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Conceptual design of HTS magnets for fusion nuclear science facility

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ABSTRACT

Second-generation high temperature superconductors (HTS) are available for producing >20 T at the magnet bore compared to 13–16 T for lower temperature superconducting (LTS) toroidal field magnets proposed in recent fusion energy systems studies (FESS) of Fusion Nuclear Science Facility (FNSF). HTS may enable higher fusion power density and smaller device size. High current density cables of multi-layered REBCO tapes have achieved >10 kA at 4–20 K operation in short sample tests for fusion. High current density cables are required for engineering design of FNSF to allow space for interior plasma components. High current density HTS magnets are particularly attractive in reducing the size of a fusion device, beneficial for compact tokamaks, due to their space constraints. Successful HTS magnet development may enable the design of smaller and cheaper fusion pilot plants with a mission of demonstrating net electricity. It may also offer significant cost and performance advantages in non-fusion applicants such as nuclear magnetic resonance (NMR) and magnetic resonance imaging (MRI). We developed HTS magnet design concepts for a compact FNSF radial build in order to define the coil size, winding pack mechanical loading and engineering requirements. Partnering with vendors in the US, PPPL is also testing high current cable prototypes aiming at enabling low cost cable technology toward 100 A/mm^2 engineering current density over the winding pack desired in high field model coil development for compact fusion pilot plants.

1. Introduction

Fusion Nuclear Science Facility (FNSF) is part of the US effort to bridge science and technology gaps between ITER, currently under construction in south of France, and fusion power pilot plants. FNSF accommodates an extreme fusion neutron environment for operating at a significantly higher neutron fluence over the machine lifetime, and a smaller size for access to higher magnetic fields of longer pulses than ITER. FNSF studies address complex engineering issues of integrating fusion nuclear components into an extreme neutron environment, and focus on various nuclear material aspects of power plant relevant interactions in steady state operations [1].

Toroidal field magnet design using low temperature superconductors (LTS) for FNSF has been developed early in fusion energy systems studies (FESS) using a baseline design point of 7.5 T field at the plasma center [2,3]. Fig. 1 presents a cross sectional view of FNSF.

High Temperature superconductor (HTS) has been identified in ARIES-AT and in a recent community plan for fusion energy [4,5] as the potential game changer for enabling higher fusion power density and a smaller device size for FNSF and future fusion pilot plants (FPP). Enabling high field HTS magnets is one of the critical challenges for advanced tokamaks [5] such as FNSF in optimizing the machine size and system performance. PPPL is partnering with US industry in the magnet community to identify issues in a pre-conceptual design of high field HTS magnets for FNSF. A R&D effort is a prerequisite to its design.

HTS cables of multi-layered REBCO tapes have achieved over 10 kA current at 4–20 K operation in short sample tests for fusion [6–8]. High current density cables are required for engineering design of the compact FNSF and FPP to allow space for interior plasma components [2]. A Conductor on Round Core (CORC®) cable insert solenoid was recently tested at currents exceeding 4 kA in the 14 T background magnetic field [9]. Despite recent progress on HTS cables, quench protection and winding pack stress management remain to be outstanding issues for large scale applications for fusion.

Guided by systems code analysis, we developed a conceptual toroidal field (TF) magnet design using HTS technology for the FNSF to access a target of 9 T higher field on the plasma center, so to demonstrate potential opportunities to de-risk pilot plants in a compact radial build for

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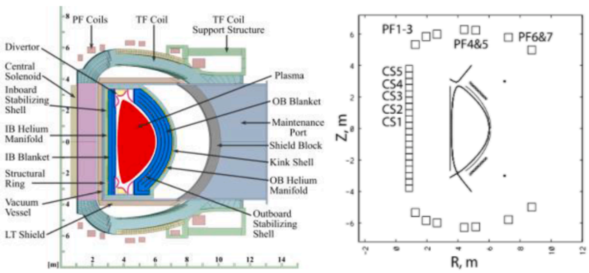

Fig. 1. Cross sectional view of fusion nuclear science facility.

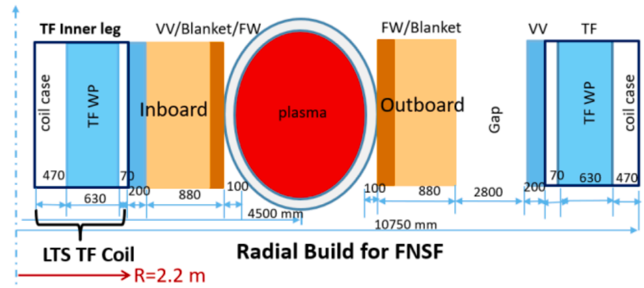

Fig. 2. The FNSF radial build [1–3].

the FNSF. We report what HTS can offer for the FPP in terms of defining a coil winding pack size, mechanical loading and TF magnet engineering design requirements. Results of the HTS magnet design are compared with LTS design in terms of radial build space, winding pack current density, coil cooling and quench protections. Neutron radiation affects performance of superconductor, copper stabilizer and coil insulation [10–13]. Neutron radiation also creates activation issues discussed in this paper.

2. System design parameters

21. In-board radial build

FNSF is smaller in size than a DEMO or ITER but it generates a higher toroidal magnetic field for plasma confinement [14–16]. It will be operated at thirty times higher neutron fluence with three orders of magnitude longer plasma duration at higher operating temperatures for structures surrounding the plasma [1–3]. A baseline system design point was chosen for the detailed magnet system analysis with $R = 4.8$ m, $a = 1.2$ m, $I_p = 7.9$ MA, $B_t = 9$ T, normalized beta < 2.7, $n/N_{Gr} = 0.9$, $F_{BS} = 0.52$, $q_{95} = 6.0$, $H_{98} = 1.0$ and $Q = 4.0$. Table 1 presents the FNSF system design point parameters. For TF operation, high current cables are needed to lower inductance for coil protection during fast discharges. Selection of fast discharge time constant is balanced between magnetic loads induced on the vessel and maximum temperature rise during fast discharges. Fig. 2 shows the FNSF radial build from systems code studies.

2.2. High temperature superconductors

The REBCO coated conductor is now commercially available in long length and it is becoming increasingly mature and attractive for high field applications [17–22]. Tremendous progress has been made over the past years on the improvement of REBCO tape critical current performance. The recently measured tape (*SuperPower M4-396*) performance has a minimum critical current of I_c (77 K, s.f.) = 137 A and a minimum I_c of 800 A at 4.2 K, 6 T [7,9]. Potential issues with the REBCO tape include its I_c angular dependence [19] or performance anisotropy, transverse load effects impose a ~200 MPa compressive stress limit on tapes without I_c performance degradation [21], delaminations of the tape at high fields due to screening current and the non-uniform current distribution [6], as well as challenges in implementing an effective cooling design. Unlike the brittle Nb_3Sn wire, coated conductor has a much higher axial tensile strength that makes it attractive for high field fusion applications.

Table 1
FNSF magnet system design point parameters.

Conductor	ITER LTS	LTS	FNSF YBCO	YBCO
	Nb3Sn	OST RRP	6-around-1	CORC®
R0	6.2	4.8	4.8	4.8
a (m)	2	1.2	1.2	1.2
Ip (MA)	15	8	8	8
B0 (T)	5.3	7.5	9.3	9.0
Iop (kA)	68	63	64.0	10.7
Bmax (T)	11.8	16.5	20.3	19.7
JWP (A/mm2)	17	27	39.1	44.8
A-turns (MA)	9.1	11.3	14.0	13.5
# turns	134	180	218	1268
# TF coils	18	16	16	16
Discharge volt (kV)	10	15	15	
Dis. time const (*s*)	11	12	12	
Fusion power (MW)	500	450	450	450

2.3. High current density cables

High current superconducting cables used for fusion reduce coil inductance and ensure that large TF magnets are protected during quench [6–8]. FNSF TF coil design used CORC® cables developed by Advanced Conductor Technologies (ACT). The cable design being considered includes a cable wound from 50 tapes of 4 mm width, containing 50 μm substrates for the High Field (HF) region; a cable wound of 38 tapes for the Middle Field (MF) region, and a cable consisted of 24 tapes for the Lower Field (LF) region. The CORC® cable performance is scaled for an operating current of I_{op} (70% I_c) = 10.67 kA at 4.2 K, 20 T. The overall cable current density is over 150 A/mm^2, over three times higher than ITER TF Nb_3Sn cable current density. A 6-around-1 CORC® cable-in-conduit conductor (CICC) is also considered for the TF coil winding pack to achieve a performance of 64 kA at 4.2 K, 20 T. Table 2 presents CORC® cable parameters used for TF coils. Fig. 3 presents single CORC® CICCs operated at 10.7 kA and a 6-around-1 CORC® CICC of over 64 kA. The slotted copper cable support structure provides better support and protection.

Figs. 4 and 5 present an 8 mm diameter single cable performance and a CORC® cable and wire test sample. The 5 mm diameter copper core used is based on the minimum tape bend radius. The single CORC® cable current density at 4.2 K, 20 T is over 210 A/mm^2.

The minimum bend radius for single CORC® cables is less than 200 mm which is much lower than the inner corner radius of the TF coil inboard transition (>1 m).

2.4. FNSF magnet system

FNSF magnet system consists of sixteen TF coils wedged together at in-board legs, a central solenoid (CS) to generate a required 100 Weber double flux swing and seven sets of poloidal field coils [2,23]. Unlike ITER, the mission of FNSF is focused on the fusion nuclear aspects and thus requires a sufficiently high performance near steady state plasma

Table 2
The CORC® cable selected for FNSF TF coil design.

CORC®	Diameter	# of Tapes	I_c of single tape
HF	8 mm	50	305 A 4.2 K, 20T
MF	7.2 mm	38	406 A 4.2 K, 14T
LF	6.375 mm	24	635 A 4.2 K, 8T

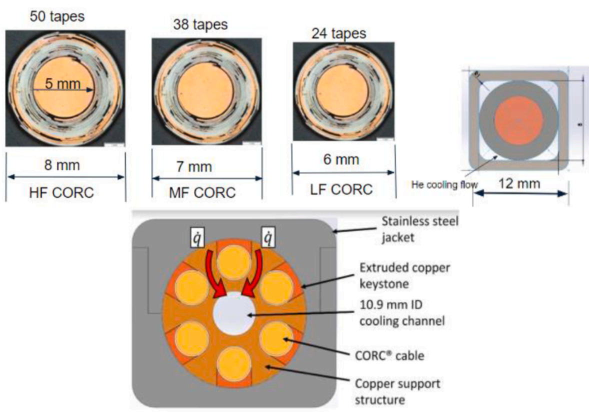

Fig. 3. A single CORC® CICC and a 6-around-1 CORC® CICC.

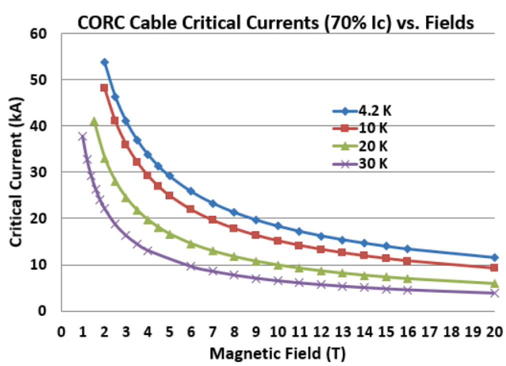

Fig. 4. I_c performance of a 50 tape 50 μm substrate CORC®.

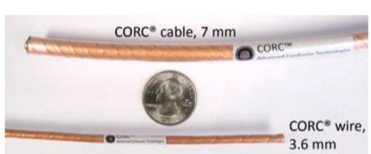

Fig. 5. A section of the sample CORC® cable from ACT [8].

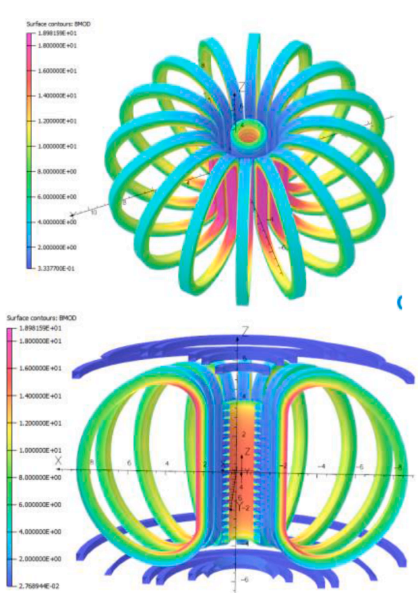

Fig. 6. Magnetic field distribution on the FNSF coil system.

with long durations and high duty cycles. Fig. 6 presents the magnetic field distribution on the FNSF coil system. A free standing CS coil is initially considered but a bucked and wedged TF-OH design is selected for a better structural support of TF inboard leg and OH coils due to radial build space constraints [2]. The upper and lower PF coils are widely separated as the result of horizontal maintenance from a large equatorial port on the outer leg side of the vacuum vessel. The CS coil is centered at 0.85 m radius in the radial build, it is composed of five coils above and five coils below the midplane with a total height of ~8 m. The maximum coil current occurs at back bias and at the end of plasma current ramp up [23]. For horizontal maintenance, a series of fiducial plasma equilibria are set up to examine currents and plasma states and the poloidal field (PF) coils are far from plasma requiring high currents so HTS is also considered for PFs. The maximum current density at the end of the plasma ramp for the CS and PF coils is about 80 A/mm^2.

3. Toroidal field magnets

The wedged TF coil design initially considered for in-board coil support structure relies on wedge pressure to take the large centering force from the TF magnet system. Fig. 7 shows the single HTS TF coil and the inboard leg cross section in the FNSF radial build where a large nose is required for reacting to coil vertical force.

3.1. TF winding pack layout

For the TF coil winding pack, forced flow helium cooling in a cable-in-conduit (CICC) configuration was investigated. More advanced gas cooling and conduction cooling schemes are considered but more challenging issues such as cooling efficiency and uniformity are to be resolved when applying to large scale fusion coils. The 6-around-1 CORC® CICC is presented in Fig. 8 with conductor grading in the winding pack layout. It is a 36 mm diameter square conduit with the 30/28/26 mm 6-around-1 CORC® cable core and 5 mm corner radius. The central cooling channel diameter is 10.9 mm and the CORC® cable dimension is adjusted from high to low field regions to enhance structural integrity of the TF winding pack by increasing jacket thickness. A thicker cable is needed with more tapes in the high field region while a lower number of tapes is needed to give space for a thicker jacket in the low field region to react to the transverse Lorentz force accumulation from high to low field regions. The TF field drops quickly radially in the inboard leg as shown in Fig. 8. The conductor grading is adopted for coil structural optimization. A similar single CORC® cable CICC design is presented to achieve a higher CICC current density (~80 A/mm^2).

The first option for the TF winding pack design is a single CORC® CICC of the 8 mm diameter cable with 50 YBCO tapes. The cable is supported in the square CICC jacket as shown in Fig. 3. For a design of 10.67 kA (4.2 K, 20 T) operating current (70% critical current), the total number of turns is >1500 for High Field of 14–20 T, Mid Field of 8–14 T and Low Field of <8 T. The second option is using the 64 kA 6-around-1 CICC for winding pack where a total number of turns is lower, to reduce inductance for coil protection during quench. A total stored energy is over 50% higher than the ITER TF coil as shown in Table 4. The total

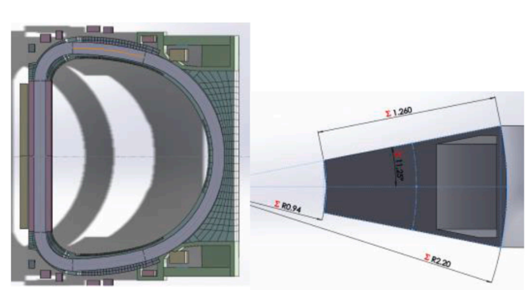

Fig. 7. Cross section of the wedged TF coil with a large nose.

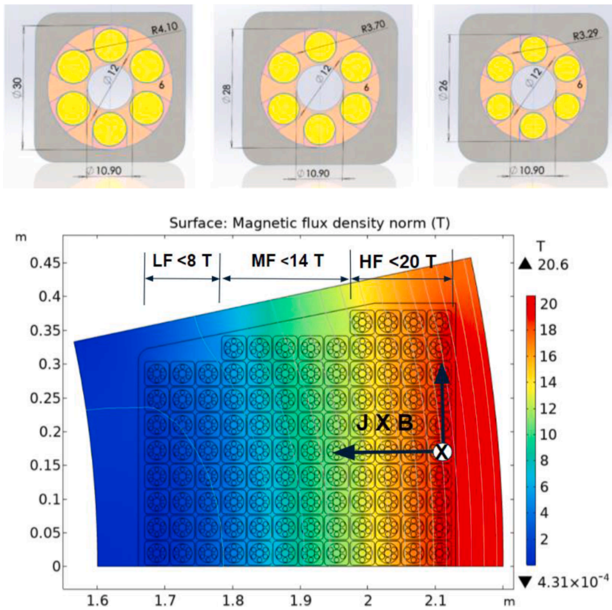

Fig. 8. In-board leg conductor grading and B field distribution.

Table 4
TF coil parameters.

	ITER	FNSF 6-around-1	FNSF single CORC®
WP Volume (m^3)	12.28	10.32	11.15
CICC current density (A/mm^2)	45	50	75
Stored energy / coil (GJ)	2.3	3.6	3.5
Centering force (MN)	403	1134	1066
Vertical force (MN)	208	410	414
Inductance (H)	1.01	1.8	61.2
Centering force / m (MN)	53.76	141.7	133.2

centering and vertical forces are 2–2.5 times higher than that in the ITER TF coil inboard leg. The inductance per TF coil drops to 1.85 H, comparable to ITER TF coil if a 6-around-1 CICC using CORC® is selected as shown in Fig. 8.

3.2. Winding pack stress analysis

Stress management in a TF coil winding is a critical issue for HTS fusion magnet design. There are generally two approaches to support high current cables in CICC winding packs. A sufficient volume of coil support structure is necessary to ensure an acceptable winding pack stress distribution. This can be realized by either a thicker jacket so each CICC turn in the inboard leg can be independently supported in high field region so to minimize accumulation of transverse Lorentz force from layer to layer toward central axis, or a build-in conductor support structure such as the radial plate used in the ITER TF winding design. A thicker jacket can enhance transverse stiffness of the jacketed conductor and thus mitigate transverse load accumulation effect by reducing transverse deflection of cables in the winding pack from layer to layer. This will effectively reduce the gap from cable transverse deflection in the high field side. The thicker jacket option may not be as effective as a radial plate between layers but radial plates can be an additional source of heating during transient events such as fast discharges and current ramping and quench of the magnets. The stress analysis for the TF in-board leg also demonstrated such a built-in radial plate like support structure can improve winding stress management in a high field coil in-board leg. Fig. 10 presents the Von Mises stress distribution in the TF leg cross section from a wedged TF design option. The results were obtained from COMSOL finite element model where multiphysics solvers were

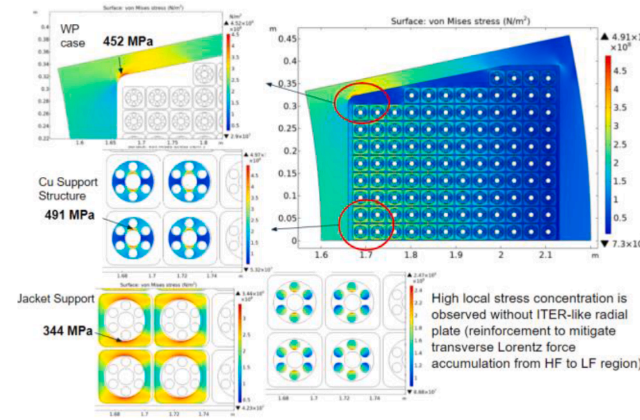

Fig. 10. Stresses in winding pack without radial plates.

used to couple magnetic field analysis to solid mechanics. The 2D generalized plane strain solver was used for the winding pack stress analysis. The model includes the TF in-board leg winding pack details. Steel material properties were used for the jacket and coil case; copper properties for the 6-around-1 cable support structure, 75% copper Young's modulus was used for cables. The winding pack stresses from system code analysis [25] are 300 MPa from vertical tensile force and 394 MPa from wedge pressure as a result of centering forces. For typical SS316 jacket and structure materials used in fusion magnets, the design limit is 660 MPa for primary membrane stress and 1 GPa for membrane and bending. A high local stress concentration is observed without the ITER-like radial plate as reinforcement to mitigate the transverse Lorentz force accumulation from HF to LF regions. Fig. 11 presents the local stress distribution in the CICC jacket and the CORC cable support structure.

3.3. Structures and cooling scheme

The forced flow of liquid helium cooling in the CICC with parallel flow in the TF winding pack is considered. Nuclear heating of 2–5 kW per coil for the FNSF design is the driving heat load for TF coil [26]. Influence of flow rate to the design of a parallel flow scheme is developed by a helium pipe flow model [27]. For adequate coil cooling, supercritical helium at inlet (6 bar and 4.2 K) is forced-flown through the 6-around-1 CICC cable. Table 5 presents the TF coil cooling parameters as compared to ITER [28]. Supercritical helium flow is the coolant for ITER and liquid helium forced flow for FNSF. The parallel flow path was designed for heat removal without subcooling of liquid helium. The inductive and thermal couplings between strands during quench and current sharing and non-uniform current distribution due to a variation

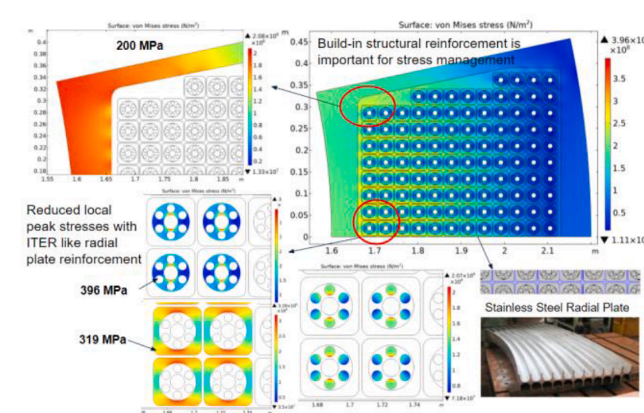

Fig. 11. Winding pack stresses with radial plate reinforcements.

Table 5
TF coil cooling parameters.

	ITER	FNSF 6-around-1
Operating pressure (MPa)	0.6	0.6
Inlet temperature (K)	4.5	4.2
Total mass flow rate (kg/s)	2.0	1.0
Total pressure drop (MPa)	0.1	0.4
Outlet temperature (K)	5.0	5.5
# of turns	134	218
# of parallel flow path	14	22
CICC mass flow rate (g/s)	8	45
Nuclear heating / coil (kW)	1	2–5

of contact pressure were investigated. The support structure is isothermal with cooling channel wall and helium flow. Issues identified include uniformity of thermal and electrical resistance among tapes used in a high current cable configuration. A low temperature is needed for stability, given a critical temperature of 15 K at 20 T and 16.7 kA in cables expected during operation.

3.4. Radiation and activation

The FNSF core components will expose significantly higher neutron fluence than ITER so neutron radiation and activation issues shall be addressed. The neutron irradiation to LTS and HTS materials and the radiation limits were studied [2–3,11–13,22–24]. High performance Nb_3Sn wire has a peak critical current at 3×10^{22} n/m^2 fast neutron radiation and REBCO coated conductors have a lower radiation resistance of 2×10^{22} n/m^2 for 20 K or less operation before experiencing performance degradation. Independent of artificial pinning in the tape, the pinning mechanism is largely dominated by induced defects after irradiation [11]. The peak TF winding pack fast neutron fluence calculated for FNSF design is $\sim 1 \times 10^{22}$ n/m^2. The copper stabilizer lower radiation limits can become a design driver. Conductivity changes in copper stabilizer are based on both transmutation rates due to thermal neutron flux and lattice defects. Conductivity changes from radiation damage due to fast neutrons do correlate with a decrease in irradiation temperature and an increase in its tensile strength. Previous studies showed, however, only 5% resistivity changes in Copper for 1.5×10^{23} n/m^2 fluence irradiation [29]. For YBCO and Bi-2212 used in fusion magnets, silver activation, which is the neutron interaction resulting in the release of secondary radiation, depends on effective cross section and neutron energy of interaction. The main activations from the 4% silver in YBCO tape but could be up to 50% in Bi-2212 conductor include two reactions, $^{107}Ag(n,\gamma)^{108}Ag$, $^{109}Ag(n,\gamma)^{110}Ag$ and the activation products are ^{108}Cd, ^{110}Cd. Fig. 12 presents the effective cross sections for activations at all neutron energies. Analysis performed for neutron energies of 0.0253 eV and 14 MeV and a neutron fluence of 10^{22} per m^2 showed that only a very small fraction of silver is activated and the impact to HTS performance is almost negligible.

4. Conclusions

Fusion Nuclear Science Facility is an intermediate step to accommodate the extreme fusion nuclear environment and address complex integration of components in the high fluence nuclear environment and plasma physics requirements. HTS such as REBCO is available for high field, high current density FNSF TF magnet design. The single CORC® and 6-around-1 CICCs are presented here for TF coil winding pack design. The HTS magnets provide high field access for FNSF plasma operation. Further investigations will focus on identifying structural issues for the integrated coil systems and the HTS coil quench management. Stress results on CICC jackets and CORC® cable support structure also imply that further studies on improving the winding pack current density are possible and will be beneficial.

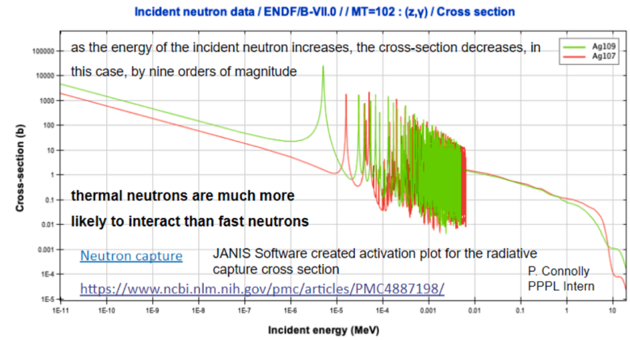

Fig. 12. Silver activation cross sections in HTS materials for fusion magnets.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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