§51. Conceptual Design of Force Free Helical Reactor (FFHR)

Sagara, A., Motojima, O., Watanabe, K., Imagawa, S., Yamanishi, H., Mitarai, O. (Kyushu Tokai Univ.)
Satow, T., Chikaraishi, H.

There have been obtained great amount of R&D results on both of physics and engineering in the course of the Large Helical Device (LHD) project which is now progressing in NIFS. In fact the LHD scaling for energy confinement and plasma density have been proposed, and the LHD is the first machine which consists of superconducting helical coils. By using these latest results, a preliminary attempt on helical type fusion reactor design has been performed to make clear the engineering issues related to a demo relevant helical reactor on the supposition of starting construction from 2,015 and power generation from 2,025.

The main features of helical type reactor are well known as

(1) steady-state operation with a small fraction of recirculating power,

(2) plasma operation with no dangerous current disruptions,

(3) natural divertor.

In addition to these, there is one more attractive feature,

(4) force free configuration of helical coils, which allows us to simplify the coil supporting structure or to use high magnetic field instead of high plasma beta. Using high magnetic field as one of the new promising design in the present work, a wide variety of force free helical reactor (FFHR) designs is investigated under common specifications of a D-T reactor with the lifetime of 30 years and the thermal output of 3GW.

In the long range scenario for economical feasibility power plants, the present work is just at the first stage of the conceptual design phase for a demonstration power plant. At the beginning of this work an highly innovative guide line, but obtainable within an extrapolated range from the LHD data base, has been exploited as follows;

(1) plasma confinement which is sufficient for ignition condition,

(2) magnetic field which is more than 3 times higher than that in LHD, thus allows one order reduction in the beta value,

(3) mechanical stress in the coil supporting structure which is comparable to that of 30 kg/mm^2 in the LHD.

These criterions are quite innovative, because there have never been proposed such targets in both helical and tokomak reactors.

In this work the design window is widely investigated from 4 aspects : (1) magnetic field design on plasma equilibrium and stability, (2) supporting structure design on electromagnetic force, (3) superconducting magnet design on coil winding and stored magnetic energy, and (4) nuclear blanket design on neutron wall loading, tritium breeding and radiation shielding. The typical results are summarized in the table [A.Sagara, O.Motojima et al, ISFNT-3, 1994].

Type of FFHR	A	B	C	remarks
major radius	$R = 20$ m			
plasma minor radius	$a = 2$ m			A_p=10
plasma volume	$V_p = 1580$ m^3			
fusion power	$P_{th} = 3$ GW			
reactor lifetime	30 years			
toroidal field on axis : Bo	*12 T*	*7 T*	*5 T*	
total stored magnetic energy : (GJ)	1290			
average beta : $<\beta>$ (%)	*0.7*	*2.2*	*4.5*	
enhancement factor of LHD scaling	*1.5*	*2.25*	*3.5*	
Helical coil aspect ratio	$A_c = 6$			
number of pole	$l = 3$			
toroidal pitch number	$m = 18$			
pitch parameter	$\gamma = 1$			tentative
maximum field on coils : $B_{perp.max}$	*14 T*	*10 T*	*9 T*	~ $0.87B_{max}$
current density : J	27 A/mm^2			or 50
coil current : MA/coil	66.6	38.9	27.8	
superconducting material	Nb_3Al , $(NbTi)_3Sn$			
Plasma maximum dencity : (m^{-3})	2E20	1.9E20	1.5E20	
maximum temperature : (keV)	22	24	29	
effective ion charge	$Z_{eff} = 1.5$			
alpha confinement fraction	$h\alpha = 0.7$			
alpha dencity fraction	$f\alpha = 0.05$			
synchrotron efficiency	$R_{eff} = 0.7$			
First wall material	JLF-1, V5Cr5Ti, or ODS-steel			2 cm
operation temperature	600 ~ 800°C			
neutron wall loading	$P_n = 1.5$ MW/m^2 for 30years			~ 450dpa
Blanket material	Flibe(40vol.%) + Be(40vol.%)			28 cm
oretation temperature	inlet 400°C / outlet 600°C			η_e ~ 30%
T breeding ratio	TBR=1.1			
structure material	JLF-1, V5Cr5Ti, or ODS-steel			
coil to plasma clearance : δL(m)	*1.1*	*1.25*	*1.3*	
Shielding material	JLF-1 + B_4C(30vol.%)			55 cm
fast neutron flux at SC (> 0.1MeV)	~ E 18 n/cm^2 / 30years at SC			~0.001dpa
Divertor material	cooled with flibe			
heat loading	$P = 1.6$ MW/m^2			45° incidect
pumping method				
Vacuum vessel				
material	Ti 6Al4V, SS316, or JLF-1			5 cm
operation temperature	T < 100 °C			by water