Fusion Engineering and Design 89 (2014) 2606–2620

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Conceptual integrated approach for the magnet system of a tokamak reactor

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HIGHLIGHTS

- We give a conceptual approach of a fusion reactor magnet system based on analytical formula.
- We give design criteria for the CS and TF cable in conduit conductors and for the magnet system structural description.
- We apply this conceptual approach to ITER and we crosscheck with actual characteristics.
- We apply this conceptual approach to a possible version of DEMO.

ARTICLE INFO

Article history:
Received 29 January 2014
Received in revised form 14 June 2014
Accepted 18 June 2014
Available online 17 July 2014

Keywords:
DEMO
Superconducting magnets
Pulsed reactor

ABSTRACT

In the framework of the reflexion about DEMO, a conceptual integrated approach for the magnet system of a tokamak reactor is presented. This objective is reached using analytical formulas which are presented in this paper, coupled to a Fortran code ESCORT (Electromagnetic Superconducting System for the Computation of Research Tokamaks), to be integrated into SYCOMORE, a code for reactor modelling presently in development at CEA/IRFM in Cadarache, using the tools of the EFDA Integrated Tokamak Modelling task force. The analytical formulas deal with all aspects of the magnet system, starting from the derivation of the TF system general geometry, from the plasma main characteristics. The design criteria for the cable current density and the structural design of the toroidal field and central solenoid systems are presented, enabling to deliver the radial thicknesses of the magnets and enabling also to estimate the plasma duration of the plateau. As a matter of fact, a pulsed version DEMO is presently actively considered in the European programmes. Considerations regarding the cryogenics and the protection are given, affecting the general design. An application of the conceptual approach is presented, allowing a comparison between ESCORT output data and actual ITER parameters and giving the main characteristics of a possible version for DEMO.

1. Introduction to fusion and superconductivity [1]

Fusion by magnetic confinement requires large magnet systems to confine the plasma inside the vacuum chamber. The production of the magnetic field with superconducting magnets in the large vacuum chamber of ITER (835 m^3) is one of the main technological challenges, which must be tackled. In the early sixties, when applied superconductivity was merging with the first small Nb_3Sn magnets, it was quickly identified that this technology was compulsory for fusion [2].

Tore Supra (TS) [3] and the large helical device (LHD) [4] have now been in operation long enough to demonstrate that superconducting magnets at low temperature can reliably provide confinement for plasmas at 100 millions Celsius degrees. Lessons can be drawn from their accumulated experience. This can be of interest at a moment when several superconducting tokamaks are in construction especially the construction of ITER, a superconducting tokamak reactor with 700 t of superconductors.

Regarding tokamaks, four superconducting machines: TS, EAST [5], KSTAR [6] and SST-1 [7] are now in operation. Two others JT-60SA [8] and ITER [9] are now in an advanced stage of production. In addition, the superconducting stellarator W7-X [10] will soon enter the commissioning phase.

1.1. Which technology for DEMO?

By the way, the most emblematic fusion project is ITER by its size and the international teams, which are involved in its construction.

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http://dx.doi.org/10.1016/j.fusengdes.2014.06.012
0920-3796/

Nomenclature

a tokamak minor radius (m)
a_{cs} CS conductor side (m)
A tokamak aspect ratio
A_{Cu} copper section in a cable (m^2)
A_{He} total helium section in a cable (m^2)
A_{noncu} non-copper section in a cable (m^2)
B_t magnetic field at plasma centre (T)
B_{cs} magnetic field in CS coil (T)
B_{maxtf} maximum magnetic field on TF cable (T)
B_{vert} vertical field to maintain plasma equilibrium (T)
$\beta_{N,th}$ normalization of the beta toroidal parameter
β_p poloidal β
$C_{\text{-Ejima}}$ Ejima coefficient for plasma resistive consumption
δ_{95} triangularity at the plasma edge
e_{CS} radial thickness of the CS (m)
e_{TF} radial thickness of the TF (m)
Δ_{int} radial distance between plasma edge and TF cable (m)
ΔH_{circ} enthalpy difference on a cooling circuit (J kg^{-1})
ΔP_{circ} pressure drop along a cooling circuit (Pa)
ΔT_{margin} temperature margin for cable design (K)
$\Phi_{cablecs}$ CS cable diameter (m)
h half of the CS height (m)
I_{BS} boot strap plasma current (MA)
I_{cond} conductor current (A)
I_{CS} CS conductor current (A)
I_{ext} externally driven plasma current (MA)
I_{plasma} plasma current (MA)
I_{TF} TF conductor current (A)
J_{cable} cable current density (A/m^2)
J_{cablec} cable current density at inflexion point(Am^2)
$J_{cableCS}$ CS cable current density (A/m^2)
J_{CS} CS conductor current density (A/m^2)
J_{noncu} non-copper current density in a cable (A/m^2)
k characteristic parameter of a perfect D TF coil
κ_{95} elongation at plasma edge
l_i Plasma reduced internal inductance
L_{plasma} plasma inductance (H)
L_{TF} TF coil inductance (H)
$\dot{m}$ helium massflow in a cooling circuit (kg s^{-1})
N number of TF coils
n_e plasma volumetric average density (m^{-3})
P_{fus} reactor fusion power (MW)
P_{en} reactor electrical power output
Ψ_{CS} CS magnetic flux (Wb)
$\Psi_{consumption}$ total magnetic flux consumption during a plasma discharge (Wb)
$\Psi_{contribution}$ total magnetic flux contribution to a plasma discharge (Wb)
Ψ_{ind} inductive magnetic flux plasma consumption during plasma ramp up (Wb)
$\Psi_{plateau}$ magnetic flux available for plateau (Wb)
Ψ_{PI} magnetic flux consumption for plasma breakdown (Wb)
Ψ_{res} resistive magnetic flux plasma consumption during plasma ramp up (Wb)
Ψ_{vert} vertical magnetic flux contribution (Wb)
Q fusion energy amplification
R tokamak major radius (m)
Ripple magnetic field ripple at plasma external edge
r_0 characteristic parameter of a perfect D TF coil (m)
r_1 average radius of the TF internal leg (m)
r_2 average radius of the TF external leg (m)
r_{etf} TF internal leg external winding pack radius (m)
r_{ecs} CS external radius (m)
r_{ics} CS internal radius (m)
r_{itf} TF internal leg internal winding pack radius (m)
r'_{itf} TF internal leg internal radius (m)
ρ helium volumetric mass (kg m^{-3})
$\sigma_{centeringavtf}$ TF winding pack average centering stress (MPa)
$\sigma_{hoopavtf}$ TF winding pack average hoop stress (MPa)
$\sigma_{hoopavcs}$ CS average hoop stress (MPa)
$\sigma_{hoopcsss}$ CS stainless steel hoop stress (MPa)
$\sigma_{maxvault}$ criterion for TF vault maximum stress (MPa)
σ_{maxwp} criterion for TF winding pack maximum stress (MPa)
σ_{Tmax} Criterion for stainless steel maximum stress (MPa)
$\sigma_{vaulttf}$ TF vault stress (MPa)
T_e plasma average electron temperature (10 keV)
t_{inscs} CS conductor insulation thickness (mm)
$t_{plateau}$ plasma discharge plateau duration (s)
t_{sscs} CS conductor stainless steel thickness (mm)
T_{cs} current sharing temperature (K)
T_{op} coil operation temperature (K)
τ time constant of a coil during a fast safety discharge (s)
V_{loop} plasma discharge voltage (V)
W_{cc} cryogenic power for cold compressors (kW)
W_{circ} cryogenic power for the circulating pumps (kW)
W_{cl} cryogenic power associated with the current leads (kW)
W_{elec1} electrical power at the compressors for cryogenic power at 4.5 K (kW)
W_{elec2} electrical power at the compressors for cryogenic power at 80 K (kW)
W_{losses} cryogenic losses on a circuit (kW)
W_{nh} cryogenic power associated with nuclear heating (kW)
W_{pl} cryogenic power associated with pulsed losses (kW)
W_{TF} TF magnetic stored energy (MJ)
Z_{eff} effective Z to account for impurities in the plasma

ITER is a tokamak reactor. The main aspects of superconducting magnets for a fusion reactor are logically based on ITER technology. ITER can be considered as an introduction to the conceptual design of the magnet system of the reactor. A few major, not exclusive, design points of ITER are the following:

- Cable in conduit conductors are used for all magnet systems (see Section 2).
- The toroidal field (TF) coils are D shaped.
- The TF centering force is contained by wedging of the inner legs of the coils, which form a vault.
- The TF winding pack is taken in a thick steel case.
- Nb_3Sn as superconducting material has been used for the TF and the central solenoid (CS) systems, with the Wind and React technology requiring in addition a delicate Transfer process for the TF system.

Beyond these choices, which will be also selected for the future tokamak reactor, it is important to identify the design criteria which have been used in the dimensioning and check that their extrapolation is consistent.