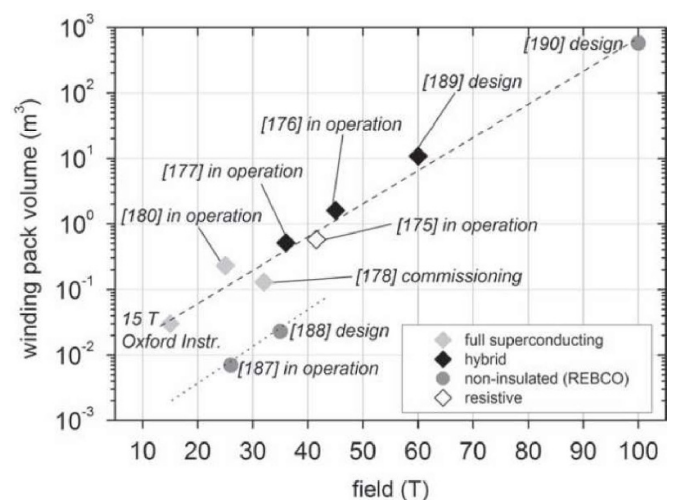


Figure 17. Winding pack volume for various high-field solenoids. Lines are for guiding the eye. Courtesy picture [105].

The two mentioned burdens of delamination and protection that result in dysfunction of the magnet might be solved by the non- or partially insulated coil approach [105]. Non-insulated REBCO coils are being presented as self-protecting due to the low resistance between turns providing different radial and circumferential paths for current and heat distribution after the quench. But such coils are not feasible for fusion application due to their extreme charging times. This can be mitigated by altered inter-turn resistance leading to faster coil charging times. Such approaches have already been investigated; for example, by TE for the development of the demonstrator fusion magnets and by Robinson Research Institute for MRI [111].

Concluding remarks

The REBCO superconductor certainly plays a remarkable role in achieving nuclear fusion, and vice versa, nuclear fusion plays a crucial role for REBCO. The material is not necessary obligatory for next-generation fusion systems such as DEMO and beyond, but it is for sure unavoidable for compact fusion systems. However, the compact fusion enterprises

are investing a lot in our future due to aggressive development and decentralizing potential plant installations, thereby boosting the further development and volume production of REBCO conductors. The development includes unit length, yield, delamination, delivery time, quench protection issues and, very importantly, cost. Due to these advancements in REBCO conductors, compact fusion may be realized, but at the same time by cross-fertilization, REBCO will be taken more seriously and come closer to the break-even point for

other applications. Definitely, we shall start to benefit from the huge advantages of using REBCO over low-temperature superconductors for magnet applications. To mention are the extremely high operational stability; important advantages in cryogenics efficiency and cost when operating in the 20 to 30 K range; as well as less critical mechanical limits, less volume and less weight. All this translates into significantly enhanced reliability and thus availability of the magnet application for their users.

8. Very high current conductors for reduced operating voltage

Pierluigi Bruzzone and Rainer Wesche

Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-5232 Villigen PSI, Switzerland

Status

The fusion magnets of the last generations are cooled by forced flow of helium and surrounded by a vacuum. The vacuum environment sets severe requirements for the electrical insulation, frequently referred to as ‘Paschen tightness’. A small crack or defect in the electrical insulation, which ends up exposing the conductor metal surface to the vacuum even through a long percolation path, leads to an arc (Paschen failure) with catastrophic consequences for the integrity of the magnet [112].

The use of high-grade insulating materials is not effective in preventing Paschen failures, which mostly happen where the application of electrical insulation on conductor/winding meets discontinuities, e.g. helium inlets, voltage taps and structure edges. In many cases, the insulation imperfections become evident only upon cool-down (when pockets of pure resin crack under thermal stress) and/or operating loads. The Paschen test, a high-voltage test for a magnet in low-pressure gas atmosphere, is now routinely applied for the acceptance of fusion magnets. The experience of W7-X, JT60 and ITER [113, 114] points at plenty of Paschen failures during acceptance tests despite the QA procedures enforced during the coil manufacture. The repair of a ground insulation defect before installation of the magnet is usually possible. However, if worsening of the Paschen limits happens in operation [114], the consequences may be severe, mostly in the case of TF coils, whose replacement is prohibitively demanding.

The ultimate mitigation of the issues with Paschen limits is lowering by design the operating voltage in fusion magnets.

Voltage and current for EUROfusion DEMO

In fusion magnets, as in any superconducting magnet with insulated turns, the maximum operating voltage, V_{\max} , is inductive, either at the quench emergency discharge or in pulsed operation, i.e. the voltage is proportional to the self-inductance, L , which is proportional to the square of the number of turns, N_t . For a given magnet, the product of the operating current, I_{op} , and N_t is constant, and so is the stored energy E . Hence, the maximum voltage is inversely proportional to the operating current:

$$V_{\max} = L \cdot (dI_{\text{op}}/dt) \propto N_t^2 \cdot (dI_{\text{op}}/dt) \propto 1/I_{\text{op}},$$

or

$$V_{\max} = 2E/(\tau \cdot I_{\text{op}}),$$

where τ is the discharge time constant.

For the DC operating TF coils, the maximum voltage occurs at the quench emergency discharge. In this case, a lower voltage could also be achieved by applying a slower discharge rate and substantially increasing the copper cross-section to preserve the hot-spot temperature criterion. However, the engineering current density would decrease because of the larger copper content and the size (radial build) of the coil would dramatically increase [115]. For the CS and the PF coils a slower discharge does not mitigate the high voltage issue as the maximum voltage occurs during pulse operation. The most effective way to reduce the voltage in fusion magnets is an increase of the operating current.

The ITER project was pioneering the high-current conductors, with 68 kA for the TF coils proposed in 1995. With the EUROfusion DEMO project, the size of the tokamak is substantially larger than that of ITER [5]. The first measure to reduce the terminal voltage at discharge is to connect each individual TF coil to a pair of current leads, i.e. each coil has its own protection circuit, opposite to ITER, where two coils are series-connected to the protection circuit. Further on, a parametric investigation is carried out in [115] to explore the maximum voltage as a function of the conductor current and discharge time constant, as shown in figure 18. In DEMO TF coils, the discharge time constant should not be shorter than 35 s to limit the electromagnetic loads on the plasma vacuum vessel. According to the green curve ($\tau = 35$ s) in figure 18, an operating current of 118 kA, corresponding to only 126 turns per coil, would be sufficient to limit the terminal voltage below 4 kV, i.e. below ± 2 kV to ground, lower than ITER despite the much larger stored energy. To fulfill the hot-spot criterion of 150 K, to increase the discharge time constant a decreasing value of J_{cu} must be retained in the design; see also [115].

Design aspects for conductors with very high operating current

Besides the reduction in the operating voltage, the increase in the operating current has two more advantages in the design. The smaller number of turns implies a reduction of the cross-section for the electrical insulation at constant thickness of the turn insulation, increasing the engineering current density and reducing the radial build [115]. For the same overall helium cross-section in the winding pack, the force-flow cooling is also positively affected by the reduced number of turns: the cross-section for helium in the conductor increases with the conductor size and the hydraulic length decreases with the number of turns, implying that the residence time of helium decreases and the mass flow rate increases for the same pressure drop.

On the other hand, a few challenges arise with very big conductors. Figure 19 compares an illustration of a 118 kA/12 T DEMO TF react-and-wind conductor (similar design approach to that in [116]) with the 68 kA/11.7 T ITER TF wind-and-react conductor.

The minimum winding radius is crucial for the deformation of a large-size conductor. A rectangular shape helps to reduce the bending strain and plastic deformation compared to

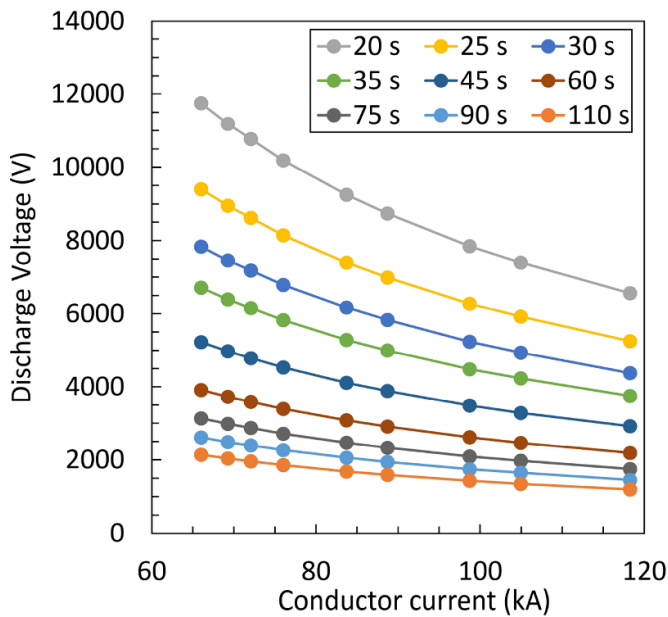


Figure 18. EUROfusion DEMO TF coil. Discharge voltage versus conductor current for various discharge time constants.

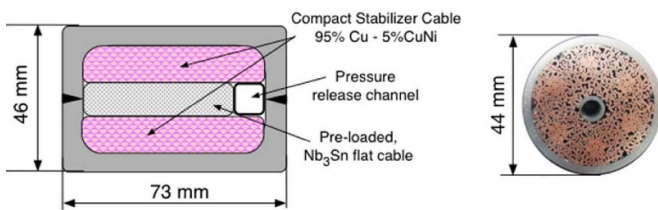


Figure 19. An illustration of the 118 kA EUROfusion DEMO TF conductor (left) and the ITER TF conductor (right).

square and round shapes, where bending at short radius leads to large keystoneing and ovalization. For the 118 kA conductor sketched in figure 19, the radial size is ≈ 46 mm, comparable with ITER CS and PF conductors, which implies acceptably low jacket keystoneing for a winding radius >2.3 m (bending strain $<1\%$). The rectangular shape is anyway the mandatory choice for react-and-wind Nb_3Sn conductors [116].

The large cables for fusion magnets are usually encased in rigid structural armor ('jacket'), which protects the cable from load accumulation from neighboring conductors. The cable, which is mostly made of brittle high-field superconductors (Nb_3Sn and HTS), must withstand only the internal loads, which are proportional to the field B , the cable space current density J and the cable size—for a square cable of size D , the peak load is $B \times J \times D$. The larger the cable, the larger

the loads. For an ITER-like cable-in-conduit, wind-and-react, round with $\approx 30\%$ void fraction, a size increase of 50% would likely lead to unbearable stress/movements. On the other hand, a react-and-wind pre-loaded flat cable as in figure 19 left has better chances of withstanding the large internal loads without damage.

The AC loss, mostly the coupling current loss in the cable, is an obvious concern for large conductors because the cable pitch is proportional to the cable size and the coupling loss is proportional to the square of the pitch. The ITER conductors have introduced steel barriers between the cable elements to cut the large current loops. On the other hand, the use of resistive barriers must be carefully considered as they hinder the current redistribution across the cable and the radial convection of the coolant, jeopardizing the stability.

The path forward

Prototype conductors must be assembled and tested to demonstrate the performance according to the design prediction. The 118 kA DEMO conductor represents a reasonable target. The test range for very high current conductors must also be accounted for. For the largest conductor test facility (SULTAN), 120 kA is the very upper limit. For current above 120 kA an upgrade would be necessary. On the other hand, a small-scale model coil made by a very large conductor is not a reasonable project because of the plastic deformation at small bending radii.

Other magnetic components also need to be upgraded for higher current, e.g. the power converters, the HTS current leads, bus bars and breakers. For the bus leading to the power converters, the large current may make it attractive to use HTS conductors at 77 K, rather than copper or aluminum bars.

Concluding remarks

A reduced operating voltage for fusion magnets is highly desirable to mitigate the manufacturing risks and enhance the overall reliability of a fusion plant. An upgrade of the range of operating current, say up to 120 kA, is the most effective way to reduce the voltage both at quench emergency discharge (TF coils) and in pulsed operation (CS and PF coils). Prototype conductors, based on Nb_3Sn react-and-wind and possibly on HTS technology, are deemed feasible with moderate effort, paying attention to the critical design aspects. Most fusion devices aimed at full-size demonstration of fusion power plants will benefit from superconducting cables with high operating current.

9. Fusion superconducting cables: design criteria

Thierry Schild

ITER Organization, 13067 St. Paul-lez-Durance, France

Status

Superconducting magnets for fusion machines have been developed since the early the 80s up to today with increasing size from about 50 tons of NbTi conductor for Tore Supra or 90 tons of Nb₃Sn conductors for T-15 up to 650 tons of Nb₃Sn strands for ITER. All the fusion machines ever designed are still first-of-a-kind machines as they have never been built more than once. As the magnetic system is usually about one-third of the total tokamak cost, magnets are designed and manufactured with significant margins. The higher the margins are, the higher the system cost. As large fusion devices are worth billions of euros, it is essential to optimize margins to support cost-effective system development.

The superconducting conductor is composed of superconducting material (commonly NbTi and Nb₃Sn), copper and mechanical material (usually stainless steel or reinforced aluminum). The superconductor weight for a given field and operating temperature is driven by the choice of temperature margin, ΔT (the difference between the current sharing temperature and the operating temperature plus the temperature increase due to transient heat loads). In high-energy physics and fusion, ΔT is usually initially set at about one kelvin (ITER, W7-X) or more (KSTAR, EAST). The copper mass is mostly driven by the quench protection with the objective to limit the maximum temperature raised in case of quench in the range of 200 K. The mechanical material mass is driven by the maximum stress allowable in the material. In the case of Nb₃Sn, the mechanical reinforcement also plays a major role in limiting the strain in the superconducting strand as this material is brittle and its transport current capability is strain-sensitive.

For large fusion devices, the optimum conductor layout, which requires a very large current under a very high field (up to 70 kA and 12 T for ITER TF) is the helium forced flow cooling CICC. This is a multistage cable made of superconducting composite copper strands inserted into a mechanical jacket.

Current and future challenges

The most valuable material in the CICC is obviously the superconducting material. It is then interesting to see how this material weight is sensitive to the ΔT value. As an example, the Nb₃Sn ITER-2008 critical current correlation [117] is used to plot the evolution of the superconductor amount required as a function of ΔT (see figure 20). The parameters used to calculate the cable critical current are given in figure 20. The percentage presented in this plot is $S_{T_{cs}-4.5K}/S_{ref}$. S_{ref} ($T_{cs} - 4.5K = 0K$) is the superconductor cross-section needed to have the cable critical current, I_c , at the operating temperature, 4.5 K, equal to the operating current, I_{op} . $S_{T_{cs}-4.5K=1K}$ (137% in figure 20) is the superconductor cross-section needed

to have the cable critical current, I_c , at one kelvin above the operating temperature, 5.5 K, equal to the operating current, I_{op} .

Assuming a classical heat load during plasma operation, the temperature rise is in the range of 0.5 K–1.0 K. This figure shows that the usual margin of 0.5 K–1.0 K raised the superconducting material weight beyond 50% (for ITER TF conductor this margin is 0.7 K) compared to a theoretical no-margin design.

These design criteria are based on the assumption that, above T_{cs} , the coil will quench, and below T_{cs} the coil is superconducting. For medium-scale NbTi CICC with a high n -index (a few 10s), this assumption has been proven on fusion machine model coils such as the W7-X DEMO coil. In [118], we can see that this coil was operated in a steady condition at a few tens of millikelvins below T_{cs} , and then quenched going above this value. For Nb₃Sn CICC with a low n -index (below 10 [16, 119]), the situation is very different as the coil can be operated even above T_{cs} in a kind of semi-resistive mode [16]. In this case, only the cooling power available is limiting the maximum operating temperature. Figure 21 shows the evolution of the heat load per meter in the peak field region as a function of the operating temperature, assuming it is above the current sharing temperature. To plot this figure, $S_{T_{op}-T_{cs}}$, the needed superconductor cross-section is calculated for a temperature below the operating temperature. As an example, $T_{op} - T_{cs} = 1K$ means the cable critical current is equal to the operating current, I_{op} , for a temperature 1 K below the operating temperature. Then, for the superconductor section $S_{T_{op}-T_{cs}}$, the cable critical, $I_c(T_{op} - T_{cs})$, is calculated at the operating temperature. Obviously, this critical current value is lower than the operating current. It is then possible to calculate the heat load per unit length that will depend on the n value according to the classical following formula. E_c is the electrical field criteria:

$$P = E_c I_{op} (I_{op} / I_c (T_{op} - T_{cs}))^n.$$

It can be seen that operating 1 K beyond the current sharing temperature (defined here with an electrical field criteria of $10 \mu V m^{-1}$) increases the heat load by a factor 3 for an index equal to 5 and by a factor of about 10 for an index equal to 10. Depending on the field value along the conductor length, a heat load in the range of 1–10 W m⁻¹ is acceptable for the cryogenic system. Acceptable means it does not lead to thermal runaway.

It should be noted that, for the purpose of the comparison, this calculation has been done assuming that the individual strand critical performance is not dependent on the cable n value. In fact, a very low index is usually the signature of strand degradation. An index equal to 1 would then mean the strands are all resistive, and dissipating unacceptable heat, as only copper would carry the current.

Advances in science and technology to meet challenges

Both temperature margin and electrical field criteria lead to an over estimation of the requested superconducting material

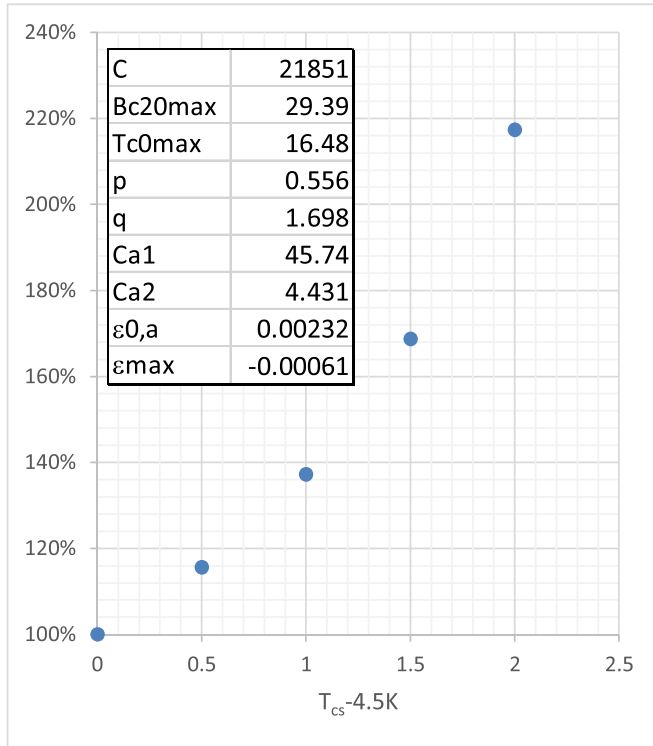


Figure 20. Percentage of Nb₃Sn needed in a conductor as a function of the temperature margin. 100% is the material needed to carry the nominal current at 4.5K without any margin. Peak field is 12 T in this case.

quantity for low index Nb₃Sn CICC because these conductors can be operated beyond the T_{cs} threshold. This is likely to also be valid for an HTS conductor with low index value. Due to the issue of Nb₃Sn CICC degradation management [16], the ITER project revised the initial conductor criteria [120] in this way but *a posteriori*. It is important to note that even large Nb₃Sn that does not experience degradation also has a low transition index.

The challenge is to determine the acceptable maximum heat load in the high-field region as a replacement of the temperature margin to design the conductor.

By mid-2020, three CICC machines were in operation in the world (EAST, KSTAR and W7-X). W7-X will be commissioned at its full magnetic current in 2021 [121]. JT-60SA is expected to start operation in September 2020 [20]. Usually, during the commissioning phase of these machines, the investigation of the operating margin is not planned because of the obvious related risks. Indeed, doing such margin tests on a model coil or individual serial coil is manageable, but on a full magnet system it usually leads to investment protection considerations. Again, the low index CICC may change this paradigm as it is possible to operate coils in the current sharing zone still with a significant margin before quenching the coil. As an example, to follow the ITER TF conductor degradation process, it is envisaged to locally increase the coil temperature near the high-field region up to some voltage appearance along the machine lifetime. It is considered as feasible

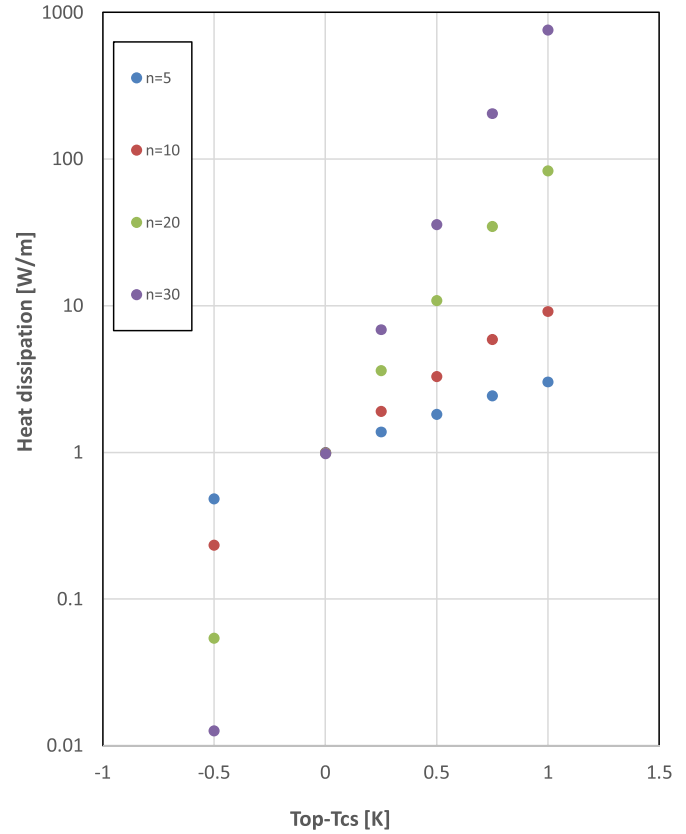


Figure 21. Heat dissipation at peak field region assuming I_c is defined with an electrical field criteria of $10 \mu\text{V m}^{-1}$ for a conductor index n as a function of the operating temperature minus the current sharing temperature. A positive value means then that ΔT is negative. The operating current is 100 kA.

only because it was demonstrated during Nb₃Sn CICC model coil tests that operation even in the current sharing zone is still stable. Such tests performed on machines operating a Nb₃Sn magnet system would provide a unique database for future machine design.

In parallel, the development of a fusion-oriented multi-physics platform such as [122] is needed to simulate the magnet system using the above data as a benchmark. The challenge here is to develop fast processing to be able to use these codes in an optimization loop.

If the development of fast processing system codes is achieved and successfully benchmarked, the conductor design could then be integrated into an optimization loop taking into account the cryogenic plant (balancing, for example, the cryogenic plant kW/€ cost with the kg/€ superconducting material) and the realistic time and spatial heat load distribution over the coil during the reference plasma scenario. The main optimization constraint is then the coil thermal stability.

Concluding remarks

The superconducting magnet is a cost driver for any fusion machine (see ITER and DEMO cost sharing in figure 5 in

the introduction chapter). The last decade's development of a large Nb₃Sn CICC has shown that the usual temperature margin criteria could be reviewed in an optimization loop balancing the heat load due to operation in semi-resistive conditions and the requested cryogenic power. This approach at the conductor and coil design stage requires reliable and fast processing codes that have to be qualified preferably on existing or soon-to-be-operating fusion devices. The final objective is to develop the most cost-effective design approach. Such design optimization approach should also be considered on

other magnet key cost drivers such as manufacturing tolerances versus error field.

Acknowledgments

The author would like to thank Neil Mitchell for his availability to discuss ITER conductor development and design criteria.

The views and opinions expressed herein do not necessarily reflect those of the ITER organization.

10. Superconducting switches for energy extraction in fusion magnets

Nikolay Bykovskiy, Alexey Dudarev and Matthias Mentink

CERN, Experimental Physics Department, Geneva, Switzerland

Status

The magnetic energy E stored in fusion magnets ranges from hundreds of MJ for the machines already in operation up to about 50 GJ for ITER and over 150 GJ pursued for a DEMO power plant. Quench protection for such magnets is commonly based on external discharge resistors and massive commutation switches, resulting in an exponential current decay during a fast discharge of the magnet. Stabilizing copper is also used in fusion conductors to enhance their quench capacity, though its amount has to be minimized for higher efficiency of the system. In fact, to keep the hot-spot temperature at 150 K, the copper current density scales as $j_{cu} \text{ (A mm}^{-2}\text{)} \approx 500/\sqrt{\tau \text{ (s)}}$, where $\tau = 2E/I_{op}U_m$ is the decay time constant, I_{op} the operating current and U_m the maximum voltage over coil terminals. Hence, in order to achieve high efficiency ($j_{cu} > 100 \text{ A mm}^{-2}$), τ must not exceed some tens of seconds, which together with $I_{op} < 100 \text{ kA}$ and $U_m < 10 \text{ kV}$ requires that $E < 10 \text{ GJ}$, highlighting the need for sectioning of large superconducting magnets.

The ITER TF coils, for instance, are split into nine sectors each containing two coils, resulting in 11 s for the time constant [123]. As sketched in figure 22, nine discharge units and nine pairs of HTS current leads and long bus bars are then necessary, leading to demanding cryogenic requirements [124]. Each discharge unit is essentially composed of a discharge resistor, mechanical switch, vacuum and pyro circuit breakers, which are essential to ensure backup switching in case of any failures in the current commutation unit. The components are distributed in two different buildings at the site, and the power supply is situated over 100 m away from the main tokamak building.

Design solutions of ITER are widely adopted by the DEMO project. However, due to the larger scale, some of them will have to be reconsidered. For instance, accounting also for mechanical aspects of the vacuum vessel, the discharge time constant is further increased up to 35 s, allowing two TF coils per sector operated up to 90 kA. As a result, the copper current density is reduced and the busbars dissipate several MWs, about 1% of the generated electrical power [125, 126]. A potential workaround for these issues is discussed below.

Current and future challenges

As an alternative to the use of magnet feeders at ambient temperature, ‘cold’ switches were proposed for the coil sectioning [127]. Their basic properties have been evaluated [128] and some promising materials identified [129]; thus, experimental demonstration should finally be conducted. The switches are used essentially as follows; see figure 23:

- They are closed in regular operation. Adjacent coil sectors are shorted and related heat loads are minimized.
- They are open in the case of a quench. Voltage to ground among coils is not exceeded and the stored energy is extracted.

In the case of fusion magnets, discharge resistors at room temperature, connected in parallel by safety leads, are also required to effectively extract large energy and to avoid over-sizing of the switch. Safety leads made of stainless steel maintain low cryogenic loads. Hence, using only one pair of HTS current leads and bus bars, the electrical power loss and complexity of integration can be reduced substantially. In contrast to the mechanical switches, the high voltage developed on the cold switches during the magnet fast discharge is not associated with arcing issues, so vacuum breakers are no longer necessary. Sub-sectioning of individual coils by the cold switches is also made possible, which allows further reduction of the discharge time constant.

The stored energy is dumped almost entirely into the discharge resistors, whereas only a small fraction should be taken by the switch for a cost-efficient design; thus, the switch resistance is in the order of Ohms compared to $\approx 0.1 \text{ Ohms}$ for the discharge resistor. The switch needs a large thermal sink in order to further reduce the energy absorbed directly by the conductor enthalpy. Using a large thermal mass insulated from the conductor (e.g. existing components of the cryostat) is a simple passive solution, in contrast to more efficient active cooling systems, which would require further development. Synchronous operation of the switches is crucial, as it is for the commutation switches in the conventional circuit.

Superconducting materials are the most promising to achieve low ‘on’-state and high ‘off’-state resistances for high-current and high-voltage operation. Considering other potential technologies, mechanical switches would suffer from arcing issues in a helium environment, whereas MOSFET and IGBT units are of relatively low voltage and current ratings, respectively.

Advances in science and technology to meet challenges

The cold switch comprises a number of long composite superconductors with a highly resistive matrix, which are arranged in non-inductive windings and also include quench heaters or other appropriate means for the switching. Taking the ITER TF coil parameters as a reference, a fast-discharge of 41 GJ using nine switches is evaluated for the two promising materials; see table 6: NbTi wire in CuNi matrix operated at 4 K and non-stabilized REBCO tape operated at 4 K and 50 K. The main technological aspects are then assessed as follows:

Material. The NbTi/CuNi wire is commercially available in long lengths, but the REBCO tape is customized by reducing the silver layer thickness to $0.1 \text{ }\mu\text{m}$, which is about the minimum possible using etching [130]. As a result, the required length is four times lower compared to regular tapes ($\approx 2 \text{ }\mu\text{m}$

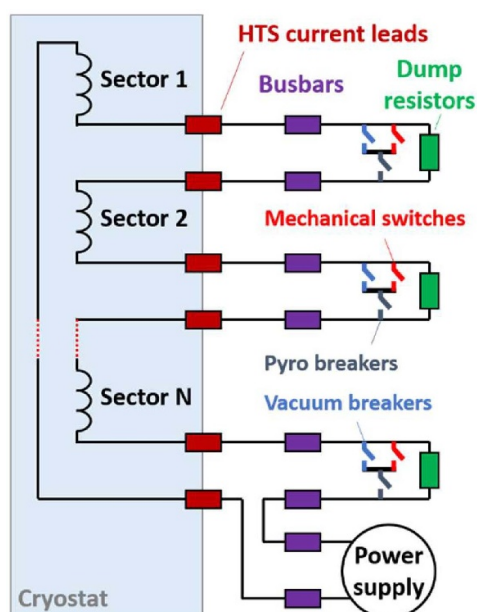


Figure 22. Schematic diagram of coil sectioning with discharge units connected between adjacent coil sectors by HTS current leads and bus bars. Operating current flows through closed switches of the discharge units in normal operation, whereas the switches are open in the case of a quench and the magnet's stored energy is evacuated by the dump resistors.

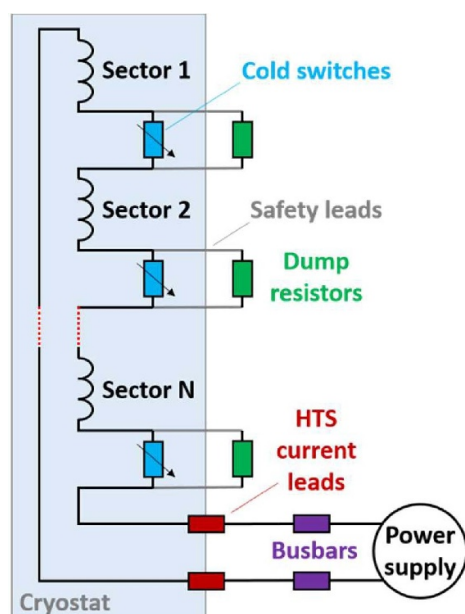


Figure 23. Schematic diagram of coil sectioning using superconducting switches and dump resistors connected in parallel by safety leads. The switches are controlled similarly to those of the dump units in the conventional circuit. Only one pair of HTS current leads and bus bars is required for the operation.

thick Ag) because of the much higher resistivity in the normal state. However, the feasibility of the tape treatment over hundreds of meters has to be demonstrated. Cost reduction for REBCO is necessary and cannot be resolved by boosting its current capacity.

Table 6. Material estimate for a superconducting switch operated at $I_{op} = 68$ kA, $U_m = 10$ kV, $E = 4.5$ GJ, $T_{max} = 200$ K.

	0.8 mm wire	12 mm \times 30 μ m tape
Material composition	0.20 mm ² NbTi 0.30 mm ² CuNi	0.01 mm ² REBCO 0.36 mm ² Hastelloy <0.01 mm ² Ag
Operating temperature	4 K	4 K 50 K
No. of wires or tapes	60	7 48
Length of sub-cables	0.5 km	1.4 km 0.6 km
Energy to reach normal state	0.2 kJ	300 kJ 650 kJ
Length of material	30 km	10 km 27 km
Material cost	≈ 80 k\$	≈ 600 k\$ ≈ 1600 k\$

Winding. Sub-cables of some hundred meters are needed to achieve sufficiently high resistance in the normal state. They are insulated from each other and arranged in a bifilar manner, thus minimizing their self-inductance. To reduce the amount of conductors for a given current capacity, the switches should be placed in a low-field region of the magnet system.

Thermal sink. About 98% of the energy is evacuated by the discharge resistor, only about 0.1% by the conductor itself and 2% by the heat sink, which is assumed to provide a factor 20 increase for the switch enthalpy, thus reducing by a factor 5 the material length.

Reliability. This is a key concern for the NbTi option, given that the minimum quench energy is reduced due to the high normal resistance. Proper mechanical fixation of the conductor is essential.

Quench. Protection of the switch is provided by its switching system and the discharge resistor. NbTi features require relatively low energy to achieve the normal state, whereas demanding requirements are imposed by REBCO to ensure fast switching. In the case of thermally activated REBCO, embedded heaters have to be co-wound with the tapes to address the slow quench propagation.

Concluding remarks

Economic considerations in the magnet design are crucial to realize a competitive fusion power plant. Using superconducting switches for magnet protection is promising to drastically reduce the complexity of integration, electric power dissipation and overall cost. Using one coil per sector, or even subdividing individual coils into sections by the switches, is rather straightforward in contrast to the conventional design based on the magnet feeders installed at room temperature. The NbTi/CuNi wires are considered the most promising for the application, providing a cost-efficient and robust design.

Hence, further development of the required technology for the high-current and high-voltage switches and the following experimental validation can pave the way towards high-performance fusion magnets. Using the NbTi/CuNi wires at

CERN, operation up to 3 kA of a four-wire switch featuring 3 Ohms normal resistance was recently demonstrated and a 6-around-1 cable with insulated central heater was procured for further design upscaling (to be presented elsewhere).

11. Overview of HTS joint technology for segmented coils

Franco Julio Mangiarotti

CERN Technology Department, Geneva, Switzerland

The assembly of HTS magnets with segmented coils has a number of advantages compared to traditional coil winding, such as the simpler and modular manufacturing, the possibility to use short lengths of high-performance superconductor, and increased flexibility in the design, manufacturing and maintenance of other reactor components. This concept was first studied in Japan by Hashizume *et al* [131] and in the USA by Olynik *et al* [47]. Those designs evolved into the recent FFHR-d1 [132] and ARC [7] designs.

FFHR-d1 is a heliotron whose helical coils are divided in segments of one helical twist pitch [133] and joined in series with a mechanical lap joint. In the joint, the HTS tapes are in face-to-face contact, pressed by the copper stabilizer and steel jacket. The steel jacket is welded to ensure good mechanical strength. This winding method is called ‘joint winding’ and is illustrated in figure 24 [131]. The reactor has a total of 3900 joints. An industrial robot assembles and welds one joint individually in about one day; two robots could work simultaneously on two joints at opposite ends of the winding for a total assembly time of 3 years [134]. For FFHR-d1, the reduction of magnet manufacturing complexity drives the use of segmented coils: traditional winding would require a large poloidally and toroidally rotating machine installed in the reactor building to continuously wind the helical coils, while the segmented coils could be pre-fabricated elsewhere and assembled on-site. In addition, the joint-winding method would simplify the lead extraction from the magnet by allowing for pancake winding.

The ARC design is a tokamak whose TF coils are divided into two segments at the top and outer midplane. In the joint, the HTS tapes are covered with a copper protection layer, which is the contact interface. The forces are taken by an external structure. The joints are not permanent, as they are disassembled to perform maintenance in the reactor. This design is called ‘demountable coils’, and is illustrated in figure 25 [7, 135]. A radiation-resistance robot assembles and disassembles all conductor joints in each TF coil simultaneously; the time required was not estimated. For ARC, the main advantage of the use of segmented coils is the simplification of the vacuum vessel and divertor maintenance: instead of using robotic arms for sector maintenance, the entire vacuum vessel can be manufactured elsewhere with tighter tolerances, and be installed as a single piece after the coils are disassembled [7].

Current and future challenges

The FFHR-d1 cable and joint concept is relatively advanced. The 100 kA HTS cable, named STARS (stacked tapes assembled in rigid structure) [134], was designed for this application, and a prototype conductor was tested to full

current. A demonstration joint was assembled and tested, reaching a joint resistance of $15 \text{ p}\Omega\text{m}^2$. This joint consisted of two terminations with three stacks of REBCO tapes each, arranged in a staircase pattern, with a bridge-type mechanical lap joint; in this joint, each REBCO tape in the conductor is in direct contact with the tape in the bridge [132]. By heat-treating the joint to 100°C , demonstrated in single tape joints, the joint resistance was further reduced to $3.5 \text{ p}\Omega\text{m}^2$ [136]. Alternative joint architectures are under investigation, such as an edge-type joint that reached $25\text{--}50 \text{ p}\Omega\text{m}^2$ in a small-scale experiment. The main technological challenges for this concept reside on achieving these results with a real scale joint, applying all techniques together in a reasonable time, with an industrial robot.

The ARC joint design is less advanced. Two conductor designs have been proposed: round and twisted [7, 137, 138] and square and not-twisted [135]. The expected current is around 70–100 kA. A small-scale, proof-of-concept joint test of one joint design has been done up to 3 kA [135], obtaining a joint resistance of $110 \text{ p}\Omega\text{m}^2$. This joint consisted of two terminations with a stack of REBCO tapes, soldered in a slot in a copper block with one of their edges exposed; a silver-coated copper plate was soldered on top of the exposed edge of the tapes, and the electrical joint was made by pressing the two silver-coated faces [135]. More recently, a round cable design has been tested up to 50 kA with a double-saddle joint, consisting of a copper intermediary piece between two round cables [138]. This joint design is effective for cable tests, obtaining a joint resistance of around $25\text{--}100 \text{ p}\Omega\text{m}^2$; however, its low tolerance to misalignment makes it adequate for a reactor design [135]. One of the main technological challenges for the ARC concept is to demonstrate a full-current joint and cable, possibly with a lower joint resistance.

Furthermore, the ARC design requires several individual joints to be assembled simultaneously. Some concepts have been discussed [135, 138] but they have not been demonstrated. The second main technological challenge for this design is to build and demonstrate a multi-joint assembly, including the procedure to make and separate the joints, and the repeatability of its performance at high current.

Advances in science and technology to meet challenges

There are three common technology advances required for both joint concepts. One fundamental difficulty is electrical insulation. Neither the FFHR-d1 prototype joint nor the ARC proof-of-concept joint assessed the electrical insulation across the joint. For ARC, small-scale tests have been done [139], but its scalability to several simultaneous joints has not been assessed. The insulation scheme at the joint level needs to be carefully designed, its application in the joint tested, and its fabrication demonstrated.

The assembly procedure of a single full-sized joint also needs to be demonstrated. In this regard, the FFHR-d1 is more advanced, as the joint prototype is very similar to the final

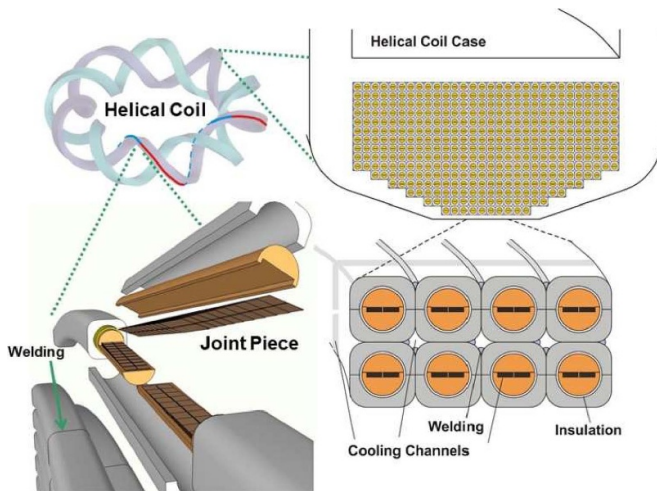


Figure 24. Illustration of the FFHR-d1 coils. Top left: helical coils. Bottom left: joint between conductors of two adjacent segments. Right: cross-section of the helical coils. Image from [132].

design; however, this joint was bolted instead of welded, and as discussed above it did not have the electrical insulation. The ARC joint was small-scale and limited to the electrical conductive parts, thus not testing any of the support and alignment elements.

The third technology to be demonstrated is the robotic operation. FFHR-d1 benefits from a robot to assemble the joints in shorter time; ARC requires it because the joints will become radioactive during operation and manual intervention would not be possible. The robot should perform the joint electrical assembly, electrical insulation, mechanical support (welding, bolting and key insertion, depending on the design) and testing.

In the case of ARC, some additional challenges need to be addressed. The final cable design needs to be determined, either by developing and testing a new concept or by adopting an existing tested design. The joint design then needs to be adapted to the cable design; this can be challenging in the case of twisted conductors. A conductor and joint prototype then needs to be demonstrated at the operation conditions of current, magnetic field and temperature. In addition, the multi-joint concept reliability and repeatability needs to be assessed and demonstrated.

Concluding remarks

The use of HTS segmented coils in fusion reactors allows for greater flexibility in the design and manufacturing of magnet systems and other internal components, such as the vacuum vessel or the divertor. Demonstrating the technical and technological feasibility of the joint-winding and demountable coil concepts is critical for the development of the FFHR-d1 and

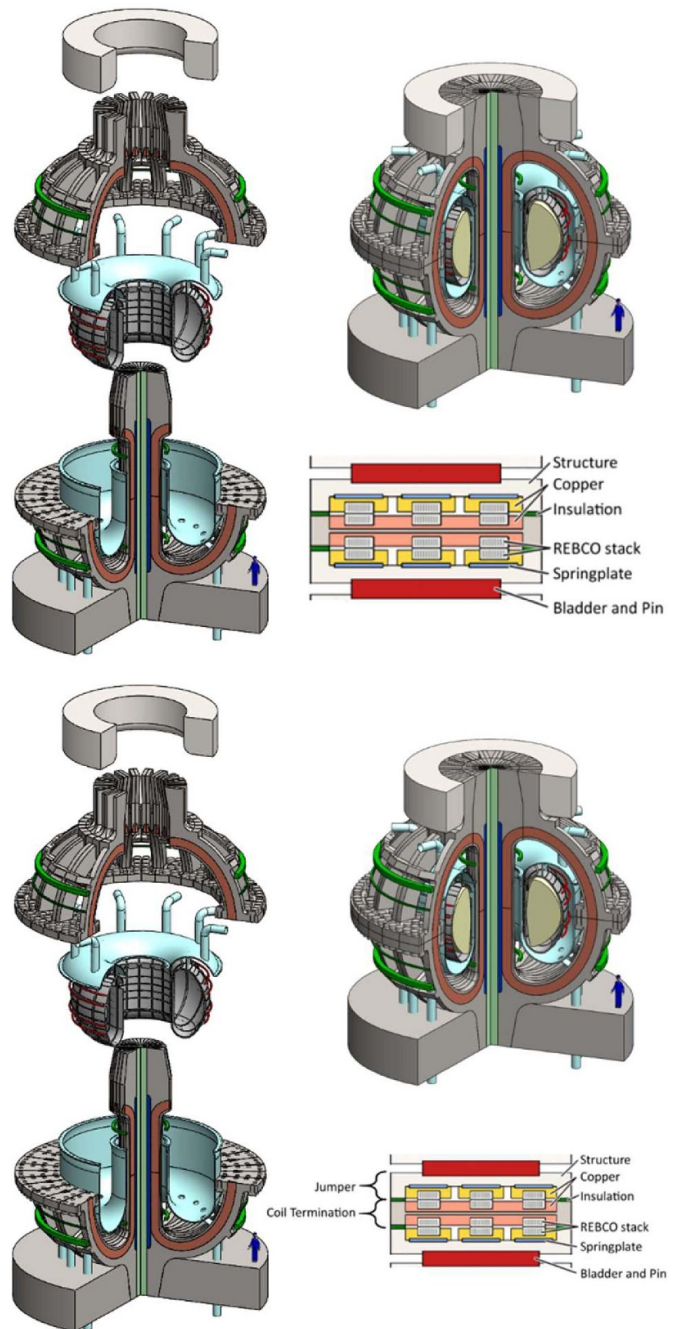


Figure 25. Illustration of the ARC coils. Left: reactor disassembled. Top right: reactor assembled. Bottom right: conceptual cross-section of a joint. Images from [7, 135].

ARC reactor designs, which rely on this technology. Although the FFHR-d1 joint concept is more advanced than ARC's, both require substantial development in terms of manufacture and operation. Once demonstrated, this technology has the potential to simplify the manufacture and/or assembly of any other HTS reactor design.

12. Advances in React-and-Wind Nb₃Sn coils for fusion

Kamil Sedlak

Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-5232 Villigen PSI, Switzerland

The largest challenges in building fusion power plants are often claimed to be technological ones, e.g. the development of materials sustaining enormous heat flux on the reactor walls. A challenge that is in our opinion of at least the same importance is related to the costs of the produced electricity [140], which blows up with the increasing size of the machine. The corresponding increase in the material costs is of lower importance—the main issue is the increased complexity of manufacture, logistic and handling cost, and the manufacturing time. An extreme example of addressing this problem is the compact HTS-based tokamak concept, e.g. SPARC developed by CFS or an ST developed by Tokamak Energy Ltd. In this article, we propose a roadmap toward the conventional Nb₃Sn-based tokamaks, which is based on an improved conductor design leading to more compact magnets, perhaps with a slightly elevated magnetic field, and reduced manufacturing complexity compared to the present state-of-the-art fusion magnets. This goal is achieved by adapting the react-and-wind (RW) conductor manufacturing path.

RW conductor

An obvious advantage of the RW method is the lower axial strain ($\varepsilon \approx -0.3\%$ [39, 141]) in Nb₃Sn cable compared to the ITER-like wind-and-react (WR) cable ($\varepsilon \approx -0.7\%$ [142]). The lower strain in the RW conductor is a direct consequence of the jacketing of the reacted cable at room temperature. The beneficial effect of the lower strain has been experimentally demonstrated by an RW conductor developed for an alternative ITER TF coil [141, 143] and EU DEMO project [34, 39, 116]. When tested in ITER-like conditions (10.9 T, 68 kA), the EU DEMO conductor with 132 mm² of Nb₃Sn achieved a higher current-sharing temperature ($T_{CS} = 7.42$ K [40]) than a typical ITER TF conductor with 238 mm² of Nb₃Sn ($T_{CS} = 6.3$ K–6.5 K [142], internal tin). Both conductors, compared in figure 26, are forced-flow-cooled.

However, the main potential of the RW technique does not lie in the strain reduction, but in the very different jacketing procedure that opens up completely new possibilities in the coil design and manufacture. The natural choice for the jacketing process is longitudinal welding of the two rolled or extruded steel half-profiles surrounding the reacted cable. Even though the total length of welds might look unfavorably large, in the last ~30 years longitudinal welding has become a standard, reliable and cheap industrial process, e.g. in the car industry. Contrary to WR, the jacket and welds are not exposed to heat treatment, which avoids material embrittlement and simplifies quality assurance. The fundamental advantage is complete freedom in the jacket thickness and shape, and virtually unlimited conductor manufacturing length.

Layer winding and conductor grading

The benefits of RW technology are best exploited when combined with layer winding. Each layer can be graded independently in Nb₃Sn, helium, copper and steel.

Grading in superconductors is an obvious option that leads to a significant reduction of the total required amount of Nb₃Sn strands, which directly reflects into the reduced material costs. As an example, the RW winding pack designed for EU DEMO [117] consists of 12 layers, and the lowest-field layer (6.2 T) needs only 25% of Nb₃Sn compared to the highest-field layer (12.2 T). The overall saving of Nb₃Sn compared to a non-graded winding pack of the same performance is ~50%. When, in addition, the Nb₃Sn saving due to the lower strain in the RW technique is taken into account, the total saving in Nb₃Sn reaches ~73% compared to an ITER-like DEMO design (222 tons of Nb₃Sn strands for RW compared to 835 tons of strands for pancake-wound WR winding pack).

The grading of the steel conduit together with the freedom in its shape is another important benefit of the RW technique. The jacket wall thickness (in both the radial and toroidal directions) can be adjusted exactly according to the local stress in a given layer of the winding pack, minimizing the radial build of the coil and consequently reducing the overall size of the machine. At first glance it might seem that the manufacturing costs of a fusion machine are proportional to its volume, i.e. they grow with $\sim R^3$, as this is how material costs, including standard machining, depend on object size. However, the manufacturing complexity, high-precision machining and handling of very large and heavy components do not scale linearly with the weight, but are much steeper. For instance, a crane rated to 800 tons (approx. weight of the EU DEMO TF coil) is not just two times more expensive than a 400-ton one. Some industrial processes, e.g. forging weight or the welding depth required for the TF case, are limited in dimensions, and become extremely challenging in DEMO-size machines. Therefore, any reduction in the radial build of the coils, and consequently of the whole machine, becomes extremely valuable.

An important advantage of the RW technology is the simplicity of the coil winding done in a continuous process, in which joints are made at the winding table before bending the conductor. Unlike in WR, there is no transfer-insulation process, which simplifies the winding procedure such that it resembles the winding of NbTi or copper coils. This simplifies tooling and reduces risks during coil manufacturing.

Possible future investigations

The manufacture and testing of 60–70 kA RW conductor prototypes [40] and a diffusion-bonded joint with 0.54 nΩ resistance [144] provide a solid basis for the next generation of fusion coils. Let us now outline some other (speculative) ideas pushing the R&D further:

- High- J_c Nb₃Sn strands: The progress in the development of Nb₃Sn strands with artificial pinning centers presented

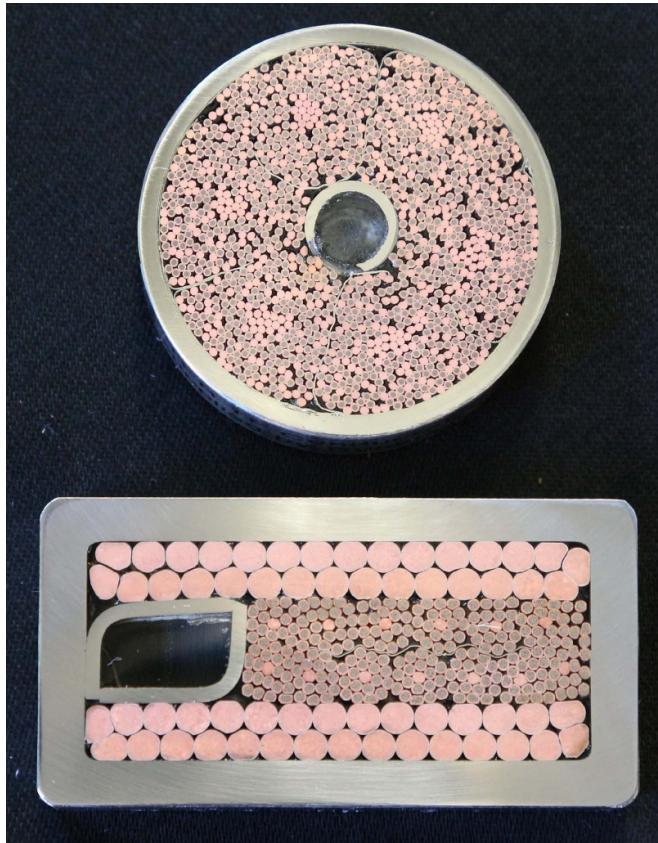


Figure 26. Comparison of the ITER TF conductor based on WR technology (upper plot) with the latest EU DEMO TF conductor (lower plot) consisting of a flat RW cable next to a rectangular cooling channel, both surrounded by two Rutherford cables made of coated Cu wires as a stabilizer. The two pictures are to scale. The amount of Nb₃Sn in the RW cable is 55% of that of the ITER one, while the segregated Cu cross-section is exactly two times larger due to the prolonged discharge time (35 s in DEMO vs. 11 s in ITER).

at conferences during the past years looks promising [145]. Once these high- J_c strands become available in industrial quantities, they could be employed in the highest-field layer. An approximate 10% increase of the field on the plasma axis would become feasible, allowing us further reduction of the tokamak size. Even without exploiting the field increase, at least the cable could be slightly more compact, reducing the dimensions of the steel jacket and thus of the whole winding pack.

- A jointless TF/CS coil: as the jacketing by longitudinal welding can be done in unlimited lengths, one can imagine manufacturing the conductor length for the whole coil in one go. Joints always constitute weak points. They require a lot of R&D and quality assurance, increase the risk of coil failure, significantly complicate and slow down the coil winding, and call for cold testing with the current. A usual outcome of the RAMI analysis is a request to provide access to the joints for a repair, or even design the machine in a manner allowing replacement of a failed TF coil. The jointless coil has two major challenges. Manipulation with the conductor spool and coil winding becomes

very demanding, requiring huge tooling. The second challenge is the conductor grading, requiring smooth grade-to-grade transition during the cabling and jacketing processes, likely requesting variable strand diameter or strand joints, strand-to-copper wire joints, change of steel profile inner dimensions, etc. One can anticipate that the conductor grading would need to be simplified or, in an extreme case, sacrificed. Concerning the helium cooling, inlet and outlet cooling pipes can be installed (welded) at the winding pack edges, so that every layer is cooled by fresh helium.

- Solder filling of the strand bundle: the performance degradation of ITER TF conductors on electromagnetic and thermal cycling, whose significance depends on the conductor producer, is a warning lesson for any new conductor concept. Though it seems that degradation can be avoided in RW conductors [40, 143] (in [40] by transverse pre-load applied on the cable during jacketing), an option of solder filling of the strand bundle has been investigated [141], and related R&D is envisaged also within the EU DEMO project. Moreover, the solder-filled cable relaxes the hydraulic containment function of the jacket, and hence also the issue of jacket fatigue in the case of a CS coil. A high-resistivity soldering alloy is required for keeping the overall AC loss low.

Drawbacks and risks of the RW concept

RW forced-flow conductors have been used in several (fusion) coils in the past, e.g. in the Mirror Fusion Test Facility (Livermore, USA), Levitated Dipole Experiment (MIT, USA) and more recently in a 25 T cryogen-free magnet [146] at IMR (Tohoku University, Japan). The disappointing experience from the T-15 tokamak, where the TF coils became resistive with ohmic heating in the kW range right from the first operation in 1988, presumably due to handling during which bending strain reached $\pm 0.8\%$ [147], brought a lot of aversion to the RW technology. However, the example of the main DC magnets of the SULTAN test facility [148], also built at the end of the 80s and in heavy operation since then, proves that the RW magnets can be very reliable, withstanding many electromagnetic and thermal cycles as well as fast discharges. The necessary conditions are not extraordinary demanding—the cable must be flat enough [141] to withstand bending strain during straightening for the jacketing process, and bending back to the final shape. In the case of the 12 T SULTAN coil, the 3.3 mm thick cable was heat-treated at the radius of 0.6 m, jacketed straight, and finally wound to the coils with a radius of 0.3–0.4 m. Since the minimum bending radii in the big fusion magnets are an order of magnitude larger, the bending strain can be restricted to an even safer range compared to SULTAN coils. In the EU DEMO with a cable thickness of ~ 11 mm, the actual bending strain in operation at various positions along the D-shape varies in the range of $\pm 0.1\%$, and during the straightening for jacketing $\pm 0.25\%$. Both ranges can be considered very safe. Industrialization of the historical RW conductor production is outlined in [149], pages 223–225. The process needs to be updated for large-sized coils.

During the manufacture of DEMO RW prototypes [117], some of the conductor samples made of heat-treated cables were assembled and disassembled several times—to change the segregated Cu stabilizer for a different one, or to make a joint sample out of a conductor sample. No degradation of J_c was observed after the re-assemblies, indicating that the relatively rigid flat RW cable is not impractically fragile or delicate to handle.

Concluding remarks

The Nb₃Sn conductors made with RW technology show great potential in providing very compact fusion coils that are easy

to manufacture, of low AC loss, and that can reach a higher magnetic field compared to their WR counterparts. The RW conductors can be directly linked to HTS ones, which are also based on the reacted conductors, e.g. in a hybrid coil. The research done on the prototype RW conductors and joints provides very encouraging experimental results and stimulates further exploration of this concept.

Acknowledgment

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13. Plastic resins and composites at low temperatures

David Evans

Advanced Cryogenic Materials Ltd, Oxfordshire OX14 2HQ,
United Kingdom

Status

Pastics, resins and composites are being increasingly used in low-temperature devices in the form of electrical and thermal insulation, vacuum sealants and matrix materials. A major role is as impregnation materials, where they often have the dual purpose of bonding and insulating. Resins and composites have high dielectric strength, and provide reliable turn and layer insulation in magnets. Unidirectional rods are used as high-strength, low-thermal-conductivity support struts, and liquid resins can have low viscosity and a long useable life for impregnating tightly wound

Under low-temperature service conditions, these materials may be subjected to stresses resulting from their difference in thermal shrinkage with metals, tensile and shear forces from other sources and frequently in an ionizing radiation environment. These conditions, along with the anisotropic nature of their thermal and mechanical properties, place an onerous burden on the design engineer.

Additionally, plastic materials are well below their rubber-/brittle transition temperature (glass transition temperature T_g) at operating temperatures, meaning that they are brittle and that the relaxation of thermal stresses or stress concentrations cannot take place; radiation damage that may result from the evolution of gases is still not confirmed or understood.

Serious research and reporting of the low-temperature properties of plastic materials dates from around 1979 with a series of topical conferences [150] dedicated to these materials and major international meetings where low-temperature properties formed a major element (see for instance [151]). There were, however, a number of important meetings and workshops initiated by the fusion community prior to 1979. Since that time, the many published reports on physical properties at low temperatures have been scattered throughout the technical literature. There is a real need to gather these data into one readily available source.

In more recent years, with a few exceptions, developments in non-metallic materials technology have not kept pace with the expanding use of these materials in fusion technology. The use of cyanate ester/epoxy blends for impregnating magnet structures that are subject to high accumulated doses of high-energy radiation was a development of one such system, carried out with a contract sponsored by the ITER organization. The US ITER insulation program was responsible for much work on radiation stability and gas evolution but, as useful as this project was, there are still unanswered questions.

Engineering design data on bond strengths are sadly lacking, and not just at low temperatures—the situation is unsatisfactory at room temperature.

Current and future challenges

Radiation and evolved gases. The degradation of mechanical properties of epoxide resins and composites, as a function of the total integrated dose of ionizing radiation, has been documented in numerous publications and summarized, at least up to the late 1990s, in [152–154]. The basic structural features that contribute to radiation stability are understood [155] but what is not yet fully understood is the influence that evolved gases may have on the long-term behavior of resins and composites.

For a range of epoxies, gas evolution rates as a function of the total dose, derived from irradiating finely powdered resins, are presented in table 7 [156].

It was shown [157] that when specimens were irradiated as small cubes, the measured gas evolution rates were up to 50% lower than those of comparable material irradiated as fine powders. This raised concerns that when materials are irradiated at low temperatures, the sudden release of gases on warming could lead to an internal pressure and an increase in dimensions and/or failure.

Following this revelation, a number of studies were undertaken on composite materials. One report [158] showed thickness swelling of up to 2.6% when 3 mm thick S-glass composites were reactor-irradiated at 5 K to a total dose of 57 MGy. A second report [159] showed even more dramatic swelling, and the results from the three types of composite irradiated are presented in table 8.

However, a range of five different epoxy resins, in combination with three different hardeners, was prepared as machined cylindrical specimens, without fillers or fibers, 10 mm in diameter and 10 mm long [160]. Specimens were measured individually (reproducibility was ± 0.01). Rectangular composite specimens were irradiated in the same experiment and no measurable swelling was recorded in any specimen and no change in the Young's modulus was apparent.

At the neutron fluence and energies used in the experiments reported here, it is reasonable to assume that all apparent swelling occurs within the resin phase only and the glass is dimensionally stable. Dimensional changes would not be expected to occur in the fiber direction and all changes would be limited to resin expansion (and/or) delamination in the through thickness direction.

Cyanate ester resins offer radiation stability, a long useable life and low viscosity, and possibly a low gas evolution rate. These materials are costly, sensitive to contamination and the cure procedure must be carefully controlled in order to avoid a 'runaway' exotherm. When a single coil can use a cubic meter or more of resin, cost can be a major consideration.

Is 'swelling' real and likely to prove a problem in fusion reactors and therefore necessitate using cyanate esters? Or could we use more economical, easier to control epoxies that offer a long life and low viscosity but have higher rates of gas evolution?

In the authors' view, gas resulting from irradiation is 'in solution' in the resin and does not constitute a separate

Table 7. Rate of gas evolution for three hardeners (cc's /g/MGy) at differing total dose levels [156].

Resin	Hardener	Dose (MGy)					
		3	6	9	15	20	30
DGEBA	A	0.6	0.5	0.4	0.4	0.4	—
	B	0.3	0.3	0.2	0.3	0.3	0.3
	C	1.2	0.7	0.7	0.7	0.7	0.6
EPN	A	0.5	0.6	0.4	0.4	0.4	0.4
	B	0.3	0.2	0.2	0.2	0.2	0.3
	C	1.3	1.0	0.8	0.7	0.7	0.7
TGDM	A	0.6	0.5	0.5	0.5	0.5	0.5
	B	0.4	0.4	0.3	0.3	0.3	0.3
	C*	0.8	0.7	0.7	0.6	0.6	0.6
	C**	1.4	1.2	0.9	1.0	1.0	1.2

* Cured at 160 °C. ** Cured at 100 °C.

DGEBA—diglycidyl ether of bisphenol A; EPN—epoxy novolak resin; TGDM tetraglycidyl diamminodiphenyl methane. Hardeners: A: aliphatic amine (room temperature curing), B: aromatic amine, C: acid anhydride.

Table 8. Through thickness swelling of composites irradiated at 5 K [159].

Composite form	Through thickness swelling (%)		
	Radiation dose* (MGy)		
	4	20	39
Disc A	1.3	2.4	5.2
Disc B	1.0	4.2	9.0
Tube	0	0.6	0.5

*Converted from fast neutron fluence of 1×10^{21} , 5×10^{21} and 1×10^{22} n m⁻², with associated gamma dose.

Disc A: 12 mm diameter, 0.5 mm thick; Disc B: as for A but part covered with stainless steel disc; tube 8 mm internal diameter, covered on ID and OD with stainless steel foil, 1 mm wall thickness. (Swelling of 9% on 0.5 mm thickness = 45 microns; swelling of 0.5% on 1 mm wall thickness = 5 μ m).

volume at a high pressure that, if it did exist, could induce swelling.

Measurement of shear strength and bond durability. The reliability of adhesively bonded composite structures is still not fully established, and techniques for providing engineering data on shear strength are not readily available. Existing standard test procedures do not offer design data or address bonding of composites. In addition, when composite materials are used in conjunction with metals, through thickness tensile forces may also be present. It has been shown [160] that dramatic reductions in the through thickness tensile strength of joints occur when thermally induced tensile stresses are also present.

Many of the published data on shear strengths have been obtained using the notoriously unreliable single-lap shear test, and although the 'double-lap' specimen eliminates peel forces, it is more difficult to prepare and still does not generate 'design data'. Current standard test methods do not include the ability to vary the bond-line thickness without interfering with the test volume.

Torsional shear testing of adhesives is said to be relatively free of stress concentrations but the technique is little

used. However, a torsion test [161] was used to measure the fatigue life in shear as a function of the surface roughness and bond-line thickness. The results on bond-line thickness changes are shown in figure 27.

Using the same test procedure and a constant bond-line thickness of 1 mm, the fatigue lifetime was found to peak sharply at a surface roughness of 2.2 μ Ra.

In contrast, it was reported [162] that the maximum fatigue life was obtained at Ra 5 μ .

There is almost a total absence of reliable, short- and long-term data on bond strengths at low temperatures.

Breakdown of wire insulation. A recently highlighted problem concerns the failure of the insulation on High Voltage (HV) wires as they exit from the ground insulation around superconductive magnet coils and other structures. When manufactured, these units may pass all electrical tests; however, numerous failures have been recorded when structures/coils have been thermally cycled.

A number of possible reasons exist to explain these failures, including possible interaction with resins and hardeners used during installation and assembly. This interaction could result in brittleness and lead to failure when the insulation is subjected to cool-down or electro-magnetic stresses (environmental stress cracking—ESC).

Insulation failure due to the interaction of polyimide varnish with cyanate ester resin has been confirmed, and in the authors' view, failure as a result of interaction with aliphatic amine hardeners (ESC), resulting in brittleness/cracking/crazing, followed by stress induced on cool-down, is a strong possibility.

Advances in science and technology to meet challenges

Does radiation-induced gas evolution from organic materials result in 'swelling' that may cause resin/composites to crack or lead to unacceptable dimensional changes in magnet coils?

Based on huge variations in the results from studies to date, opinion is divided, but a resolution of the question would

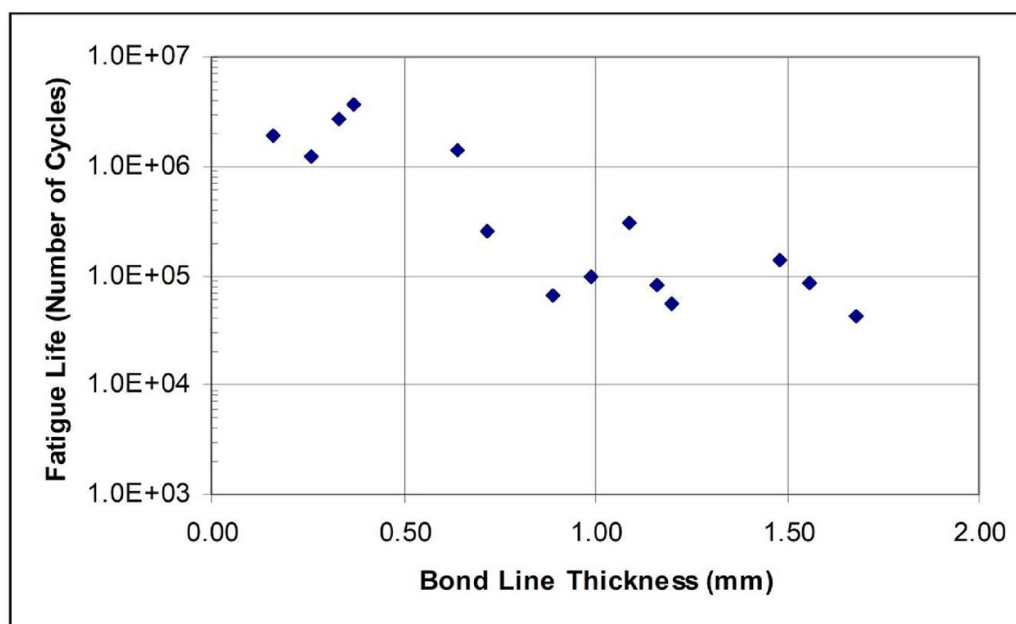


Figure 27. Fatigue life and adhesive thickness ($R_a \sim 2.0 \mu$ and shear stress amplitude 4.0 MPa) [161].

provide guidance into the long-term viability of polymeric materials in fusion devices. If swelling was shown to be unlikely, this would open up the possible field of resins for use in a radiation environment. There are a number of new (to fusion applications at least) resins and aromatic amine hardeners that have low viscosity and offer a long useable life. They are believed to be radiation-stable but are known to have relatively high gas evolution rates. Use of these materials would simplify the impregnation procedure for magnets, while minimizing the risk of a runaway exotherm and being more economical than cyanate esters.

Currently, facilities for material irradiation are scarce. A cost-effective irradiation facility could be created using spent reactor fuel rods. Such rods are usually stored underwater for a ‘cooling-off’ period and could be utilized during this time for the (gamma) irradiation of materials. Dose rates would be low (<0.1 MGy/hour?) but irradiations could continue on a 24 h basis, making for realistic time scales.

A concerted program on the measurement of shear strength of adhesively bonded joints at low temperatures is long overdue (even at room temperature information is poor and unsatisfactory for design purposes). Such a program could usefully investigate the influence of the bond-line thickness and surface roughness of metals and composites, including the use of peel ply on composite surfaces. The long-term durability of bonded structures operating under extreme conditions is unknown, as is the combined effect of through thickness tension with shear forces.

The electrical failures of insulated high voltage wires that have been experienced in coils and sub-assemblies have proven to be difficult to replicate in smaller scale test assemblies. To investigate the possibility that these failures are influenced by the chemical effects of resins and hardeners on insulation, a reliable test procedure is required. This would be

followed by a detailed program on the evaluation of effects of resins and hardeners on varnish types, together with the development of an understanding of possible damage mechanisms and mitigation procedures. As an example, extruded polyimide varnish on wires has a largely amorphous structure and is prone to ESC. Annealing the material at an elevated temperature is said to reduce cracking and crazing, possibly because internal relaxations within the polymer allow the transformation to a more crystalline structure that is known to be less prone to ESC effects.

Much information on the low-temperature properties of polymeric materials has been published over the years in a range of journals, conferences and a few books. However, there is a real need for a comprehensive text book that gathers and critically reviews all available information. This would provide engineers and designers with a common ‘single-source’ reference text. This would be a mammoth task (for instance [154] runs to over 400 pages) but would fill a gaping hole in an engineer’s library.

Concluding remarks

Plastics, resins and composites represent a small but essential fraction of any fusion device. They are radiation-sensitive, with thermal and mechanical properties that may be anisotropic. The production of adhesively bonded joints for low-temperature operation is still a ‘black art’ rather than a sophisticated science. Information on the properties of these materials is widely scattered throughout many publications and, in many cases, lacking precise descriptions of the materials being evaluated.

Radiation damage studies are partly inhibited by the lack of readily available radiation sources, and interest in

evaluating new, radiation stable materials, is tempered by an incomplete understanding of the effects of radiation-induced gases on the dimensional stability of resins and composites.

Design information on the strength and durability of bonded joints at low temperatures is virtually non-existent, and design engineers are further limited by the absence of a readily available data source of material properties.

14. CORC® cable and magnet R&D in preparation for the next generation of fusion facilities

Danko C van der Laan^{1,2} and Jeremy D Weiss¹

¹ Advanced Conductor Technologies LLC, Boulder, CO 80301, United States of America

² University of Colorado Boulder, CO 80309, United States of America

Status

HTSs offer tremendous opportunities for the advancement of fusion science toward fusion energy power production. They allow for more compact fusion machines that operate at magnetic fields exceeding 20 T, compared to LTSs. HTSs can also be operated at elevated temperatures (20 K–30 K), potentially allowing for demountable TF coils [7, 163] that provide easy maintenance access during which the vacuum vessel containing the fusion core is replaced in its entirety, removing typical sector maintenance limitations. They also allow the PF coils to be placed within the TF coils. In the United States, HTS magnets are being considered for use in proposed future fusion facilities, such as the ST option for the Fusion National Science Facility (FNSF) [164], the sustained high-power density (SHPD) facility [165], alternatively named the National Tokamak User Facility (NTUF) [166], as well as the private industry devices SPARC and ARC [7].

Several high-current HTS cables that are based on RE-Ba₂Cu₃O_{7-δ} (REBCO)-coated conductors have been developed for fusion magnet applications, of which an overview is provided in [167]. Advanced Conductor Technologies is commercializing high-current HTS CORC® cables and CICC for use in fusion magnets, and has successfully demonstrated demountable joints between such conductors [168]. Figure 28 shows a range of CORC® fusion conductors, including a low-resistance plate joint between ten CORC® cables developed specifically for use in demountable TF coils.

The basic superconducting components needed for HTS fusion magnets have been developed, allowing for the next major step toward realization of HTS-based fusion facilities. Within the next 5 years, significant prototype fusion magnets should be developed, and their operation demonstrated, to advance HTS fusion magnet technology to the level required for large fusion facilities, such as the FNSF or the NTUF, for which construction may start in the next 10 years [165, 166]. Demonstration magnets that should be considered include high-field CS coils that operate at high current densities and high current ramp rates, and demountable TF coils that require many low-resistance joints and extensive joint support structures. A brief overview of some of the remaining technical challenges that these demonstration magnets should address is presented using examples based on state-of-the-art CORC® cable technology.

Current and future challenges

The performance of several HTS cable technologies has been demonstrated on a laboratory scale. A limited number of full-scale CICC samples, most relevant for TF coils, were tested at currents up to 60 kA in a 12 T background magnetic field in the EDIPO and SULTAN test facilities at the École Polytechnique Fédérale de Lausanne, Swiss Plasma Center [169, 170]. The samples contained multiple twisted stacks of REBCO tapes [169] or CORC® cables in a 6-around-1 configuration [170]. Insufficient mechanical support against the high (cumulative) transverse stresses caused degradation in both types of samples, requiring development of revised layouts in which the CORC® cables within the CICC are mechanically decoupled (figures 28(b)–(d)). Degradation-free operation of the revised CORC-based CICC in SULTAN will likely be confirmed in early 2021.

Demountable joints in TF coils come with significant technical challenges, including the need for joint resistances of 1–5 nΩ, which is achievable in joints between CORC® and CICC. Other challenges include the joint configuration between as many as 100 coil windings, especially considering that each winding contains cooling lines for liquid or gaseous cryogen. Overall stress management of the joints is a major challenge, which depends on the location and overall layout of the joints, as it is the effect of the joint layout on the peak magnetic field that determines the overall conductor performance (figure 29(a)). A high number of CICC and joints also need to be manufactured to high-quality standards.

Recently, a CORC® insert magnet was successfully tested within a 14 T LTS outsert, resulting in a combined field of 16.77 T at a current of over 4 kA and a winding current density of 169 A mm⁻² [171]. The result is highly promising for the development of high-field CS coils, where hoop stress will likely dominate their performance. Multi-cable CICC are incompatible with the small bending radii of for instance the CS coil considered for the NTUF, therefore requiring jacketed single-CORC® windings (figure 28(a)). Demonstration of a model CS coil should include operation at high current ramp rates to ensure that current distribution between tapes remains homogeneous and that ramping losses will not overwhelm the magnet cooling.

Advances in science and technology to meet challenges

Conceptual designs of CS and demountable TF coils, each operating at a peak field of about 20 T, are provided to guide potential near-term magnet demonstrations. Both coil designs are based on state-of-the-art CORC® cable technology, using 50-tape CORC® cables with a current capacity of about 16 kA at 20 T, 4.2 K, or about 8 kA at 20 T, 20 K.

A combination of 18 TF coils with 0.7 m minor and 1.5 m major radii, operating at 4 MA winding current, would provide a magnetic field on the plasma of 10 T (figure 29(a)). Each

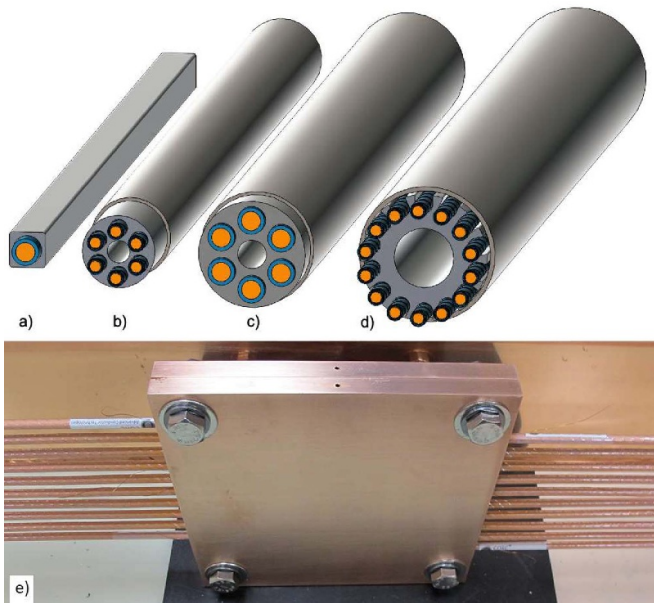


Figure 28. Overview showing CORC-based CICC under development for fusion magnet applications. (a) Jacketed single CORC[®] cable (10 by 10 mm). (b) Extruded CICC containing CORC[®] wires (22 mm diameter). (c) Extruded CICC containing CORC[®] cables (32 mm diameter). (d) Grooved 14-strand CICC based on CORC[®] wires (38 mm diameter). (e) Demountable plate joint (0.2 m by 0.2 m) between ten CORC[®] cables developed in collaboration with the United Kingdom Atomic Energy Authority.

TF coil would require 42 6-around-1 CORC—CICC windings (figure 28(c)) when operated at 4.2 K, or 84 windings when operated at 20 K. A relatively thick square jacket of 40 by 40 mm would result in a winding cross-section of about 0.07 m² (4.2 K) or 0.14 m² (20 K), which should leave sufficient room for additional external mechanical support. Total dissipation per TF coil would be 774 W at a winding current of 96 kA at 4.2 K (42 turns) or 387 W at a winding current of 48 kA at 20 K (84 turns) in case of a joint resistance of 1 nΩ. Another major benefit of demountable TF coils is that they will not require conductors of long continuous lengths and that each coil section could be tested before assembly, significantly reducing the magnet fabrication risk and cost.

The CS coil being considered for the NTUF has an inner radius of 0.27 m, an outer radius of 0.42 m and a height of 3 m (figure 29(b)). The magnet can contain 15 layers of a jacketed CORC[®] cable (figure 28(a)) of 10 by 10 mm in size, operating at 10.9 kA, resulting in a peak field of 20 T. This would be easily achievable at 4.2 K, but would require a more potent or thinner CORC[®] cable when operated at 20 K. The winding current density of 109 A mm⁻² would result in a peak hoop stress of around 595 MPa, for which the jacket would likely suffice.

Concluding remarks

With the development of fusion facilities towards energy generation on the horizon, HTS magnet technology needs

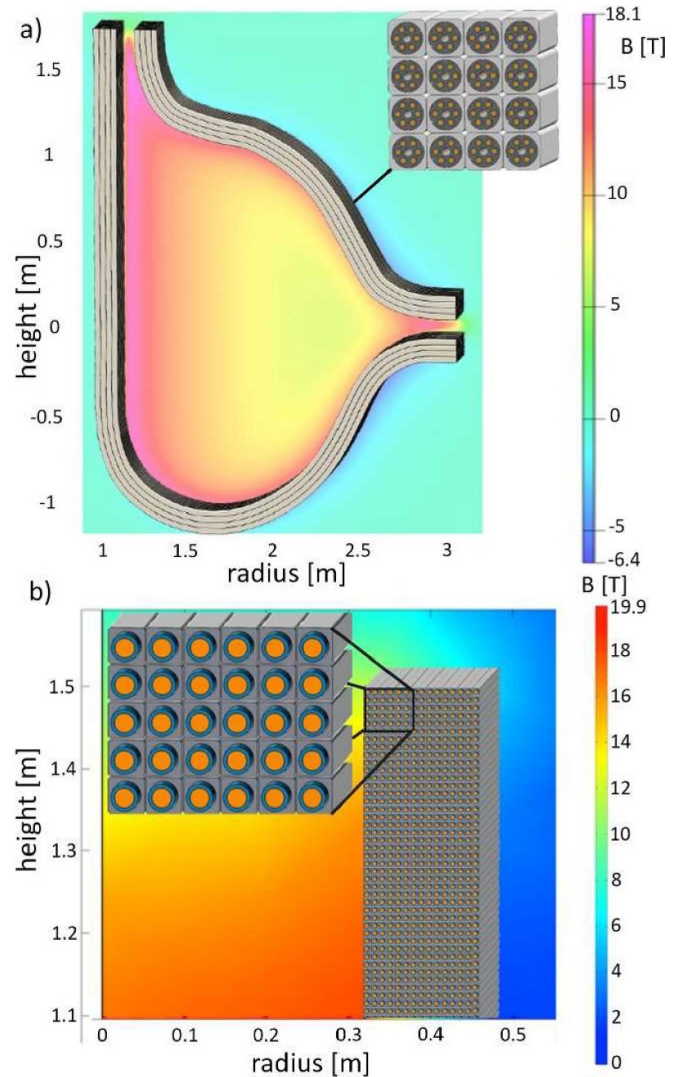


Figure 29. (a) Magnetic field distribution of a demountable TF coil with 1.5 m major radius, not including the joints, generating 10 T magnetic field on the plasma. Also shown is the magnet cross-section containing 16 CORC—CICC windings. (b) Magnetic field distribution at the top quadrant of the conceptual design of a CORC-based 20 T CS coil for the NTUF. Here the windings are formed by a single-jacketed CORC[®] cable.

to transition from laboratory scale to a practical and mature option for high-field fusion magnets in the next 5 years. The demonstration of significant TF and CS coils would address the remaining technical challenges associated with conductor development, magnet construction and operation, including quench detection and protection, while initiating large-scale CICC and joint production. Successful completion of such demonstration magnet programs is the only way to ensure HTS fusion magnet technology is ready for implementation into compact high-field fusion facilities, for which construction will likely start about 10 years from now. Independently of the conductor technology selected for such magnet demonstrations, not only the successes, but also the setbacks and lessons learnt, should be widely disseminated, allowing the fusion community to learn from mistakes, which is the best way to

ensure that the next generation of fusion facilities becomes a success.

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15. Quality of superconductors in a nuclear environment

Min Liao, Neil Mitchell and Gen Liu

ITER Organization, 13067 St. Paul-lez-Durance, France

Status

Magnetic fusion machines are different from most nuclear power plants, where the implementation of mature ASME and RCCM standards for nuclear power projects provides a secure background in engineering quality. Even if not safety-relevant, nuclear fission quality processes are applied with rigorous codes and standards based on experience, such as the ASME Boiler and Pressure Vessel Code (B&PV Code) [172], Nuclear Quality Assurance (NQA) [173], Qualification of Mechanical Equipment in Nuclear Facilities (QME) [174], ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) [175], etc. For ITER—as the world's largest research and development project—there are not so many standards and codes set for advanced superconductors and magnet technology. From the beginning of the design phase, some technical indicators and technical conditions were uncertain and controversial. For example, the stability and margins of the conductor have been one of the key issues. Superconductors and the associated technologies have to make the step to meet nuclear quality expectations while not over-constraining the technical flexibility. The demonstration of the reliability of superconductors, especially for the three basic superconducting technology elements—the conductor, structures and insulation—needs to ensure controllability and flexibility at the same time [123].

In the tokamak device, successful operation relies on the good quality of superconductors, which form the most important part of the magnet system (coils for TF, PF, CS and correction coil (CC)). The superconducting coil works under a complex electromagnetic noise environment, dominated by the changing magnetic fields, especially those from the plasma disruption and plasma control, which produce AC losses and a large transient thermal deposition, particularly on the TF coils. The superconductor has to be chosen and integrated into a conductor with design parameters chosen in order to avoid the conductor quenching under these disturbances. But, as demonstrated by ITER, there are still immaturities in our technical knowledge of important parts of the conductor and coil behavior.

Taking one example, there are three important design criteria: temperature margin, limiting current and stability margin, originally adopted for the design of ITER conductors. To allow these criteria to be exploited, in the real production process, quality control mainly focuses on how to control the superconducting performance with extrusion and heat treatment, and the superconducting stability of the conductor through the Cr or Ni coating on the surface of superconducting wires and copper wires [176]. Even nearing the end of ITER conductor manufacture, unanticipated problems were found with the conductor performance [177]. Following a comprehensive R&D program, a technical solution was found for the

ITER CS conductor, which ensures stable performance versus EM and thermal cycling, but was too late for the TF [15]. To solve the superconductor performance degradation after electromagnetic and thermal cycling, we have had to place emphasis on the degradation management with extended tests on conductor samples and revision of design criteria. But due to the limitations of knowledge, a full understanding of how to achieve controlled strain behavior of the filaments inside the jacket with the ideal undamaged curves of critical current variation with strain is still needed. Even if we developed novel Nb₃Sn technology using a short twist pitch of 45 mm in the CS conductor and a pseudo long twist pitch of 80 mm in the TF conductor, the low *n*-behavior (i.e. degraded) of Nb₃Sn conductors and its impact on the definition of the critical current and operational regime is still not clear. There is a weakness of repetitive quality control with qualified procedures in production, e.g. the key mechanical properties of Nb₃Sn strands are dominated by controlling the heat treatment (at 650 °C) and the subsequent cool-down to 4 K, but we do not understand fully what in this process affects the degradation. The situation has been controlled for ITER but is not mastered yet.

For a second example, the ways to make electrical insulation at a high voltage >10 kV needed for ITER and for the next step are uncertain and need new technology for use in a cryogenic vacuum. The design criteria and inspection processes are novel and weakly based on experience. All of these uncertainties and unexpected conditions in the design and production process may lead to the degradation of the mechanical, electromagnetic and superconducting performance of superconducting components in the actual operating environment. While tolerable for a research-based device such as ITER, with a long commissioning and non-nuclear phase of operation, this novelty would be difficult to accept for a nuclear fusion device.

As a third and final example, we find that specific solutions for the superconducting cable then produce difficulties in the related technology in other areas, with an example coming for the joints in the CS coil. The cable requires special process control on the Cr coating to allow reproducible current transfer between strands in composite cables, but developing the removal of this coating has created several non-conformities on CS joints and led to a late new design for CS coaxial joints at the coil terminals.

Lessons learnt for the next generation of DEMO are the critical importance of integrated engineering at all stages. We cannot treat a nuclear machine as a research project, improvising at each step. How to determine the acceptance criteria for the superconductor in the design phase, and how to ensure the quality control in the manufacturing process to verify the final superconductor performance is worth thinking about at the conceptual stage of the next step.

Current and future challenges

However, quality is planned, designed, and built in, not inspected in.

Compared with the uncertainty in the design stage, during ITER component design and manufacturing execution,

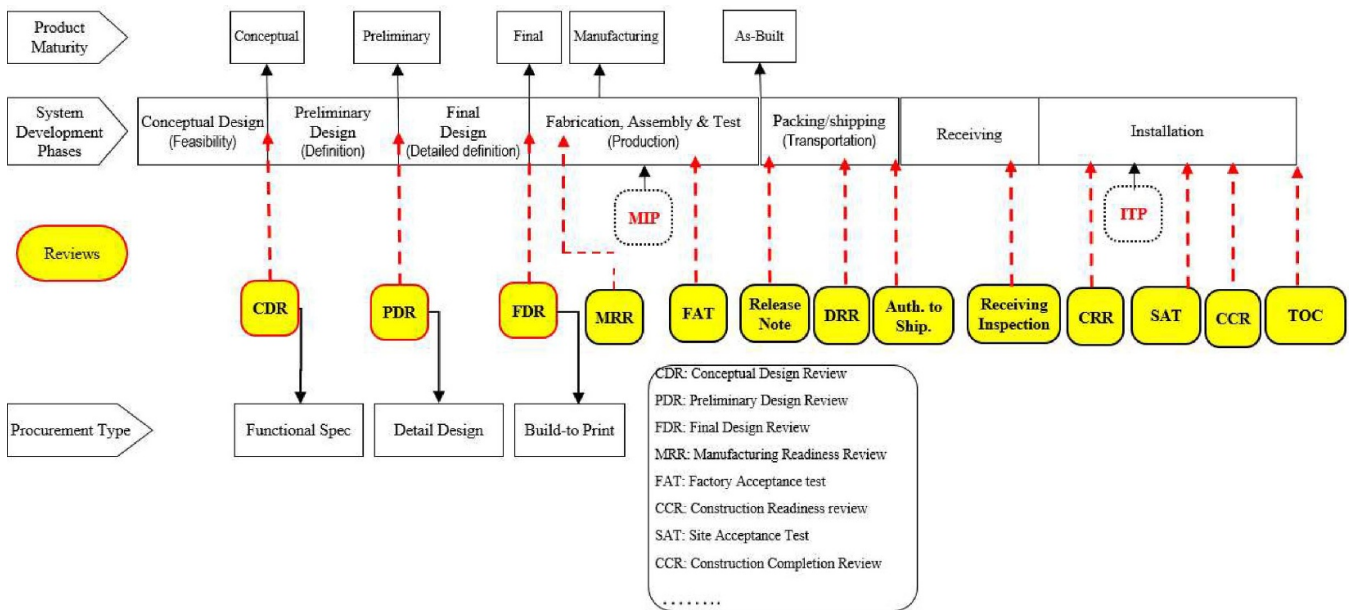


Figure 30. Gate reviews and control points for technical/quality control.

ITER has developed a series of mature quality assurance and traceable quality control methods to achieve the traceability during manufacturing and assembly. These include ITER management quality procedure gate views and manufacturing database (MD), a conductor database, deviation request/non-conformance report (NCR) databases, etc. The implementation of quality control and traceability management in the production process has well demonstrated how to guarantee the superconductors' stability and performance [178]. Due to the global decentralization of suppliers and the inconsistency of the implementation of codes and standards among worldwide suppliers, the management and traceability of special processes was particularly challenging for ITER and is a good demonstration of how the project environment can be a major consideration in the quality management processes. The ITER magnets are substantially an 'in-kind' contribution provided through 6 out of the 7 ITER partners, with a subdivision that had to reflect also the strategic interests of the partners to learn from the ITER procurement experience: ultimately successful, this system imposed multiple learning curves on the ITER organization and many extra interfaces. For the organization of future DEMO activities, the impact of the project environment on the authority of the central team to apply a strong centralized quality program must be an important consideration.

In order to monitor the progress of activities and control the quality in compliance with the defined requirements and/or technical requirements, in ITER, the Quality Supervision Plan defines review gates and control points during the whole life-cycle, as shown in figure 30. It includes not only the factory acceptance test (FAT) and site acceptance test but also for instance design reviews (CDR, PDR, FDR) and a delivery readiness review. For each control point/gate, we implement MD to keep track of the properties, reports and certificates associated with each step of the manufacturing process, as shown in figure 31. Technical information on raw materials,

records on procedures such as welding and testing, inspection certificates, final product documentation and shipment information CAN all be stored and accessed in the database. For example, with the implementation of the conductor database, during the 4 years of the conductor production period, the ITER Organisation (IO) has cleared ~6900 control points for the strand lots, and ~27 000 critical measurements are well monitored and qualified. It covered the production for 600 t of Nb₃Sn strands for the TF and CS coils, while it needed around 275 t of Nb–Ti strands for the PF and CC and bus bar conductors.

To achieve continuous improvement and standardize an approach to non-conformances that is accepted over a wide range of company and national systems, ITER implemented an NCR database in December 2018. The database can automatically follow up the NCR workflow by displaying the full NCR stage identified in the NCR procedure, and clearly identify the root cause analysis, remedial action and corrective action implementation with their associated evidence. Reviewing the statistics of NCRs for magnets on conductor and superconducting coils since 2017, as shown in figure 32, there were 793 NCRs in total, including 584 major NCRs which mostly happened in sensitive quality areas such as insulation, He leaks, welding, and in deviations from special process control [179].

A common technical root cause in these NCRs is failure near HV instrumentation wire exits from the ground insulation found during FAT tests. These are often related to local hand wrapping of Polyimide and glass tape to form the ground insulation in the region, with resin contamination of the sensitive instrumentation wires. Recurrent NCRs are typically resolved with more attention in the design, application procedure and worker qualification.

Taking a general review of the problems and production risk related to the industrial fabrication of prototypes and the whole

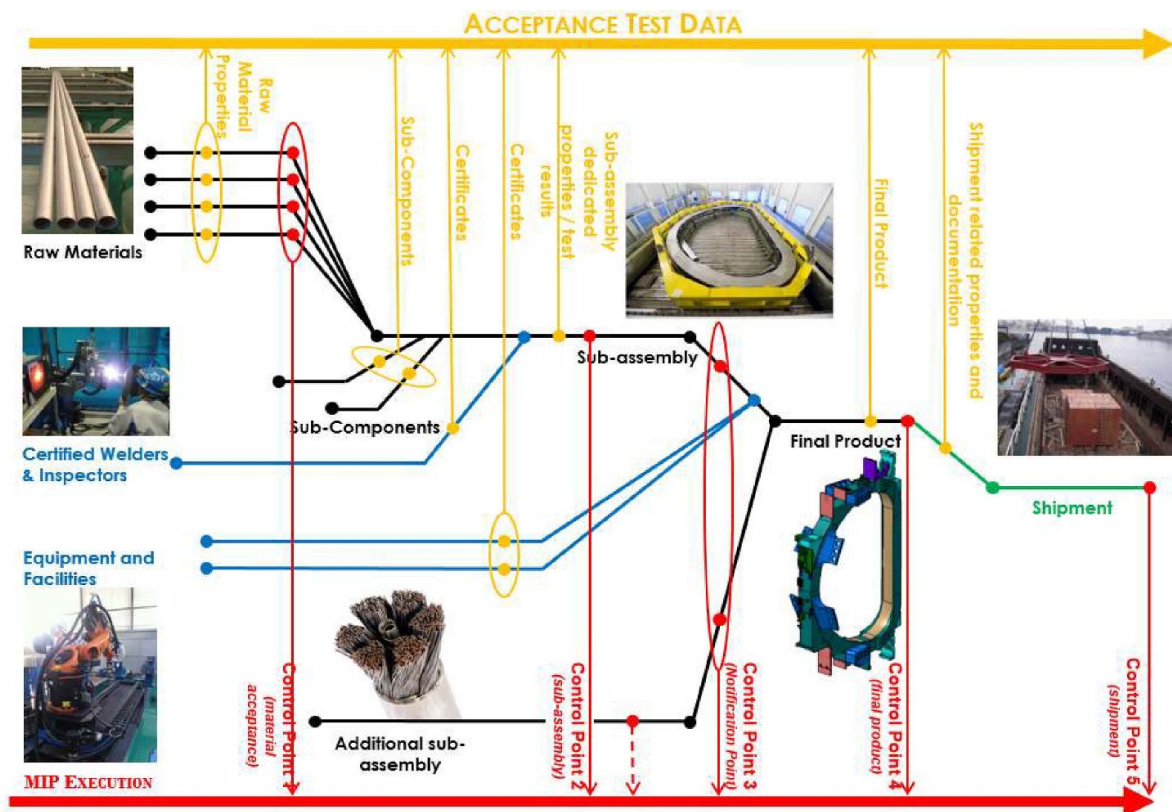


Figure 31. The implementation of manufacturing database in ITER conductor coil production.

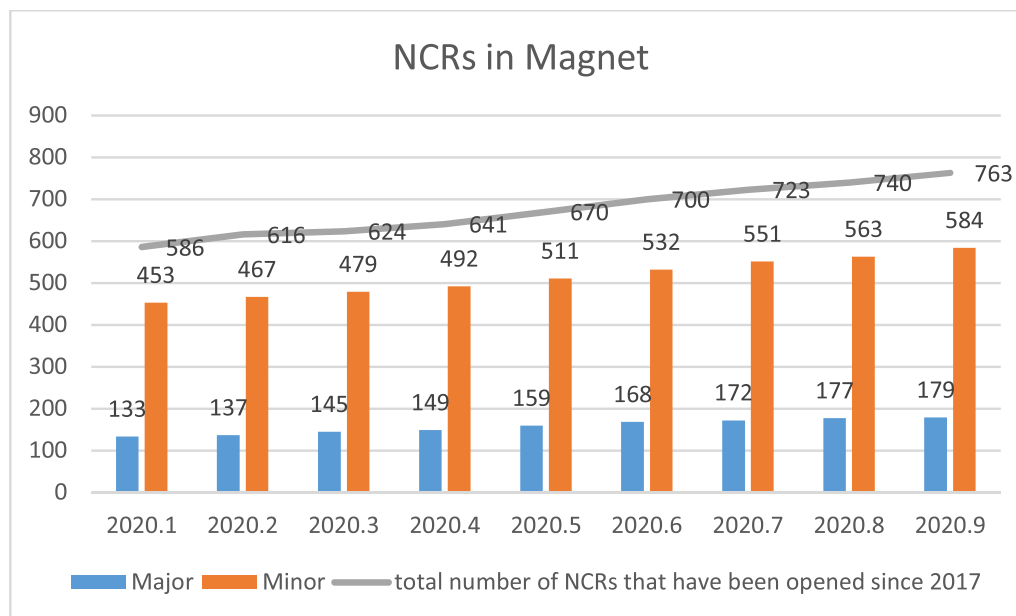


Figure 32. Statistics on NCRs for magnets since 2017, with major and minor NCR numbers.

series of superconducting magnets, the main quality issues are He leaks, HV wires cracks, welding defects and insulation failure by the Hi-pot or Paschen test. New QC methods need to be developed, as well as improved follow-up/testing during manufacturing. For example, maintaining good bonding between the resin and kapton to avoid the creation of an electron path

along the HV wire after cold tests, reinforcing quality inspection during handling in winding impregnation, remove the gap inside or resin flow down from wire extraction through feed-through. The key is to avoid the weakness of too much flexibility on qualified procedures (i.e. freedom for worker interpretation), and to choose a design and manufacturing route which

permits clear quantifiable quality controls that can be implemented easily. It is notable here that the difficulties are not in the ‘bulk’ insulation systems but are related to a particular aspect of superconductivity—the HV quench detection wires.

Advances in science and technology to meet challenges

Looking back to what we have experienced in ITER, the most important issue in the quality field is to have set up the quality policy for a nuclear environment and to have ensured the achievement of quality awareness training within the whole staff. It is the fundamental step to improve staff professional qualifications and competence and promote creative approaches among employees in order to attain quality performance and environmental awareness. For example, during qualification of the superconductor’s strand fabrication and test phase, the following main technical activities were developed into a mature quality control approach: the qualification and certification of manufacturing and test procedures (e.g. orbital welding of jacket sections, local and global He leak tests); statistical process control on critical parameters; and benchmarking of cryogenic test facilities. We managed SULTAN [15, 177] tests for the conductor, and set up a huge strand testing activity during conductor production. This could be an example of what will be required for a future DEMO.

For an ITER-type DEMO such as CFETR, we can learn from tracing the NCRs in ITER where we have technology problems. These could be solved by improving the technology, although clearly better quality control is also needed to reduce variability in the processes and to allow better testing. Design for testability needs to be a basic design consideration. Large-scale testing of complete sub-units (or possibly complete coils) including operational loads (4 K, with current) seems essential to confirm the electrical quality.

There are examples in ITER of poor design and difficulties in manufacturing quality control in areas such as uncontrollable weld configurations, uncleanable cryogenic pipes contaminated by metallic debris and high-voltage faults, all of which are being detected during FAT tests before delivery. Detection at this point means it is often too late for corrective actions to be taken. This represents immature systems engineering, and improvements are needed in the future with earlier testing and inspection, and selection of manufacturing routes that allow intermediate tests.

ITER experience shows that early electrical tests on sub-components are particularly valuable. Failures may be repairable and there is easy access. For example, it is possible to apply voltages across different insulated surfaces (i.e. between the conductor, radial plates and quench detection strip in the ITER TF coils). Thermal cycles at 77 K can provide differential thermal contraction as a limited form of proof tests. To avoid the difficulty in process control on hand fabrication methods, we need to improve the manufacturing design reproducibility. Hand fabrication in critical areas gives rise to

variability should be avoided by design, with factory fabrication and testing before mounting on the component.

In conclusion, we can develop test guidelines for superconductors and other special components, a specific intermediate test base on the manufacturing routes, and the status of special process control after component delivery. The difficulty is mainly reflected in the size limitations of test equipment to match the huge superconductors. Moreover, some test methods risk causing damage to the components, such as the cold test, high-voltage test and Paschen test, as well as structural loads. The tests themselves can be special processes and we have experience in ITER where incorrect application of procedures or apparatus can damage the component, even if not damaged at the start.

There has been remarkable progress from the ITER program in developing and applying quality assurance, with many lessons learned, and a firm footing for DEMO QA/QC has become available if the criticality is recognized. The main lesson is that this needs to be implemented in the development stage, for example during the manufacture and operation of test coils, where it might be thought of as excessive and unnecessary. Even if test coils could be easily repaired, the experience provides DEMO (large or small) QA/QC with a robust quality background. For new technology for nuclear superconductors, we need to put quality in at the start, and structure the design, manufacturing, inspection and testing around this.

Concluding remarks

Many of the quality issues found and resolved in the ITER magnet activity need to be turned into lessons learnt for the next steps of fusion. ITER will undergo an extended period of testing/commissioning and an R&D phase before starting nuclear operation lasting several years, and essentially this will provide confidence in the superconducting magnet performance being adequate to achieve steady nuclear operation. But if problems are found in this phase, they will be expensive and time-consuming to fix and will transmit a negative message about the viability of DEMOs. We need to do better for the next step.

In summary, due to the newness, uncertainty and technical limitations, confidence in magnet systems’ performance in the nuclear environment is at present lacking under complex working conditions. Therefore, we need to find a way to develop and qualify the design criteria for the superconducting performance, verify the reliability in the production process, develop a series of quality control tests, including superconducting performance tests, and finally set up the criteria as final acceptance to ensure superconductor quality and their use in future DEMO reactors. Detailed main options for quality assurance during manufacturing, including items with a high cost and schedule impact such as insulation, tolerance assessment on site, and coil cold tests need to be discussed in the future as an integrated part of the fusion roadmap.

Data availability statement

No new data were created or analyzed in this study.

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Disclaimer

The views and opinions expressed herein do not necessarily reflect those of the ITER organization.

ORCID iDs

Neil Mitchell  <https://orcid.org/0000-0002-7980-5340>
 Jinxing Zheng  <https://orcid.org/0000-0003-4624-0571>
 Christian Vorpahl  <https://orcid.org/0000-0002-5382-253X>
 Valentina Corato  <https://orcid.org/0000-0002-3314-0275>
 Charlie Sanabria  <https://orcid.org/0000-0001-5017-5309>
 Brandon Sorbom  <https://orcid.org/0000-0002-2110-6766>
 Robert Slade  <https://orcid.org/0000-0001-9377-386X>
 Nobuya Banno  <https://orcid.org/0000-0002-7141-541X>
 Anna Kario  <https://orcid.org/0000-0002-7241-3707>
 Herman Ten Kate  <https://orcid.org/0000-0001-5597-3190>
 Pierluigi Bruzzone  <https://orcid.org/0000-0002-2184-5294>
 Nikolay Bykovskiy  <https://orcid.org/0000-0003-2009-3112>
 Matthias Mentink  <https://orcid.org/0000-0001-9769-0578>
 Franco Julio Mangiarotti  <https://orcid.org/0000-0001-8299-0711>
 Kamil Sedlak  <https://orcid.org/0000-0002-2090-1709>
 Danko C Van Der Laan  <https://orcid.org/0000-0001-5889-3751>
 Min Liao  <https://orcid.org/0000-0001-6063-9365>

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