FUSION TECHNOLOGY

Annual Report of the Association EURATOM/CEA 2002

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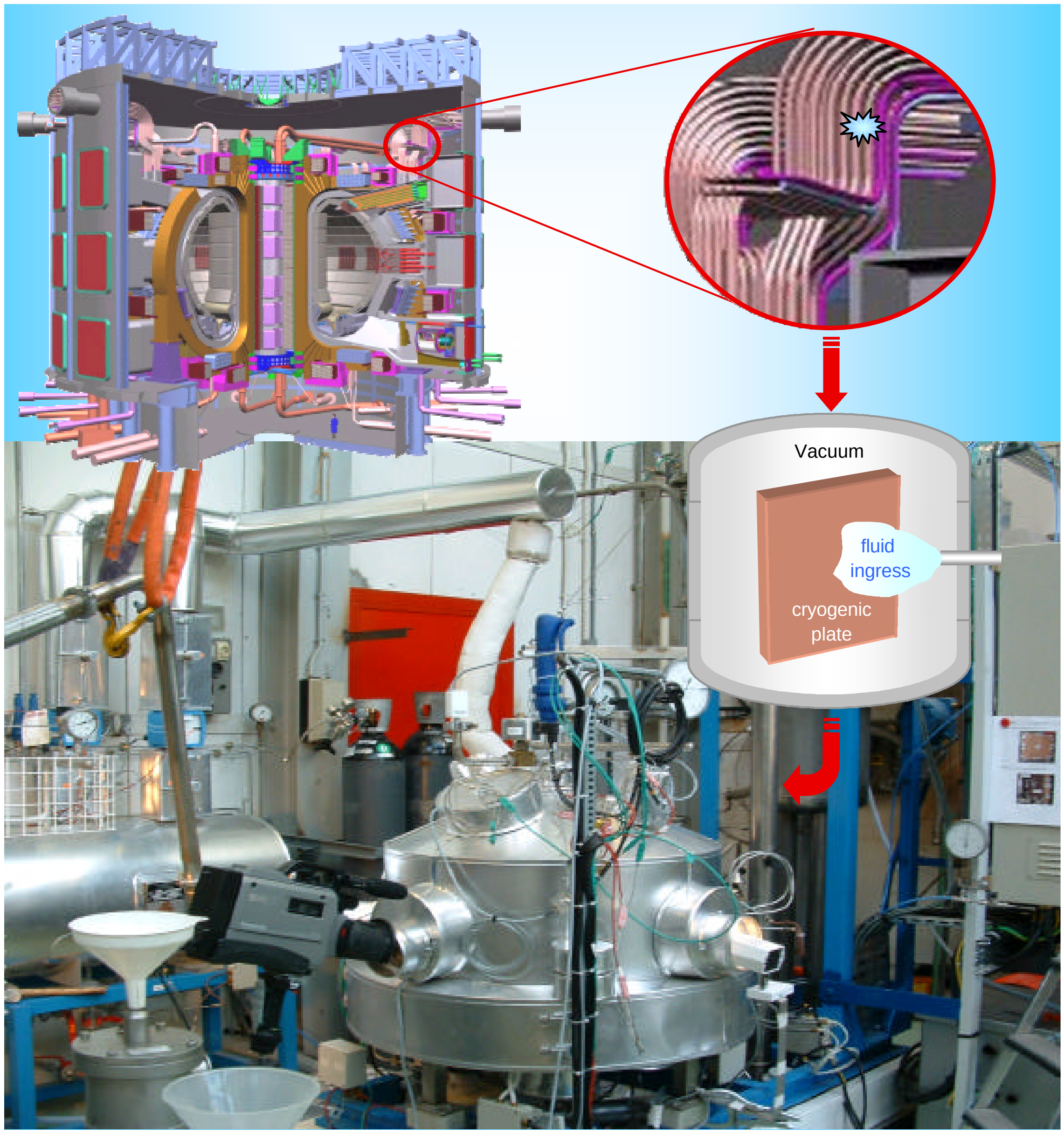

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CEFDA00-541

Task Title: MAGNET DESIGN ON PF AND CORRECTION COILS: CONCEPTUAL DESIGN AND ANALYSIS

INTRODUCTION

CEA Cadarache is requested to assist the EFDA Close Support Unit of Garching in the detailed design of the Poloidal Field Coils of the ITER machine, the design of the Central Solenoid cooling inlets and the development of some conductor analysis tools. In 2002 the task initiated in 2001 was completed by two studies on CS cooling inlets and conductor tails.

2002 ACTIVITIES

CS COOLING INLETS

Introduction

The Central Solenoid (CS) designed in the framework of ITER relies on pancake winding so as to allow flexibility in plasma shaping. The CS coil is divided into six modules made of seven hexapancakes. A single unit length of Cable-In-Conduit Conductor (CICC) is adequate for winding one hexapancake. To optimise the cooling, the six pancakes are cooled in parallel with supercritical helium.

The hydraulic scheme consists of 3 cooling inlets implemented on the inner bore, where the magnetic induction (13.5 T) is maximum, and 2 helium outlets on the outer radius of the coil at the cross-over between adjacent pancakes (figure 1).

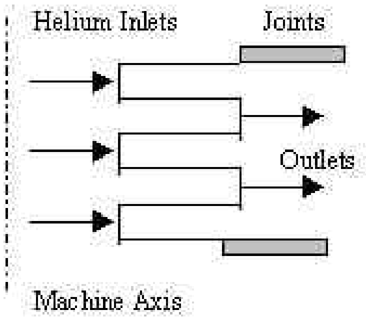

Figure 1 : CS hexapancakes hydraulic layout

Prototype design

In the framework of activity in 2001, a prototype design of the Central Solenoid cooling inlets has been proposed taking into account the feasibility and assuring the mechanical optimisation, with a local reinforcement to avoid excessive stress intensification.

However, the Central Solenoid CICC has a stainless steel jacket externally square. For these hydraulic tests, we use a mock-up implemented on the available ITER Toroidal Field Model Coil (TFMC) CICC (figure 2), with a externally circular stainless steel jacket (thickness = 1.6 mm and cable twist pitch length = 450 mm).

Figure 2 : TFMC CICC Cross section

The opening of the window is performed on the half perimeter of the circular jacket and on a length of 180 mm (figure 3).

A cover with the same distribution grooves as the CS cover is realised and an elliptical helium hole (7 x 51 mm) is performed in this cover.

Figure 3 : TFMC Full Size Cooling Inlet mock-up components

This cover is welded on the jacket ; it constitutes the cooling inlet and represents a middle point for the symmetrical hydraulic circuit and test facility (figure 4).

The inlet must provide a good distribution of the helium in the 6 CICC subcables (petals) to assure uniform cooling of all the superconducting strands.

Figure 4 : TFMC Full Size Cooling inlet mock up assembled

Hydraulic tests

The hydraulic tests performed in the OTHELLO facility with pressurized nitrogen at room temperature aim at characterizing the cooling inlet that means the associated pressure drop and the distribution of mass flows in each petal of the CICC [1].

The mass flow rate is measured in each petal of the CICC, at three different specific lengths of the conductor (at L = 2.16, 0.58 and 0.27 m). We observe that at a length of 2.16 m far from the cooling inlet, the mass flow rate in one petal vary between 7 % and 8 % of the total conductor mass flow rate which confirms previous calculations of mass flow distribution in CICC.

The ratio of mass flow in petal compared with the total mass flow then increases when the length from the cooling inlet is reduced reaching a maximum value of 10 % when the conductor length is 0.27 m (figure 5). Petal mass flow is influenced by the different flow regimes observed in the different conductor cross section locations. The dispersion of the petal mass flow is acceptable even very near to the cooling inlet (at L = 0.27 m). In this critical region, no petal has a mass flow lower than the maximum value observed far from the cooling inlet ; the diffusion of cooling fluid through petal region is therefore more efficient.

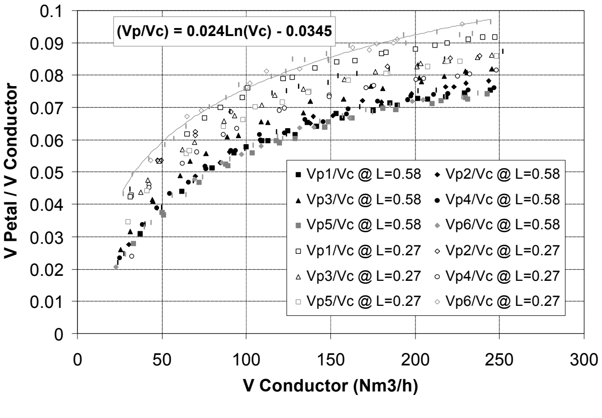

Figure 5 : Petal mass flow as a function of conductor mass flow at L= 0.58 and 0.27 m

Conclusion

In the framework of ITER, a prototype of the Central Solenoid cooling inlet has been manufactured and characterised with tests at pressurised nitrogen at room temperature. The mass flow rate measurements in all the 6 CICC subcables at different conductor cross sections confirms this cooling inlet provides an uniform distribution of the helium and assures correct cooling of all the superconducting strands. The petal mass flow being greater near the cooling inlet, the diffusion of cooling fluid through petal region will be more efficient at this critical point. The hydraulic tests leads to an estimation of 22 kPa pressure drop in normal operating conditions which represents an acceptable value considering the external cryogenic system. The total pressure drop in a CS pancake is therefore estimated 92 kPa.

PF CONDUCTOR TAIL MECHANICAL ANALYSIS

Introduction

The Poloidal Field (PF) coils of the International Thermonuclear Experimental Reactor (ITER) are designed with NbTi cable-in-conduit conductors wound in double pancakes which are connected in series by joints. These joints have to operate in the poloidal magnetic field which generate large varying forces. The conceptual design of the joints is based on the overlap concept with twin-boxes. A first mechanical proposal for these joints was to include a tie-bar to carry the tensile load and a clamping support to resist the radial and vertical loads. The mechanical analysis performed with several finite-element models of the joint area has shown that this design allows to keep both the tensile stress in the conductor and the shear stress in the insulation within acceptable limits.

However, this design is not applicable to the final coil connections to the leads where only one conductor is leaving the coil and then no linkage is possible between two facing outgoing conductors. The other solution is to transfer the tensile load by shear through the bond between a profiled tail welded on to the outgoing conductor and the adjacent conductors (inner and side turns). The tail has to be profiled in order to avoid a shear stress concentration higher to the acceptable limit of a bond between epoxy resin and steel. This report presents the mechanical study and optimisation of such a solution.

Design

A first tail design optimised by simple 1D analytic design was proposed by ITER Team. A hollowed profiled steel tail is welded to extend the outgoing conductor jacket and a corresponding steel filler is welded in regard of the tail on the outer turn of the adjacent pancake. The mechanical behaviour under the tensile load (hoop force) is insured by the bond of insulation between the lower face of the tail and the filler and also by the bond between the inner face of the tail and the inner turn of the pancake. Then there is only two active parts of the tail which are the lower and the inner faces.

This allows to use this design as well for the coil joints as for the two connections of the coil to the leads where only one adjacent pancake exists.

Model and analysis

The mechanical F.E. analysis is performed using the CEA code CAST3M.

The calculations are performed with the ITER PF1-PF6 conductor using a steel jacket. In all the calculations, the contribution of the cable is neglected. The calculations are based on a model of the tail region at the current lead conductor outgoing. For simplification, and because only the tensile load has to be transmitted, the tail area is considered to be straight.

The complete analysis is presented in [2].

After careful optimisation of the tail design, one main result is presented in figure 6 and concerns the shear stress level Sx in the insulation for the four bonding lines L1, L2, L3, L4 along which the tail is glued against the adjacent turns. It can be seen that the design value of 20 MPa is never exceeded.

Conclusion

A solution to carry the hoop force on any kind of joint was to use a classical bonded tail. So, a hollowed tail design with only two bonded faces according to the coil current lead joints layout is proposed and modelled. The F.E. calculation allowed to adjust the dimensional parameters of the tail in order to reduce the stresses at acceptable level. For an average tensile stress on the conductor of 200 MPa, the Von Mises peak stress reach 301 MPa in the steel and the shear peak stresses in the bonded insulation is 13 MPa. An important point for this design is that no steel wedge is needed facing the tail, then the manufacture is highly simplified during the winding operation.

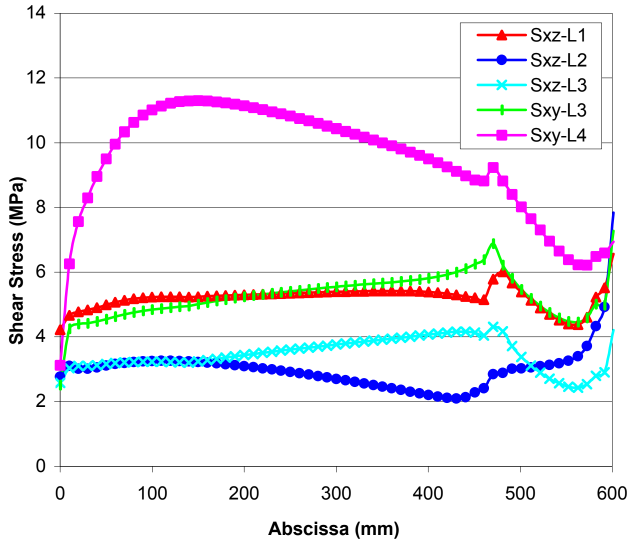

Figure 6 : Optimised design n° 2
Shear stresses along the bonding lines

CONCLUSION

The work to be performed in the framework of the Contract 00/541 is now completed and the final report has been issued [3]. The electromagnetic and thermohydraulic analyses of the conductor have shown that the conductors designed for the ITER PF Coils meet the design criteria during the reference scenario. It has been shown that the twin-box design is applicable to the PF coil joints. A design has been proposed for the ITER CS cooling inlets, mechanically analysed and hydraulically tested. The electric model of the cable-in-conduit conductors has been improved to take into account the annular temperature of helium, which enables to derive the local superconducting properties of the conductor from the temperature profile along the conductor length.

REPORTS AND PUBLICATIONS

[1] S. Nicollet et al. " Design and experimental characterisation of the hydraulic behaviour of cooling inlets for the ITER central solenoid" Presented at the 19 th International Cryogenic Conference (Grenoble, France, July 2002). To be published.

[2] D. Bessette and P. Decool, " Mechanical attachment of the conductor ends in the ITER Poloidal Field Coils", Presented at the 22nd Symposium on Fusion Technology (Helsinki, Finland, September 2002). To be published.

[3] J.L. Duchateau et al., EFDA Contract 00/541 – Final report, Note AIM/NTT-2002.014, 27/06/02

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M40

Task Title: DESIGN WORK ON MAGNET R&D

INTRODUCTION

This task, started in 1995, included design, participation to the follow-up of the manufacture of the ITER TF Model Coil (TFMC) and preparation of the tests of this coil.

The conceptual design of the TFMC was issued in 1995 [1] and the manufacture was carried out by the AGAN consortium (Ansaldo, Alstom, ACCEL, Babcock Noell Nuclear), selected after a call for tender, under a contract with EFDA [2].

The joints connecting the pancakes use the twin-box concept, developed by CEA [3],[4]. The coil was delivered in Karlsruhe, installed inside the TOSKA facility at FZK and tested in single coil mode in 2001[5].

2002 ACTIVITIES

ASSEMBLY AND COOLDOWN OF THE TF MODEL COIL WITH THE LCT COIL

After completion of the Phase I tests, the TFMC was disconnected and removed from the TOSKA vessel on 14 February 2002 to be assembled with the LCT Coil. After assembly, the TFMC and the LCT Coil were installed inside the TOSKA vacuum vessel on 03 April 2002 and connected hydraulically and electrically to the facility as well as to the instrumentation system by 26 July 2002. After closing the vacuum vessel and final checks at room temperature of helium tightness and dielectric strength, the cooldown was started on 20 August 2002 and completed on 03 September 2003.

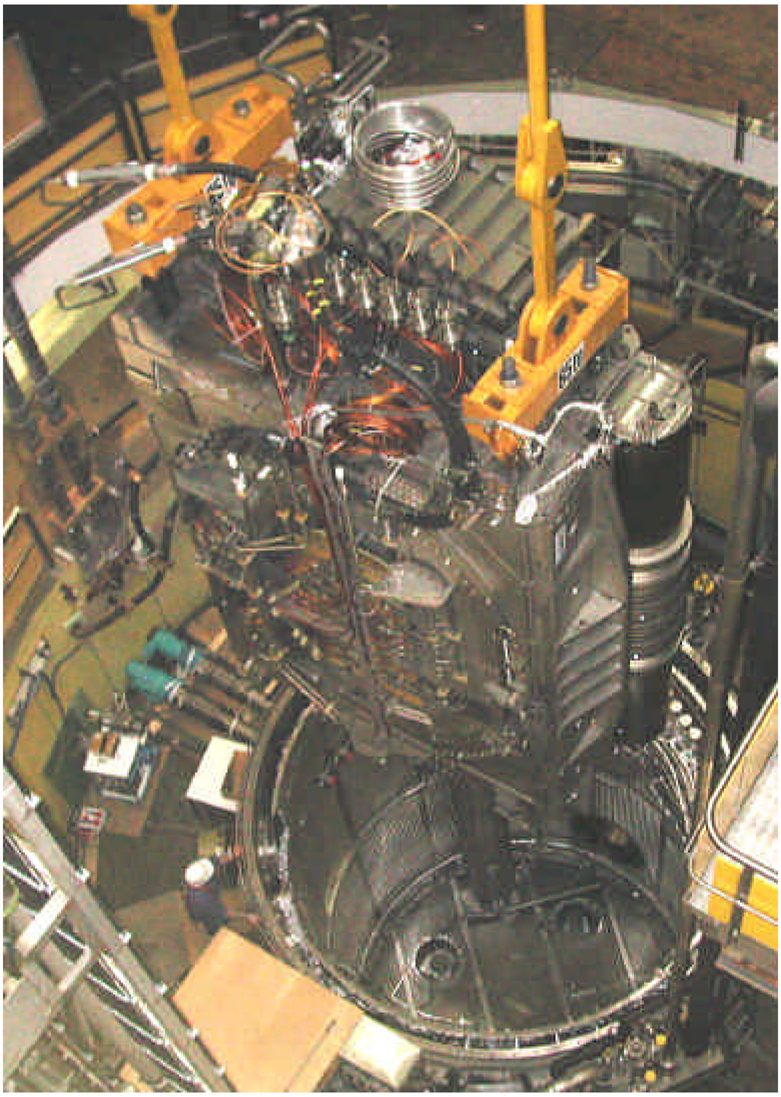

Figure 1 : Insertion of the TFMC with the LCT Coil inside the TOSKA facility at FZK (courtesy of FZK)

TESTS OF THE TF MODEL COIL IN DUAL COIL MODE

After final checks at low temperature, the coils were tested separately in single mode and the TFMC tested alone up to 80 kA, as in phase I. Both coils were then tested in dual coil test by ramping them simultaneously in current. The rated currents (70 kA in TFMC and 16 kA in LCT) were reached for the first time on 30 October 2002. After Tcs measurements and cycling tests during which the TFMC was ramped up from 0 to 70 kA, while the current was kept constant at 16 kA in the LCT Coil, a final test was performed on 21 November with 80 kA in the TFMC and 16 kA in the LCT Coil.

Results of the test analyses are reported in the report on the TOSKA Task activities.

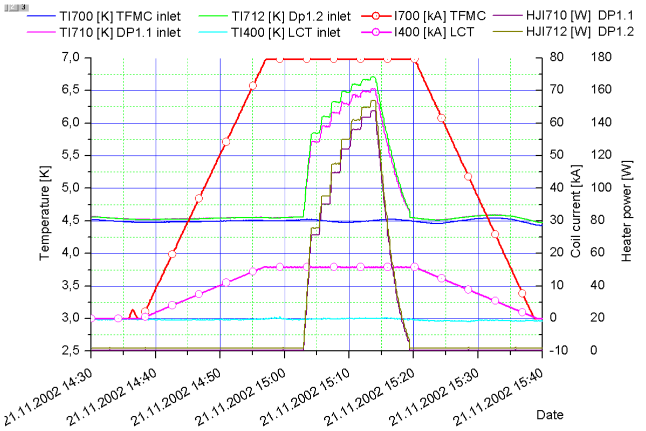

Figure 2 : Final test of the TFMC at 80 kA with the LCT Coil at 16 kA on 21 November 2002 (courtesy of FZK)

WARMING UP OF THE TFMC

After completion of the dual coil tests, tightness and electrical checks were carried out by FZK at low temperature and the warming up to room temperature was started on 05 December 2002 and completed on 19 December 2002. After final investigations at room temperature the vacuum vessel could be opened on 30 January 2003.

CONCLUSION

The assembly of the TFMC with the LCT Coil and their installation inside the TOSKA facility allowed to perform the Phase II tests of the TFMC, where the TFMC was extensively tested in dual coil mode. CEA followed-up the assembly and participated to the tests. The final test was performed with 80 kA in the TFMC and 16 kA in the LCT Coil. After completion of the tests, both coils were warmed up to room temperature.

REFERENCES

[1] Conceptual design of the ITER TF Model Coil, P. Libeyre et al., IEEE Trans. on Mag., vol. 32, pp 2260-2263, 1996.

[2] The ITER Toroidal Field Model Coil (TFMC), E. Salpietro et al., Fusion Eng. And Design, Vol. 46, November 1999, pp. 151-158.

[3] Joints for large superconducting conductors, P. Decool et al., Fusion Eng. And Design, Vol. 58-59 , November 2001, pp. 123-127.

[4] Large superconductors and joints for fusion magnets : from conceptual design to test at full scale, D. Ciazynski et al., Nuclear Fusion, vol.41 n°2 (2001) p.223-226.

[5] Test results of the ITER toroidal field model coil experiment in the TOSKA facility of the Forschungszentrum Karlsruhe, A. Ulbricht and the ITER L2 Project Team, 22nd SOFT, Helsinki, 09-13 September 2002.

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Task Title: CONDUCTOR R&D
Development of NbTi conductors for ITER PF coils

INTRODUCTION

The development of NbTi conductors for ITER PF coils has to address several questions:

- Characterisation of NbTi strands in a range of temperature between 4 K and 7 K and in magnetic field up to 10 T.

- Control of the conductance between strands by strand surface coating or by a CuNi internal barrier at the proper level in the range 10^7 $(\Omega m)^{-1}$. The solution has to take into account the temperature of the curing step.

- Connections : design and fabrication procedures have to be confirmed and tested on subsize conductors in the connection test facility JOSEFA. The strand-copper sole contact cannot be treated as for Nb_3Sn due to the absence of high temperature heat treatment.

  The removal of the coating if any has to be specifically addressed.

  The impregnation heat treatment occurring after the connection fabrication of real coils and the final joint soldering has to be considered.

- Conductor behaviour : The operation margin can be assessed on subsize conductors in JOSEFA and the final conductor and joint check can be performed on a full size joint sample in the SULTAN facility at Villigen.

The development can be led on cable and jacket dimensions compatible with the existing fabrication tools (cabling + jacketing) and the existing test facilities.

This activity is performed in collaboration with ENEA.

2002 ACTIVITIES

CRITICAL PROPERTIES CHARACTERISATION OF THE TWO CANDIDATE STRANDS FOR THE ITER PF COILS [1]

Two ITER PF-candidate NbTi strands were characterized through DC tests: J_C measurements at various magnetic fields and temperatures (see figure 1). All tests were achieved at Grenoble High Magnetic Field Laboratory (CNRS, Grenoble) in a Variable Temperature Cryostat developed in CEA Cadarache.

Finally, the main strand performances parameters determined through a classical fitting for NbTi (C_0, T_{C0}, B_{C20}) were evaluated. All these results will then be used as a reference for future samples analysis.

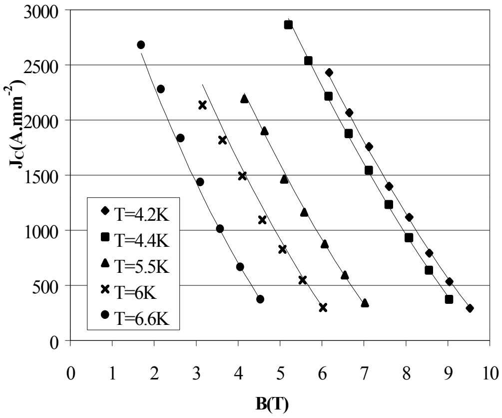

Figure 1 : Critical current density results for the Alstom strand at various B and T

MANUFACTURE AND TEST OF TWO SUBSIZE CONNECTIONS AND CONDUCTORS RELEVANT TO PF OPERATION [2]

From the two selected strands: Europa Metalli (EM) Ni coated strand and Alstom strand with CuNi barrier, two subsize joint samples (respectively PF1 and PF2) were manufactured in 2001 using the 108 strands conductor lengths.

In 2002 the Joint part of each sample was characterized in the JOSEFA facility:

- DC tests : Temperature of current sharing (T_{CS}), quench temperature (T_Q) and Joint Resistance (R_J) at various fields and temperatures.

- AC losses tests : trapezoidal magnetic field variations with various ramp rates.. Time constant were deduced consistently with R_J. Note that the transverse field orientation is here along the worst direction with respect to joint losses and induced currents, thus simulating the radial field component (i.e. the most demanding component) in the ITER PF coils.

- Stability tests : trapezoidal B pulse at different current, temperature and mass flow rate.

In DC tests, we found joint Current Sharing Temperatures (T_{CS}) consistent with strand ones as well as joint resistances that meet ITER criteria.

However the PF1 sample showed a higher resistance than expected (8 nΩ instead of less than 3 nΩ) which is thought to be due to a incomplete Ni coating removal (with mechanical brushing).

In AC losses tests, the main contribution comes from intercable currents crossing the joint plane (since B is in joint plane, normal to conductors), the results confirmed the R_J tests and led to time constants of 11 and 6 s for PF1 and PF2, respectively.

In stability tests we showed that in the range of mass flow rate studied, we can reach the so-called "quasi-static" regime, in which the global flow is choked. Classic thermo-hydraulic models can thus be used to analyse those tests.

To summarize, all joint tests were achieved and analysed in 2002. Conductor tests are planned for year 2003.

UPGRADING OF THE EXISTING JOSEFA TEST FACILITY FOR THE TEST OF TWO SUBSIZE CONNECTIONS

The facility was upgraded in 1999. The tests of the first subsize NbTi joint sample were performed within year 2001 at various temperature under magnetic field up to 3.4 T and with a transport current through the samples up to 10 kA. The first tests started in the joint position (upper position) in 2001 were successfully completed by tests in the lower position for conductor test during year 2002. The new control command of the test facility was used for the test of both samples. An upgrading of the sample heating system was launched to avoid temperature oscillations and should be fully operational in the beginning of year 2003.

MANUFACTURE OF THE PF FULL SIZE JOINT SAMPLE [4]

The CEA proposal for the conception of the first NbTi full size joint sample was to use the so called twin box concept successfully developed for Nb_3Sn joint samples. A conceptual design of the PF-FSJS was developed by CEA, and technical specifications were issued for the call for tender to industry [3]. The manufacture of the sample was entrusted by ENEA to Alstom (Belfort, France), and CEA performed the monitoring of the manufacture.

After cabling of the cable lengths by Europa Metalli (Italy), two 11 m long conductors lengths were finally jacketed by Ansaldo (Genoa, Italy) and delivered to Alstom in March 2002. One length was manufactured using the Europa Metalli nickel coated strand and the other one using the Alstom internal copper-nickel barrier strand.

Due to variations of the cable twist pitch along each full length, the two 4 m conductor lengths needed for the PF-FSJS manufacture were chosen in the best part. The remaining cable twist pitches were 390 mm for the Alstom strands leg and 410 mm for the Europa Metalli strands one. In parallel, a copper-steel (CuC1-316LN) plate bonded by the explosive technique was manufactured by Nobelclad (Rivesaltes, France).

A termination box mock-up was first manufactured for validation of the procedures and check of the real final joint cross section. The PF-FSJS manufacture was then performed taking into account the experience gained during the previous samples manufacture (SS, TFMC and TF Full Size Joint Samples) in the QA procedures. The result was that no problem of deformation of the terminations nor misalignment appeared. The manufacture of the sample and its instrumentation were performed in a short time of about 3 months.

The sample was delivered to CRPP for tests on July 10, 2002.

TESTS OF THE PF FULL SIZE JOINT SAMPLE

The PF-FSJS was extensively tested in the SULTAN facility in summer 2002. CEA and ENEA both participate in the tests of the sample at CRPP (Villigen, Switzerland). CEA wrote the testing programme in agreement with ENEA [5]. Critical current (and current sharing temperature), pulsed field loss, and stability were measured on both conductor legs within a wide range of operating conditions including the ones foreseen in the ITER PF coils. Moreover, conductors were tested before and after 500 cycles between 0 and 50 kA under 6 T.

The lower joint was also tested by lifting up the sample in the facility : voltage drop (i.e. joint resistance), quench temperature, pulsed field loss, and stability were measured under operating conditions around the ones foreseen in the ITER PF coils. It should be mentioned here that, because of the facility limits ($I < 100$ kA), the PF-FSJS tests could be relevant only to the ITER PF1 & PF6 coils, which are among the main demanding coils.

Also, at variance with the JOSEFA facility, the orientation of the pulsed field in the SULTAN facility allowed testing only against the best field orientation with regard of the joint, and not against the worst field orientation (corresponding to the radial component in the coils) which is by far the most demanding situation in the ITER PF coils.

The analysis of the test results by CEA (in parallel to the one performed by ENEA) has shown a lot of interesting results. Regarding the conductor legs, the main results are the rather high critical currents (equal or even larger than the scaled values from single strands) obtained at low current (below 20 kA on EM leg and below 30 kA on Alstom leg), while at higher currents, instabilities prevent from reaching the electric field criterion (i.e. 10 µV/m). The reducing effect of these instabilities was more pronounced on the EM leg than on the Alstom leg (see figure 2) and, in a general way, the EM leg was found less stable than the Alstom leg.

A global study on the reconstruction of current distribution through segmented Rogowski coils signal analysis showed that this difference is likely to be due to uneven current repartition between petals in the left leg (one petal underloaded, one overloaded).

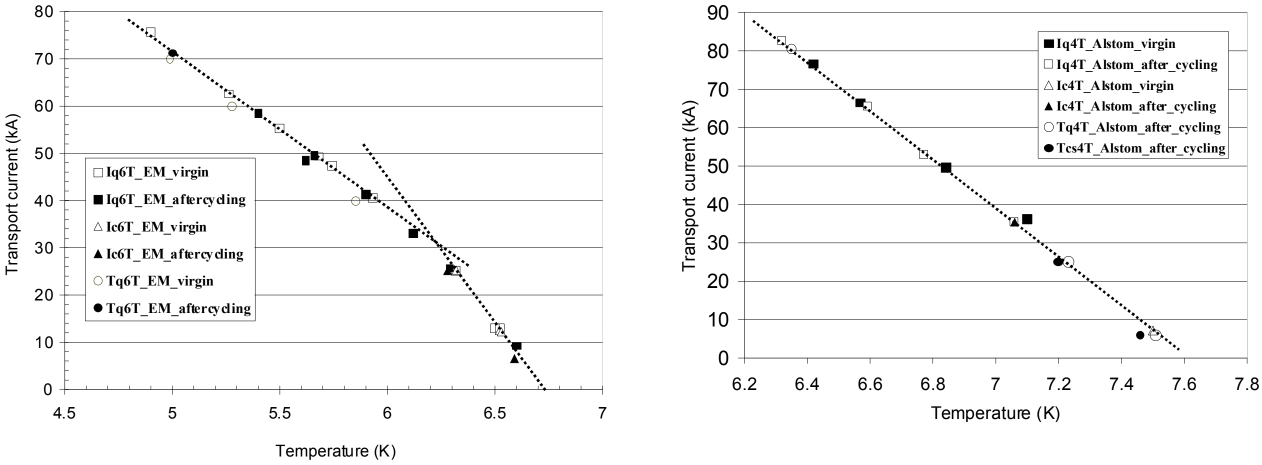

Figure 2 : PF-FSJS DC typical tests results : Left leg at B=6T (left) and Right Leg at B=4T (right)
Note the slope change for the EM leg in the unstable region

As mentioned above for the subsize samples, this is thought to be due to uneven strand-to-sole contact resistance distribution in the left part of the joint (possibly, Ni coating removal was not uniform).The pulsed field losses have shown the expected behaviour, i.e. decreasing with cycling on both legs (twice faster for the right leg, but with comparable final values).

As usually, the final losses measured at high frequency (> 1 Hz) were found close (or even slightly lower) to the ones measured on the single strands ($n\tau \sim 4 - 5$ ms) but with “apparent” hysteresis losses 3 times higher than expected. However it was found that the low frequency losses, relevant to ITER PF coil operation, remained rather high ($n\tau \sim 70$-80 ms) but with the expected hysteresis losses (see figure 3). Though this behaviour is not completely understood, those parameters are to be taken into account for future ITER design previsions.

The joint DC resistance was found at the expected level (~ 1.5 nΩ), with also the expected magneto-resistance effect (0.11 nΩ/T, see figure 4), although a high scattering along the half left (EM) joint was observed. This latter is thought to be created by the previously mentioned widely spread contact resistance spectrum in the left part of the joint.

Using a conservative model developed at CEA it was shown that the joint pulsed field losses also met the expectations under trapezoidal pulses relevant to the ITER PF coils with obviously the restriction mentioned above.

Worth to be noted is the odd behaviour of the conductor (both legs) under heating (e.g. pulsed field losses) at low mass flow rate (i.e. 4 g/s). This phenomenon was characterized by an increase of the upstream temperature above a given steady state power (~ 3 W), which was attributed to a chocking of the helium flow in the annular area of the conductor. A thermohydraulic analysis performed by CEA has shown that this phenomenon was due to the gravity effect (the sample is tested vertically) creating a thermo-siphon which tends to decrease and then to reverse the annular helium flow as the power increases.

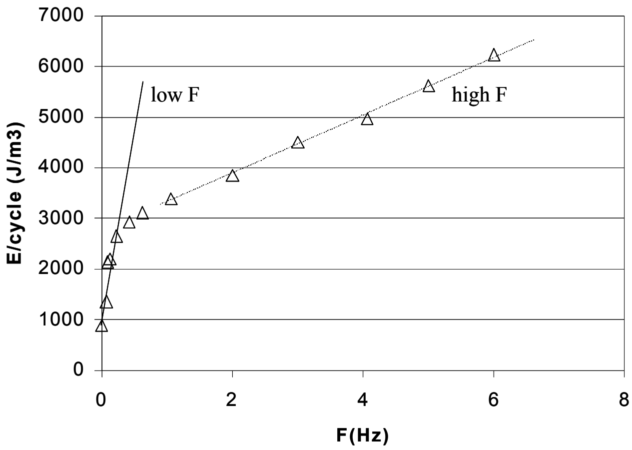

Figure 3 : PF-FSJS AC Losses tests on Right Leg after 500 cycles. Note differences between low and high frequencies, regarding both slopes (nt) and extrapolations at F = 0 (hysteresis losses)

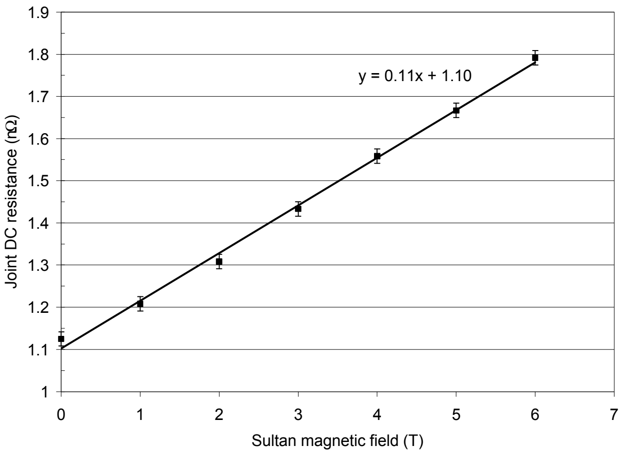

Figure 4 : PF-FSJS Joint resistance vs. applied magnetic field at 60 kA

This phenomenon, which anyway appeared in operating conditions far away from those foreseen in the ITER PF coils, looks therefore not relevant to these coils which are mainly horizontal (maximum slope of 5 %).

CONCLUSIONS

During year 2002, two full size PF conductors, each of them using one of the previously selected strands were manufactured by industry. A PF Full Size Joint Sample was manufactured by industry under CEA monitoring, and delivered to CRPP for tests. The tests were performed by CEA and ENEA in the SULTAN facility. The results obtained on the right (Alstom) leg were quite satisfactory, while the left (EM) leg showed a significant degradation of the maximum transport current due to instabilities. The reason of this limitation is not yet understood although a non-uniform current distribution among petals has been found, but its effect is not fully assessed. The joint resistance and pulsed field losses have been found to be at the expected levels, but the PF-FSJS should be tested in the PTF facility (MIT, USA) to measure the joint behaviour against the most severe pulse field orientation.

The upgraded JOSEFA test facility was successfully used for subsize joint as well as for subsize conductor testing.

REFERENCES

[1] L.Zani, H. Cloez, Z. Bej, J.P. Serries and E. Mossang, *Critical properties of ITER PF strands under varying field and temperature,* Note AIM/NTT-2001.031.

[2] P. Decool, H. Cloez, *Task M50: Manufacture of two subsize NbTi joint samples*, Note AIM/NTT/2001.028.

[3] P. Decool, *Task M50: Deliverable 5, Report on design of full size conductor samples*, Note AIM/NTT/2001.027.

[4] P. Decool, H. Cloez, *Task M50: Deliverable 6, Manufacture of the PF Full Size Joint Sample*, Note AIM/NTT/2002.016.

[5] D. Ciazynski*, Task M50: Testing programme of the PF-FSJ in SULTAN*, Note AIM/NTT/2002.013.

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Task Title: CSMC AND TFMC INSTALLATION AND TEST

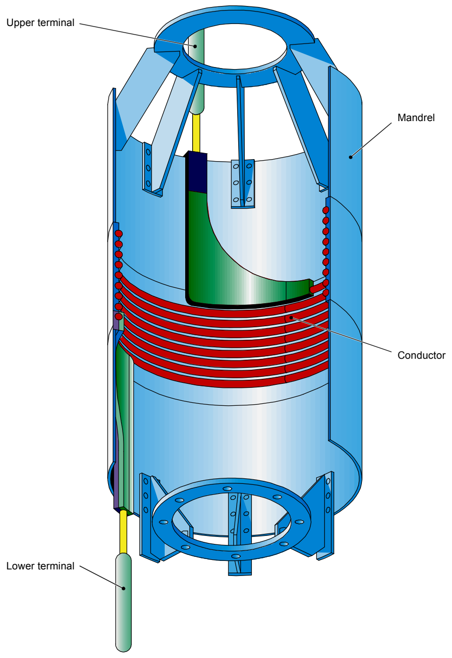
INTRODUCTION

This task covers the participation of CEA to the tests of the Central Solenoid Model Coil (CSMC) at Naka (Japan) and to the tests of the Toroidal Field Model Coil (TFMC) at Karlsruhe (Germany). This task is a continuation of the work carried out by CEA in the framework of the Task M40. The tests of the CSMC covers the tests of the main coil itself and the tests of the CS and TF Inserts. The tests of the TFMC have been performed in two steps : in phase I tests, the TFMC has been tested as a single coil and in phase II tests, the TFMC has been tested in a dual coil test, together with the EU-LCT Coil .

2002 ACTIVITIES

TESTS OF THE TF INSERT

In the framework of the ITER EDA phase, the Toroidal Field Coil Insert (TFCI), a single layer coil (figure 1), using 45 m of conductor, has been manufactured by the Russian Home Team. It was tested in October 2001 in the CSMC at JAERI under various field and temperature conditions. The TFCI conductor (figure 2) is a dual channel cable-in-conduit conductor, made of a 1152 Nb_3Sn strand cable, jacketed with a thin circular titanium jacket, whereas the ITER TF conductor is designed with a stainless steel jacket.



Overview of the RF Toroidal Field Conductor Insert (TFCI)

Figure 1 : The ITER TF Insert

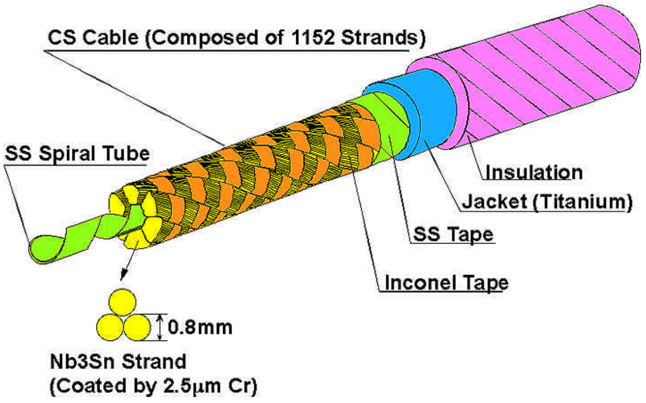


Figure 2 : The TF Insert Conductor

The TF Insert was tested up to 46 kA under 12 T. The work performed by CEA in 2002 [1] has been dedicated to the analysis of the Tcs measurements with an electrical model using the ENSIC code, developed by CEA.

When assuming a uniform current distribution among subcables, the model allowed to determine an equivalent intrinsic strain inside the strands (figure 3).

The average found value (-0.66 %) is much higher than expected with a titanium jacket, which has a thermal contraction coefficient close to that of the strands.

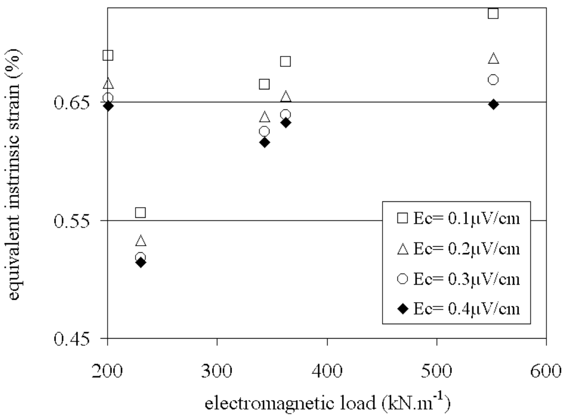


Figure 3 : Equivalent intrinsic strain as a function of the electromagnetic load

An analysis with unbalanced current distribution among subcables led to a better fitting of the evolution of the electrical field versus temperature in the current sharing area (figure 4) when assuming that 2 subcables are not connected in the upper joint to the CSMC busbars.

It was further discovered that a repair performed on a joint termination could explain this phenomenon.

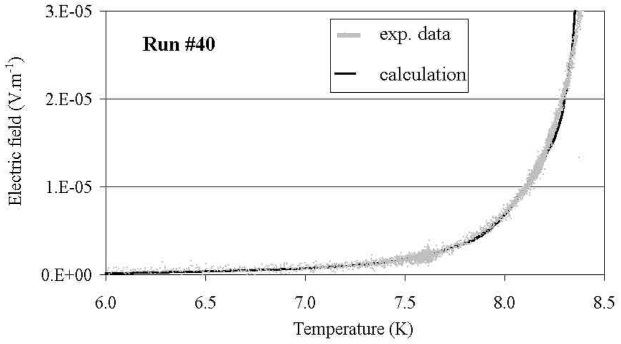


Figure 4 : Electrical field evolution with temperature in the central turn of the TF Insert

PHASE II TESTS OF THE TF MODEL COIL

The phase II tests of the TF Model Coil have been performed in Karlsruhe from September to November 2002. CEA was actively involved in these tests, as well by on-site as by remote participation and by performing analyses of the electromagnetic, thermohydraulic and mechanical properties of the coil. This work is reported in the framework of the TOSKA Task activities.

CONCLUSION

The ITER TF Insert coil, manufactured by Russia, was tested in Naka in Autumn 2001 and CEA participated to the analysis of the results. The electromagnetic analysis carried out at CEA showed that the intrinsic strain in the conductor is found much higher than expected with a titanium jacket and that a strong unbalance occurred in the current distribution, further explained as a consequence of a repair performed on one joint. The ITER TFMC was tested in a dual coil configuration in Karlsruhe in autumn 2002 and CEA participated to the tests and to the analysis of the results.

REFERENCES

[1] Electromagnetic Analysis of the ITER Toroidal Field Coil Insert Properties, L. Zani et al., ASC 2002, Houston, 04-09 August 2002.

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TW1-TMC-CODES

Task Title: DESIGN AND INTERPRETATION CODES
Determination of thermohydraulic properties
of cable-in-conduit conductors with a central channel

INTRODUCTION

The thermohydraulics of cable-in-conduit conductors has to be well described, to allow proper design of the cryogenic system of ITER. The pressure drop of central channels representative of ITER-FEAT has to be characterised. The recooling time of forced flow coils and the quench behaviour are driven by the heat transfer coefficient between the annular area and the central channel. This coefficient can hardly be theoretically evaluated; only experiments, possible at room temperature, can bring information about this coefficient.

These experimental activities are led at CEA Cadarache on dedicated facilities in collaboration with Politecnico di Torino.

2002 ACTIVITIES

PROCUREMENT OF SPIRALS AND MEASUREMENT OF HYDRAULIC RESISTANCE IN THE OTHELLO FACILITY

To characterise the thermal and hydraulic behaviour of the Cable In Conduit Conductor (CICC), the friction factor for the channel have to be determined, specially for the central channel in order to define a useful correlation and predict pressure drop as well as heat exchange coefficient influence.

From previous work and tests performed on the Toroidal Field Model Coil central spirals provided by Showa and Cortaillod [1], we first defined a certain number of spirals representative of the ITER-FEAT Coil design. These spirals have different parameters values : the hydraulic external diameter being 7.6, 8 and 10 mm, the turn length is 6.5 mm and the twist pitch length varies between 8.65, 11.8 and 12.5 mm (figure 1).

After a call for tender, the company Mécaressorts was selected to provide us with the central spiral samples. Unfortunately, delivery of these spirals was delayed by some months due to manufacturing problems.

When received, these samples were installed in the Othello Test Facility at Cadarache, to be tested with pressurised nitrogen at room temperature. The measurements, performed in early 2003 should lead to the characterisation of the central spirals hydraulic behaviour.

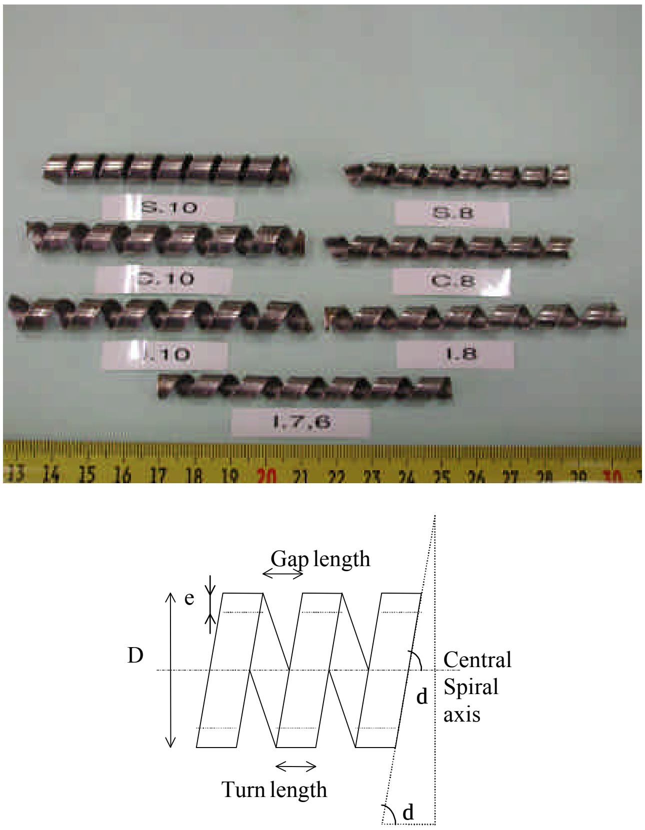

Figure 1 : Central spiral samples

EXPERIMENTAL EVALUATION OF THE HEAT TRANSFER COEFFICIENT BETWEEN ANNULAR CHANNEL AND CENTRAL HOLE OF ITER CONDUCTORS

An important parameter of the ITER magnets cryogenic cooling system is the recooling time.

The cable-in-conduit conductor being cooled by a high speed flow (1 m/s) in the central channel in parallel with a slow speed flow (0.1 m/s) in the annular area, the recooling time is depending on:

- The heat transfer coefficient between the two parallel channels.
- The fluid velocity in each channel.

A numerical and analytical model was developed to predict the temperature evolution along a CICC after a temperature step at the inlet. This model relies on the heat transfer coefficient between the two parallel channels.

To evaluate this heat transfer coefficient , an experimental facility operating in relevant Reynolds number at 100°C in pressurised water was defined [2].

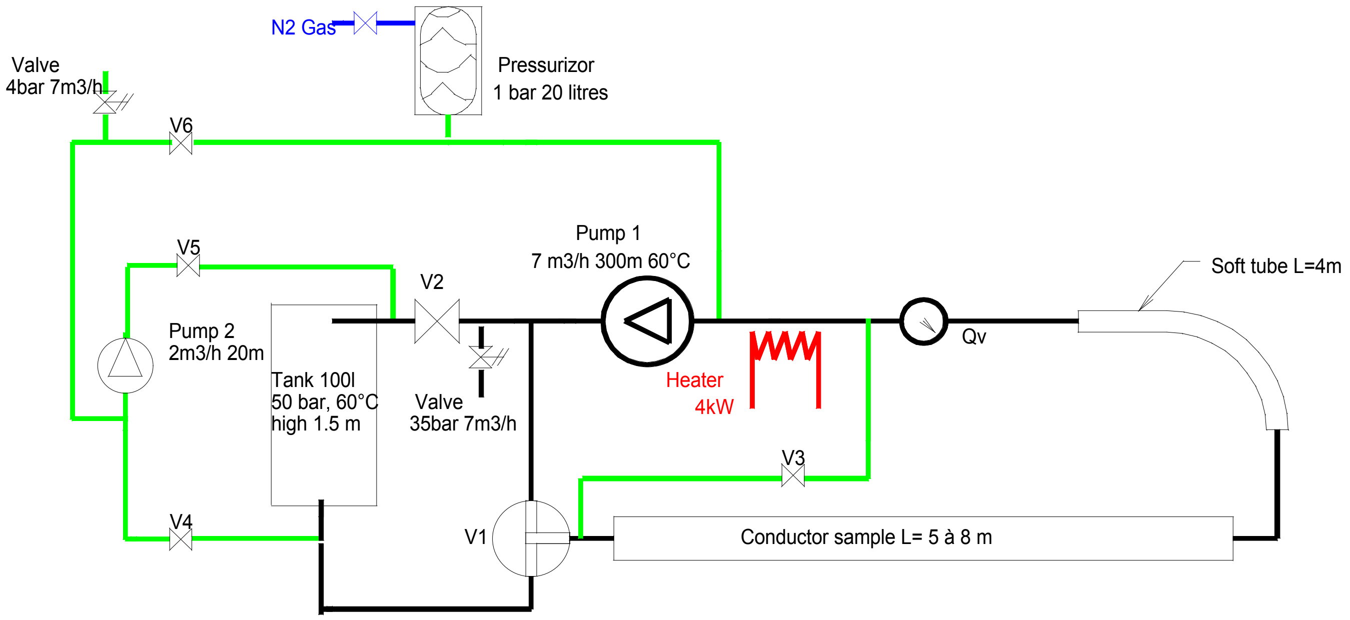

Figure 2 : HECOL facility scheme

A call for tender showed that the cost of such an installation was not compatible with the allowable budget. A review of the project with reasonable limitations especially on the range of the reachable Reynolds number led to a revised design of the test facility (figure 2), with operation reduced to 60°C [3].

The loop allows a circulation of water at 50°C at nominal Reynolds number. Thanks to the 100 liters tank pre-heated up to 60°C and the 3 way valve V1, a temperature step is imposed through the conductor sample. The temperature variations recorded at the inlet and at the outlet allow to derive the conductor heat transfer coefficient.

After the corresponding call for tender, the components and the assembly were ordered. The test loop named HECOL (Heat Exchange Coefficient Operational Loop) will be assembled and operational by March 2003. The relevant tests in water should be completed within 2003.

CONCLUSION

During the year 2002, seven central spiral samples relevant to ITER conductors with different geometrical parameters were defined and manufactured by industry.

These spiral samples were installed in the Othello test facility to be tested in 2003. The hydraulic resistance measurements should lead to an hydraulic model of the central channel of the ITER conductors.

For the evaluation of the heat transfer coefficient between annular area and central channel of ITER cable-in-conduit conductors, a new experimental facility operating in pressurized water at 60°C was defined and ordered. This facility should be operational for tests by March 2003.

REFERENCES

[1] S. Nicollet, J.L. Duchateau, H. Fillunger, A. Martinez, S. Parodi, Dual Channel Cable In Conduit Thermohydraulics : Influence of some Design Parameters, IEEE Transactions on Applied Superconductivity, vol.10, N°.1, 2000, pp.1102-1105.

REPORTS AND PUBLICATIONS

[2] A. Martinez, Conception d'une boucle à eau pressurisée pour l'évaluation du coefficient d'échange thermique entre les canaux d'un conducteur cable en conduit à double canal, Note AIM/NTT-2002.001.

[3] P. Decool, Tâche CODES : Evaluation du coefficient d'échange thermique entre les canaux d'un conducteur cable en conduit à double canal. Proposition experimentale, Note AIM/NTT/2002.024.

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TW1-TMC-SCABLE

Task Title: CABLE AND CONDUCTOR CHARACTERISATION
Determination of critical properties and losses of superconducting strands and cables for fusion application

INTRODUCTION

In order to achieve efficient work on ITER magnet design, with help of previously developed models, reliable database on components performances are necessary to use. In this purpose, a task has been defined to evaluate all performances related to superconducting conductors at strand scale and subcable scale. The strands involved are those made of NbTi to be used for the Poloidal Field Conductor Insert (PFCI) and those made of Nb_3Sn to be used for the Central Solenoid Hexapancake-Module (CSHM).

The CEA participation in this project includes the following actions:

- Heat treatment of the Nb_3Sn strand. This treatment will take place at CEA Cadarache and is the one that will be used for the CSHM manufacture.
- AC losses measurements on 36-strands subsize samples. Those samples will be procured to CEA and characterised in the JOSEFA Facility (Cadarache).
- Critical Current tests at various magnetic fields and temperatures of Nb_3Sn and NbTi strands will be performed in the Variable Temperature Cryostat (VTC) facility (GHMFL, CNRS Grenoble).

The samples to be tested will be specified and delivred to CEA by EDFA.

2002 ACTIVITIES

PROCUREMENT OF THE NBTI STRAND FOR PFCI

The strand has been manufactured by Bochvar Institute (Russia) and delivered to CEA in September 2002. A micrographic analysis of the cross-section has been launched.

SAMPLE HOLDER PREPARATION AND STRAND INSTALLATION

The strand has been prepared and installed on the VAMAS-like sample holder to be used in GHMFL tests. This action was carried out according to a precise CEA procedure, adapted for NbTi strands.

PROCUREMENT OF THE NB_3SN STRAND FOR CSHM

The decision to manufacture in Europe the CSHM is delayed. No Nb_3Sn strand was delivered.

PROCUREMENT OF THE 36-STRANDS SUBSIZE SAMPLES

No samples were delivered to CEA.

CONCLUSION

During the year 2002, the NbTi strand procurement and the sample manufacture were achieved. Measurements are scheduled for the year 2003 in GHMFL (Grenoble). No Nb_3Sn strand and no 36-strands samples were delivered to CEA.

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TW1-TMS-PFCITE

Task Title: POLOIDAL FIELD CONDUCTOR INSERT (PFCI)

INTRODUCTION

Within the framework of the ITER project, the EFDA CSU has been asked to manufacture a model coil, called Poloidal Field Conductor Insert (PFCI), to be tested in the JAERI test facility in Naka, Japan. The development, manufacture and testing of the PFCI coil shall support the design of the ITER PF conductors and coils.

The main objective of the model coil tests is to get a complete knowledge and understanding of the behaviour of high current NbTi cable-in-conduit conductors and related joints under operating conditions as foreseen for the ITER Poloidal Field (PF) coils. A conductor representative of the ITER PF coils shall be wound in a single layer coil and equipped with a numerous instrumentation composed of inductive heaters, voltage taps, temperature and pressure sensors, strain gauges, etc. The coil shall be inserted inside the bore of the ITER CS Model Coil (CSMC) at the JAERI test facility in Naka (Japan) and tested in 2004.

The coil winding features a square conductor with a NbTi superconducting cable inserted in a thick wall, stainless steel jacket. Superconducting joints are required to connect the coil to the current leads. Another joint is located at an intermediate location in the winding to test an ITER-relevant joint under magnetic field operating conditions similar to the ones foreseen in the ITER tokamak. The upper and lower terminations shall connect the winding to the existing CSMC Insert busbar system of the Naka facility, as well as to the cryogen supplies. CEA has been deeply involved in the design and analysis of the ITER PF coils through EFDA contract 00/541, and task M50. The work of CEA within task TW1-TMS-PFCITE is as follows:

- Definition and the review of the test procedure.
- Participation to operational campaigns of the PFCI and reporting of the results.
- Analysis of the results, including thermo-hydraulic, electro-magnetic, and structural simulations of the real operating conditions of the coil.
- Analysis of impact of results on ITER PF coils design.

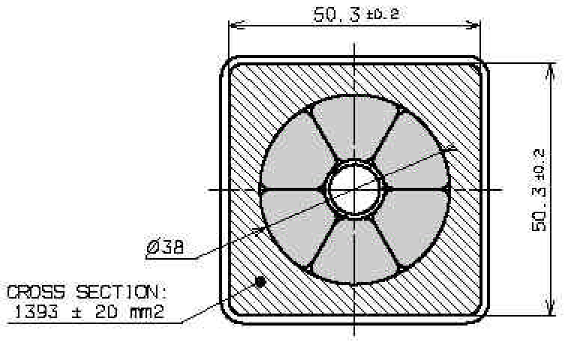

Figure 1 : PFCI NbTi conductor cross section

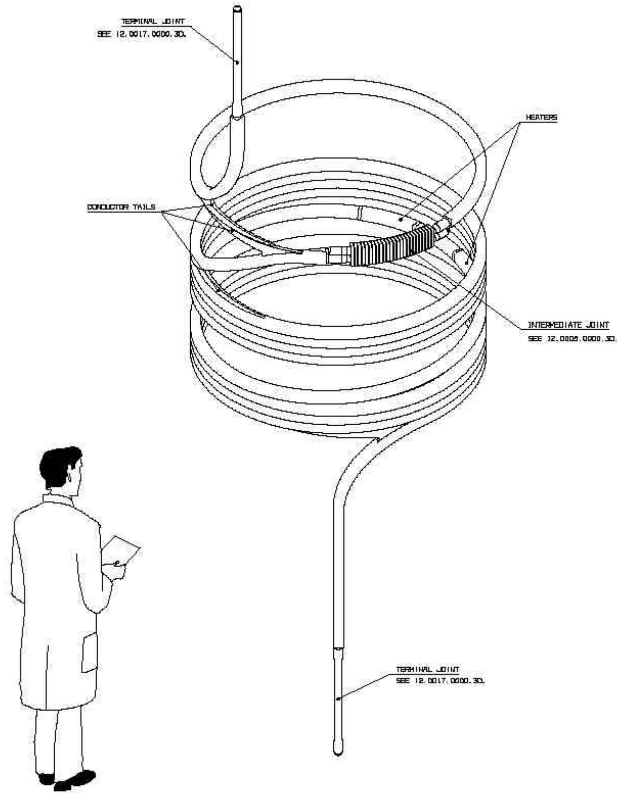

Figure 2 : PFCI winding

The NbTi cable for the PFCI was delivered in April 2002 from Russian Federation, the jacketing is expected to be completed at Ansaldo in April 2003.

Fabrication of the coil at Tesla (UK) is expected to be completed in February 2004.

Final acceptance and shipment to the CSMC test facility is foreseen in March 2004, and start of installation and testing programme at JAERI in April 2004.

2002 ACTIVITIES

For 2002, our activities were confined to collect detailed information on the conductor and coil designs, as well as on the preliminary proposals for the testing programme already issued by EFDA, ITER IT, and JAERI.

We will participate in the redaction of the final testing programme document which should be issued in 2003

Table 1 : PF Conductor Insert Coil Cable Parameters

Conductor external dimensions (after compaction)	50.3 x 50.3 mm^2
Jacket material	**Stainless Steel (Valinox)**
Cable outer diameter (before compaction)	38.7 mm
Cable outer diameter (after compaction)	~ 38.0 mm
Number of SC strands	**1440**
Number of Cu strands	0
Cabling pattern	(3x4x4x5)x6
Last cabling stage twist pitch length	490 mm
Central tube (idxod)	12 x 10 mm
Overall annular area	~ 1021 mm^2
Actual cable area in annulus	~ 974 mm^2
Wraps area + empty corners	~ 36 + 11 mm^2
Nominal local void fraction:	~ 34 %

Table 2 : Major Operating Parameters of PF Conductor Insert Coil

Electromagnetic Parameters	
- Testing conditions at 4.5 K (at JAERI test facility):	
- maximum operating current,	50 kA
- maximum operating field,	6.3 T
- maximum field change (disruption).	2 T/s
- Maximum stress in Valinox conductor jacket.	110 MPa
- Maximum operating voltage.	10 kV
- Maximum operating turn voltage.	1 kV
- Impulse test voltage (in-factory & at 4.5 K).	21 kV
Geometrical Parameters	
- Main winding envelope:	
- inner diameter,	1393.4 mm
- outer diameter,	1570 mm
- height.	1400 mm
- Length of conductor.	49.5 m
- Number of turns.	9
- Number of layers.	1
Thermohydraulic Parameters	
- Number of cooling paths:	
- main winding	1
- support structure	1
- Range of inlet temperatures for testing.	4.0 – 7.0 K
- Inlet pressure.	0.6 MPa
- Range of inlet pressure for testing.	0.3 MPa – 1 MPa
- Proof pressure.	3 MPa

CONCLUSIONS

The Poloidal Field Conductor Insert is now under fabrication in industry and should be tested in the CSMC facility (Naka, Japan) in April 2004. The testing programme document should be issued in 2003 with the agreement of EFDA, ITER IT, and JAERI.

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TW2-TMST-TOSKA

Task Title: TFMC TESTING WITH THE LCT COIL

INTRODUCTION

In the framework of the TW2-TMST-TOSKA Task, CEA is asked by EFDA to participate to the testing of the ITER Toroidal Field Model Coil (TFMC) in the TOSKA facility at FZK in Karlsruhe and to the testing of the ITER Poloidal Field Conductor Insert Full Size Joint sample (PFCI-FSJS) in the SULTAN facility at CRPP in Villigen. Whereas the TFMC was tested in 2001 in phase I as a single coil, it was tested in 2002 in phase II in a dual coil test with the EU-LCT Coil.

2002 ACTIVITIES

PHASE II TESTS OF THE TFMC

In 2002 CEA was involved in the interpretation of TFMC Phase I experiment, issuing several publications at different conferences.

The main subjects treated in these publications are the pressure drop analysis, the cold losses developed in the radial plates during transients and the estimation of the critical performances of the conductor [1], [2] [3], [4], [5], [6].

CEA participated then actively to TFMC Phase II experiment, which took place from September 2002 to November 2002. 5 people from CEA attended specific experiments of this TFMC phase II experiment. J.L Duchateau as the Deputy Testing group Leader for these tests spent 2 weeks at FZK.

In addition two people from CEA participated to the 16th TFMC Test and Analysis meeting which took place at FZK on the 28 May 2002.

The main aspects of the CEA participation in 2002 are recalled in the following sections.

Ac and eddy current losses analysis

25 kA discharge analysis

The radial plates TFMC (figure 1) in which the conductors are inserted, constitute a very representative and very specific component of ITER.

These plates play a leading part in ITER to sustain the very important centring force in the system due to the magnetic load. Due to these plates, the insulation system is also very specific and was tested in the TFMC.

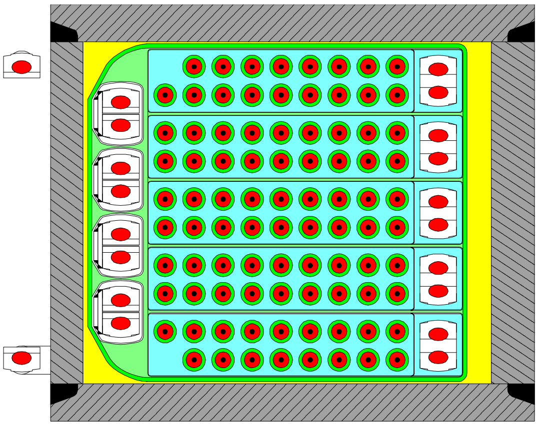

Figure 1 : Cross-section of the TFMC with plates and conductors

The heat transfer from plates to conductor during a fast discharge is now correctly modelled by gandalf coupled to a heat diffusion simulation.

It was experimentally confirmed during 25 kA discharge analysis, that this process is driven by the conductor insulation conductivity. The observed heat transfer to helium is in addition slower than expected due to insulation thermal conductivity lower by a factor of 6 compared to the fiberglass thermal conductivity.

This is not well understood and can be due to the multi layer arrangement of the insulation; this has important consequences for ITER, especially for the fast discharge, and deserves an experimental crosschecking on short insulation samples.

Ripple losses due to power supply

In the TFMC, due to the 600 Hz power supply voltage ripple, the plates constitute a secondary whose primary is the TFMC winding.

It was difficult to electromagnetically model this ripple because the 600 Hz voltage component of the power supply, is not known precisely. In addition, the model is based on the leakage inductance between plates and winding which is impossible to estimate with precision.

The thermohydraulic manifestation of the ripple was observed in particular during a long run at very low current (4 kA) to eliminate any contribution from the joints. It mainly consists in a global load of 40 W in the plates.

The phenomenon was modelled by Gandalf. The helium outlet temperature increase is in this case, driven by the helium recooling time. A temperature gradient exists between plates and helium which is inversely proportional to the insulation conductivity.

The side pancakes which are less magnetically coupled to the winding experience apparently a less important power which can be estimated by calorimetry.

Exploration of limits of the conductor

This part was the most important part of the TFMC Phase II experiment. During these tests, the inlet temperature of the P1.2 pancake is progressively increased as presented in figure 2. During this process the total voltage across the pancake, starting from zero increases progressively as a function of temperature in the so-called current sharing temperature regime. It is to be noted that the stability of the pancake is so high that the voltage comes back to zero when the pancake heating is switched off.

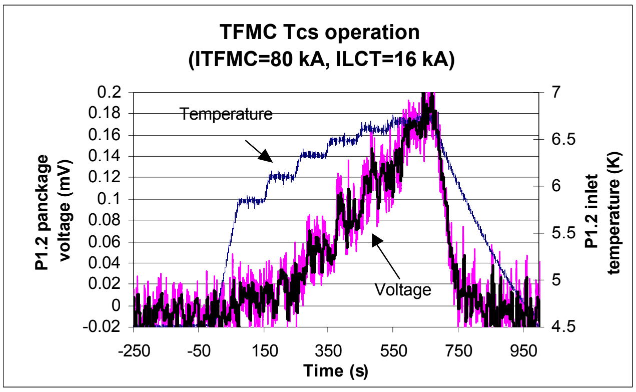

Figure 2 : Conductor limit exploration in TFMC P1.2

In the test presented in figure 2, the TFMC has been pushed to the highest possible magnetic load acceptable by the mechanical structures of the coil, such as to create the highest field on the coil.

In such a test, thanks to the field reinforcement due to the LCT coil energized at 16 kA, the maximum field on P1.2 conductor has been pushed to a value in the range of 9.7 T to be compared to 11.8 T for ITER.

This kind of test is an unique opportunity to learn about the critical properties of a conductor in conditions representative of ITER especially for the conductor current (80 kA in the TFMC and 69 kA in ITER).

As a matter of fact Nb_3Sn which is the TFMC superconducting material is very sensitive to strain, and the strain value is very difficult in practice to predict for this collection of twisted strands embedded in a stainless steel jacket.

Thus, from the characteristics presented in figure 2 it is possible to deduce a value of strain in a range [-0.75 %, -0.78 %], a slightly higher value than the one expected from short sample measurements (-0.70 %). This corresponds typically to a reduction of temperature margin by 0.3 K – 0.4 K.

CEA has studied the conductor behaviour for several runs corresponding to different magnetic loads. This strain seems also to be a linear decreasing function of the magnetic load.

Using different tools such as the ENSIC code developed within an ITER task and the very refined magnetic field map produced by the code TRAPS, the influence of several parameters have been studied carefully [7] to characterize :

- the exact temperature on the conductor in the current sharing regime,
- the exact field map across the conductor,
- the role of current distribution among the strands due to the proximity of the joint from the current sharing region (2 meters).

Mechanical analysis

This analysis was jointly carried out by FZK and CEA. A fairly good agreement is generally found between predictions given by the FEM model and results provided by the displacement sensors and by the rosettes. During single coil operation, the coil deformation is an in-plane deformation resulting under the radial load from the combination of an increase of its perimeter and a tendency to become circular. This results in a reduction of the vertical opening and an increase of the horizontal opening.

As a consequence of the bending occurring in the straight legs, the inner joints are submitted to contraction whereas the outer joints are submitted to elongation. During two-coil operation, the same deformation is occurring, at an increased level, due to the higher average field on the winding-pack, and an out-of-plane bending is superimposed, due to the attraction between the two coils and the localised support at the four wedges.

Consequently, the vertical opening and the elongation of the outer joints are higher on the LCT side than on the front side. In the single coil case, the measured horizontal displacements are under the predictions (-18 %), whereas in the two-coil case, they are slightly under the predictions on the front side (-5 %) and above the predictions on the LCT side (+14 %).

This results probably from an underestimation by the model of the coil rigidity versus in-plane bending and an overestimation versus out-of-plane bending, although the model was improved with respect to that used for the analysis of the Phase I tests. The coil case is now modelled with two elements in the thickness and the thickness of the case has been corrected in the joint area.

Concerning the coil stiffness the deviations between predictions and measurements for the stresses show similar tendencies, e.g. the values are under predicted on the LCT side by ~ 30 % and over predicted on the front side.

As far as the ICS is concerned, the displacement sensors indicate a higher rigidity versus in-plane loading. The ICS front leg shows more bending stiffness and the rear leg, which is hooked at the LCT case, less torsion stiffness as predicted by the model. The stress measurement at the highly loaded side wedge indicates a lower rigidity versus out-of-plane bending.

Nevertheless, the allowable limit was never exceeded, even during the 80 kA-16 kA test.

The already available results demonstrate that the global mechanical behaviour of the coil assembly was fairly well predicted by the model, within error margins unavoidable due to the complexity of the structure.

Joint resistances in Phase II experiment

No evolution of TFMC inner joint resistances was observed between Phase I and Phase II experiments at 80 kA in TFMC without current in LCT. The slight increases of the TFMC inner joint resistance with current in LCT follow the expected magneto-resistance effects. In addition no significant evolution was also observed on the joints with the bus bars which have been dismantled and reassembled between Phase I and Phase II.

TESTS OF THE PFCI-FSJS

Unfortunately, these tests could not be performed in 2002, since the contract for manufacture of this prototype sample could be placed only in late 2002. The manufacture and tests of this sample are now planned for 2003.

CONCLUSION

The test of the TF model of ITER was quite a success, due to a strong involvement of the EU Associations, demonstrating that the concept of this coil is sound for ITER.

It was however confirmed thanks to these tests, that the filament strain in TF conductor is slightly higher than expected in the ITER design in a range which can correspond to a decrease of the temperature margin by 0.3 K - 0.4 K . This decrease can be easily compensated, keeping the same size of conductor and a stainless steel jacket, by taking benefit of the progress in superconducting materials critical properties during these last ten years.

REFERENCES

[1] Duchateau J.L. et al. “Ac losses in the ITER TF model coil : a comparison between model and experimental results”. Presented at the 19 th International Cryogenic Conference (Grenoble, France, July 2002). To be published.

[2] Heller R., Ciazynski D., Duchateau J.L, Marchese V., Savoldi-Richard L., Zanino, R. “Evaluation of the Current sharing temperature of the ITER Toroidal field model coil”. Presented at the Applied Superconductivity Conference (Houston, USA, August 2002). To be published.

[3] Duchateau J.L. et al. “ITER toroidal field model coil test: analysis of heat transfer from plates to conductors”. Presented at the 22 nd Symposium on Fusion technology (Helsinki, Finland, September 2002). To be published.

[4] Nicollet S., Duchateau J.L., Heller R., Zahn G. , Savoldi L., Zanino R., Fillunger H., Maix R. “Hydraulic resistance of the ITER toroidal field model coil dual channel Cable in Conduit conductor pancakes”. Presented at the 19 th International Cryogenic Conference (Grenoble, France, July 2002). To be published.

[5] Nicollet S., Duchateau J.L., Hertout P. “Heat transfer from plates to conductors : from toroidal field model coil analysis to ITER model”. CHATS 2002 (Karlsruhe, Germany, September 2002). To be published in Cryogenics.

[6] Duchateau J.L., Ciazynski D., Zani L. ”Recent results on the electromagnetic behaviour of large conductors for fusion application”. Presented at the 3rd IAEA Technical Committee meeting (Greifwald, Germany, May 2002).

[7] Duchateau J.L “ Minutes of the 17 th TFMC Test and Analysis meeting” held at CEA Cadarache (04-05 February 2003).

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