HTS Cable Conductor for Compact Fusion Tokamak Solenoids

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***Abstract*—Significant progress has been made recently in the US fusion community to develop a strategic plan to enable engineering design and construction of a fusion pilot plant (FPP). Princeton Plasma Physics Laboratory (PPPL) is working on developing high current density HTS conductors for next fusion experiments. Partnering with the US industry, we are evaluating feasibility and affordability of Conductor on Round Core (CORC®) developed by Advanced Conductor Technologies (ACT) for the next compact fusion tokamak facility. High current density achieved by a CORC® cable based on a four-layer model coil recently tested at National High Magnetic Field Laboratory (NHMFL) motivated its consideration for low cost, reduced size fusion magnet application. This is of interest to PPPL because of its scalability to tokamak central solenoid (CS) coils in terms of required flux swings for plasma startup operations. Partnering with ACT, we designed and built a two-layer model coil solenoid directly wound with CORC to demonstrate its applicability for compact solenoids such as that used in the national spherical torus experiment (NSTX), NSTX-upgrade (NSTX-U) and the US sustained high-power density test facility (SHPD). The ∼160 mm diameter solenoid wound by a two-layer CORC is being tested in early 2022 under electromagnetic cyclic loading at NHMFL in a unique 160 mm bore, 14 T background field magnet facility.**

***Index Terms*—Compact tokamak test facility, fusion magnet design, high temperature superconducting magnets.**

I. Introduction

A FUSION pilot plant (FPP) is part of the US next step fusion development strategy [1] to produce net electricity from fusion and establish capability of high average power output. Recent studies of US sustained high power density (SHPD) facilities show that reduced-cost compact tokamak FPP may be feasible if novel magnet engineering and system integration challenges of high field, high current density can be addressed [1]–[6].

The US SHPD is a low aspect ratio spherical tokamak (ST) with a reduced magnet volume and operates in a steady state mode with long plasma pulses and possible high duty cycles. Its in-vessel components such as the first wall, blanket, shielding, vacuum vessel and the diverter will be exposed to the extreme fusion neutron producing plasma. Next fusion confinement facilities need high current density cables of coil winding in a smaller design space (radial build) compared to ITER and DEMO to support and potentially expand confinement integration experiment missions and serve as a FPP prototype [5]–[12]. Previous design studies [2]–[5] identified that high field and high current density magnets are particularly beneficial for low-aspect ratio STs such as the SHPD. High temperature superconducting (HTS) cables such as Conductor on Round Core (CORC) impact toroidal field (TF) and central solenoid (CS) magnet design for the FPP by providing higher flexibility and higher winding pack current density due to its small bending radius available for compact in-board radial build, isotropic properties and large irreversible strain limits [13]. CORC is one of the most studied HTS cables. High current density achieved by CORC in a 4-layer model coil [14], recently tested at national high magnetic field laboratory (NHMFL) in a 14 T background field magnet motivated its consideration for high field, small diameter CS for the next compact tokamak test facilities. Fig. 1 presents a cross-sectional view of national spherical tokamak experiment upgrade (NSTX-U) [2], a flagship machine constructed and operated at the Princeton Plasma Physics Laboratory (PPPL); next-step intermediate scale facilities such as the SHPD and low-aspect ratio tokamak FPP. If high performance of CORC is proven feasible for next compact tokamaks, a stand-alone HTS CS, also called ohmic heating (OH) coil, can be designed and built for a replacement TF-OH bundle for NSTX-U where the present OH coil is trapped onto the outer surface of the inner TF. Although the present bundle is demountable from machine center stack casing assembly, the trapped OH design demands high reliability for operation due to its inaccessibility for inspection and maintenance. It is thus a lost advantage if the TF-OH bundle has performance issues, of replacing only the OH coil which motivated the original design for both the NSTX-U and MAST Upgrade at U.K. [7].

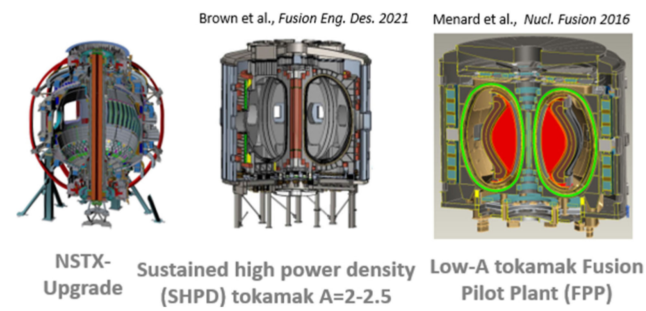

Fig. 1. Innovative next-step intermediate scale facilities beyond NSTX-U.

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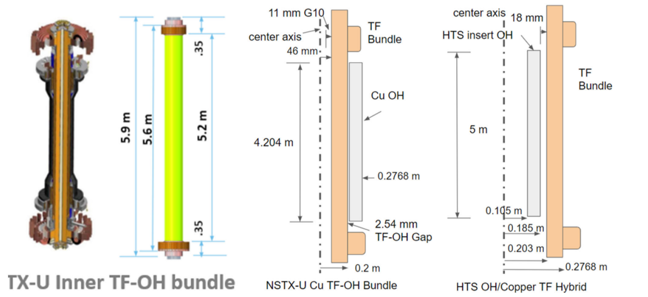

Fig. 2. NSTX-U TF-OH bundle (left) and radial build (middle), a new hybrid bundle of stand-alone HTS OH and Cu TF (right).

II. Physics Requirements

A. High Field Compact Tokamak Parameters

NSTX-U is a US national fusion experiment facility for the study of plasma confinement, heating and current drive in a low aspect ratio, spherical torus (ST) configuration. Although compact STs such as the NSTX-U/MAST provide increased plasma performance including stronger shaping, confinement and a higher stability limit, it posed significant engineering challenges for machine design, construction and assembly due to tight space in the TF-OH in-board leg radial build. The OH coil for NSTX-U is wrapped around the inner TF leg to obtain a sufficient winding diameter to provide required flux swings. In this paper, a stand-alone CS using CORC is presented to decouple OH from inner TF to meet flux swing requirements in a compact radial build for NSTX-U and SHPD [2]–[6].

The design of large-scale fusion magnets requires high current cables such as the cable-in-conduit conductor (CICC) adopted by ITER using NbTi and Nb_3Sn low temperature superconductors (LTS) [10], [11]. The achievable current density in stainless steel (SS) jacketed CICCs is typically less than 50 A/mm^2. A higher current density of 80 A/mm^2 in the winding pack is needed for CS operations in a compact inboard radial build [6], [9], [12]. HTS can provide both the high field and high current density. The total flux swing required for NSTX-U is $\sim$1 Wb at the end of the plasma ramp. For the next step ST, the max magnetic field on the midplane of SHPD CS coil is 16-20 T at the end of a plasma ramp. The required OH double flux swing is $\sim$3.3 Wb, from -1.7 Wb back bias to +1.6 Wb at the end of the plasma current ramp. Fig. 2 presents the TF-OH bundle for NSTX-U where OH is trapped on the outer surface of the TF bundle. It is advantageous to have a stand-alone OH inside the inner TF so it can be removed for replacement and maintenance. It is also much easier for reassembly whenever replacement coils are needed.

B. High Field CS and PF Coil Currents

The present copper OH coil for NSTX-U is centered at 0.24 meter radius in the radial build, 2.1 meter above/below the midplane with a total height of 4.2 meters. It is composed of 4 layers of copper conductor totaling 888 turns. The width and height of OH conductor is 15.5 mm and 16.8 mm respectively, with a 5.7 mm diameter center cooling hole. Its turn insulation includes 2 co-wound layers of 0.006" glass/0.002" Kapton, plus one half-lap layer of 0.006" glass for a total thickness of 0.048". The ground wrap outside OH winding pack includes 6 half-lapped layers of 0.01" thick and 2" wide s-2 glass to give a total thickness of 0.108". The hybrid TF-OH design has a smaller CS insert of an inner radius of 0.105 meter and an outer radius of 0.185 meter. It has a winding pack density of 152 A/mm^2 using 7.5 mm diameter CORC.

TABLE I
CS-PF Coil Currents (MA) and Winding Current Density (A/mm^2)

	NSTX-U		SHPD	
I_{plasma} (MA)	2		4.5	
Flux (Wb)	1		5.4	
Coils	I_{op} (MA)	J_{op} (A/mm^2)	I_{op} (MA)	J_{op} (A/mm^2)
CS Conductor	0.024^{+}/0.01	102.6^{+}/156.3	0.0107	212
CS	21.3^{+}/60.7*	72.0^{+}/152*	23.9*	109*
PF1a	1.16	50.4	1.71	25.1
PF1b	0.26	59.5	2.2	46.3
PF1c	0.26	43.9	1.84	47.1
PF2	0.42	19.0	0.156	5.40
PF3	-0.36	-14.2	-0.94	-15.7
PF4	-0.272	-9.7	-0.38	-6.1
PF5	-0.816	-44.0	-1.46	-20.0

*High current density HTS coils using CORC; +existing copper coil.

The CS coil for SHPD is centered at 0.26 meter radius, 1.6 meter above/below the midplane with a total height of $\sim$3.2 meters. The maximum coil currents occur at back bias and at the end of the plasma current ramp up. NSTX-U is designed to meet a total number of 20000 cycles based on shot spectrum. A few thousand cycles fatigue life may be required for SHPD CS coils since it is likely to be operated for long pulses or steady state plasma runs. Table I presents the maximum CS-PF coil currents in MA-turns from back bias when all coils are in steady state for plasma equilibrium; these currents are coil maximum values for NSTX-U and SHPD (a 12 layer design provides 5.4 Wb double flux swing).

Fig. 3 presents the magnetic field distribution in the SHPD CS and PF coils from coil currents at the back bias prior to plasma startup. The max field of 16-20 T on OH is needed for SHPD as compared to 6.5 T on a copper OH for NSTX-U.

III. Central Solenoid Design

A. High Temperature Superconductors

The HTS considered includes 7.5 mm diameter CORC by Advanced Conductor Technologies (ACT) to provide high current density in rapidly mature winding technology for coil fabrication. Fig. 4 presents critical current (I_c) of 4 mm wide SuperPower tapes as a function of magnetic fields.

A CORC of 7.5 mm diameter, composed of $\sim$50 YBCO tapes, has an operating current of $\sim$10.7 kA at 4.2 K, 20 T. Fig. 5 shows estimated I_c (70%) and a short CORC sample. Either a radial support plate with grooves for the cable or a square jacket for CORC may be needed in this no organic insulation and no vacuum pressure impregnation (VPI) design.

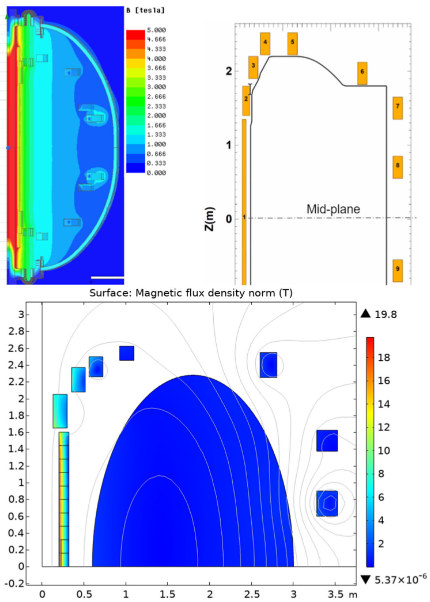

Fig. 3. The B field from NSTX-U (top left) and SHPD (top right and bottom) CS-PF currents at the back bias.

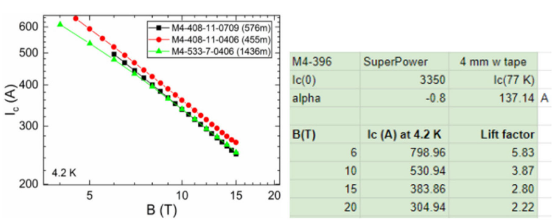

Fig. 4. The magnetic field dependence of Ic at 4.2 K of SuperPower tapes from batches used in the CORC cable. The magnetic field was applied perpendicular to the tape surface.

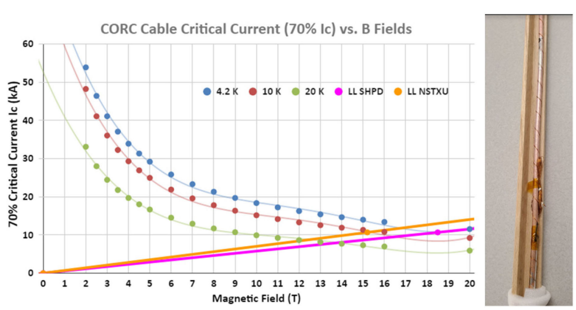

Fig. 5. Critical current (70% I_c) with load lines; CORC by ACT [12].

BSCCO conductors such as Bi-2223 and Bi-2212 were also considered and evaluation for fusion magnets was performed to identify the potential cost and AC loss advantages for fast ramp OH operations [15]. Results were compared in terms of achievable winding current density, mechanical properties, AC losses and cost but further investigation to quantify differences is needed. Unlike 2G HTS, the lack of alternative source of BSCCO suppliers may limit its present use on a large scale.

Although YBCO may have two orders of magnitude higher AC losses than Nb_3Sn and NbTi LTS conductor used in ITER, thermal stability of YBCO has an order of magnitude higher margin in meeting Stekly stability criterion. A CORC cable cooled by liquid helium at 4.2 K has a significant margin for thermal stability and reasonable current transfer capabilities among tapes within the cable.

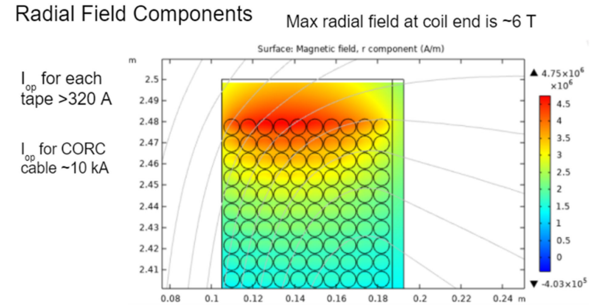

Fig. 6. Radial field distribution at OH coil ends.

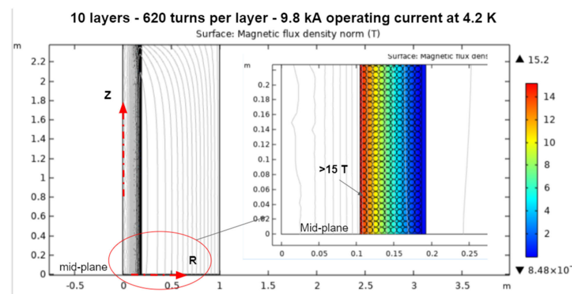

Fig. 7. The CS magnetic field at the end of plasma ramp.

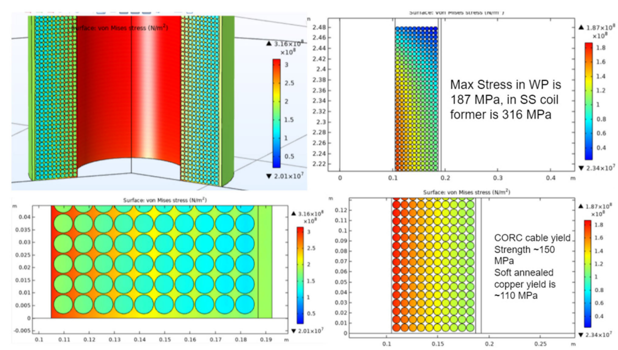

Fig. 8. Stress distribution in the CS at end of plasma ramp.

B. HTS OH Coil for NSTX-U

CORC is used in a new CS winding design for NSTX-U where a smaller OH insert is decoupled from the TF bundle as shown in Fig. 2. To provide the same flux swing, the OH of 0.2 m ID and 0.37 m OD, 5 meters tall shall generate 15 T on the coil. This is 2.5 times higher than that generated by the present copper OH during plasma operation. A winding pack of 10 layers and 620 turns per layer is developed. Fig. 6 presents radial field distribution at coil ends from current back bias from a 2D COMSOL analysis model (bonded). Figs. 7–9 present the B field and winding pack stress distribution. The max stress in the winding pack is 187 MPa, exceeding the yield strength of annealed copper. CORC cable, however, has a high irreversible strain limit. The stress in the stainless steel (SS) support structure is 316 MPa which is well below the 660 MPa design stress limit for SS316.

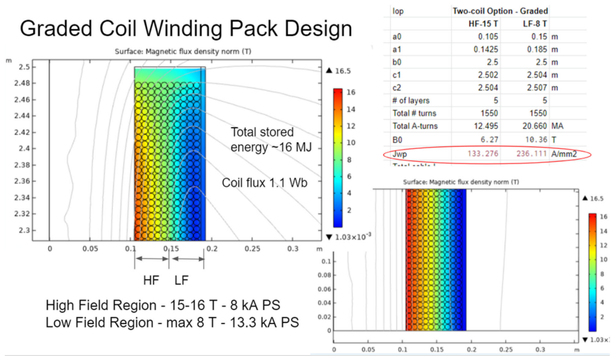

Fig. 9. A graded CS (high field and low field regions) at current back bias.

TABLE II
NSTX-U AND SHPD CS COIL PARAMETERS

	NSTXU Cu CS	NSTXU HTS CS	SHPD HTS CS
A-turns(MA)	21.3	33	41
J_{wp} (A/mm^2)	72.0	152	109
Max B (T)	6.5	15.0	16.0
Inductance (mH)	41	492	971
Stored Energy (MJ)	11.8	23.6	55.3

The insert has a 150 A/mm^2 winding pack current density. The coil inductance is $\sim$0.5 H and the total stored energy is 24 MJ as shown in Table II. The total cable length is $\sim$6 km and the total tape length is 450 km for a 50-tape cable, which gives an estimated conductor cost of \$10-20M, assuming the cost is driven by the estimated tape cost at this total length scale. A total expected coil cost is $\sim$\$20-30M.

The 6 km length of cable could be delivered in 2-3 years based on a 200 km/year tape delivery. Total number of joints depends on the single tape piece length. This design will likely have 30-40 coil modules, which implies a schedule of 3 years away from building the coil. A graded coil design of high field (15 T) and low field (8 T) can provide over 230 A/mm^2 of current density in the low field region as shown in Fig. 9. It also provides a more uniform stress distribution.

C. SHPD OH Coil

The CORC based OH design for SHPD [4], [5] has 12 layers and total 320 turns of 10.7 kA operating current at 4.2 K. The winding pack current density is 109 A/mm^2, the total 50-tape cable length is about 10 km, and the total length of YBCO tape needed is about 750 km. The conductor cost is estimated to be \$15-30M. The total cost for coil fabrication is estimated at \$30-60M. A 2D axis-symmetric finite element model was built in COMSOL Multiphysics with coil currents from Table I. The cable is assumed bonded to the coil former and winding pack structure. Fig. 10 presents the B field distribution and Fig. 11 presents the winding pack stress distribution. The max von-Mises stress on the conductor is about 250 MPa and stress on the stainless steel coil winding support is 400 MPa which is well below the 660 MPa design allowable for SS316.

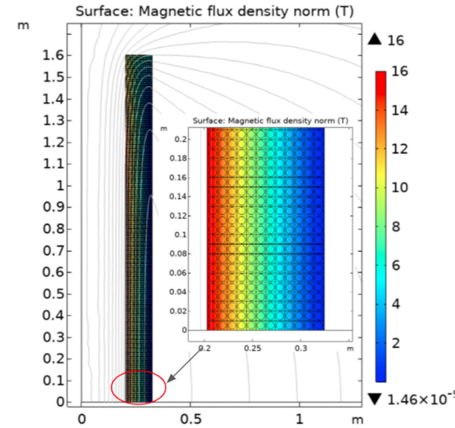

Fig. 10. Coil magnetic field distribution at current back bias (max 16 T).

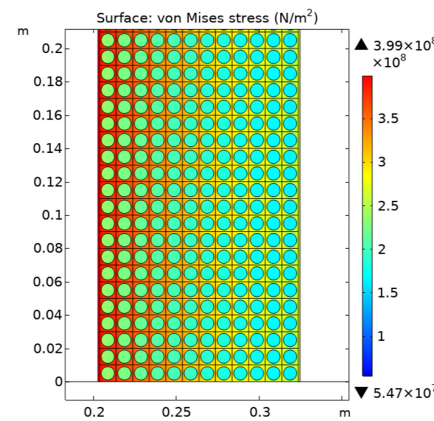

Fig. 11. The CS stress distribution at current back bias (max 400 MPa).

Qualification of a coil insulation system is costly but critical in a conventional fusion magnet such as that adopted by ITER and NSTX-U [16], [17]. Many LTS and copper coil failures found the source of coil insulation issues including quality and reliability of the vacuum pressure impregnation (VPI) [18], [19]. Radiation tolerance is also critical in the selection of organic insulation for magnets in next step fusion devices [20], [21]. Due to its higher temperature and energy margin in HTS, no insulation (NI) coils for quench protection are becoming increasingly attractive for high field fusion magnets. High stress and strain limits of CORC [13] motivated a two-layer model coil design with no organic insulation and simpler coil fabrication where no epoxy resin and VPI is needed. The CORC OH supports longer pulse operations and fast charging without quenching.

IV. TWO LAYER CORC MODEL COIL

A 15.2 cm diameter model coil solenoid using a 6 mm diameter CORC cable, consisting of 16 SuperPower YBCO tapes was developed at ACT and PPPL. The 2 layer, 12 turn HTS model coil shown in Fig. 12 wound at ACT was tested in liquid nitrogen and helium under its self-field in early 2022. A critical current of 1200 A was extracted for liquid nitrogen condition and $\sim$10 kA in liquid helium under self-fields. The coil was assembled as an insert at NHMFL in early 2022 for a cyclic load fatigue testing in the 16 cm large bore, 14 T high field background field magnet to evaluate the cable fatigue performance. This is the first fusion HTS model coil test designed based on relevant parameters for

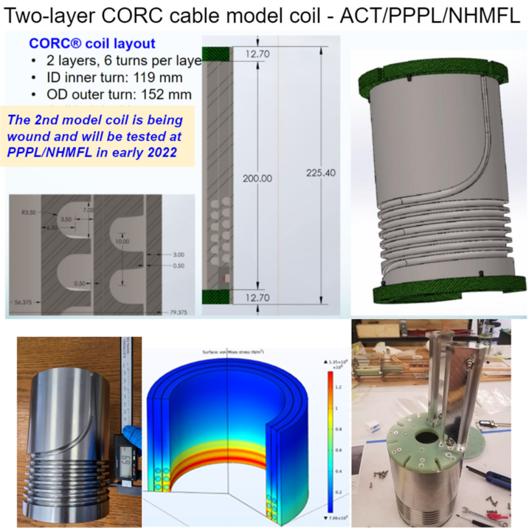

Fig. 12. The two layer CORC cable model coil solenoid for OH.

the next step ST OH operations. Results of this test will be informative to the NSTX-U and a FPP-like test facility such as the SHPD.

V. Conclusion

The HTS conductor is considered for next step confinement experiments (both advanced tokamak and spherical tokamak) with cables composed of REBCO tape such as the Conductor on Round Core (CORC). PPPL is working with US industry in the US fusion community to address technical challenges of high field CS and TF magnets for fusion pilot plant (FPP). A new design of all metal (no organic insulation and no-VPI) central solenoid was presented for both NSTX-U and SHPD. The magnetic and structural analysis performed showed that it is feasible to meet the flux swing requirements in the radial build. Detailed cooling design and quench analysis is being investigated at the system integration level in preparation for FPP-relevant coil testing. A two-layer model coil was built and is being tested in a high background field to validate HTS cable maturity and identify potential static and cyclic fatigue loading issues for the next ST CS coil operations.

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