ISSN 0915-633X

INTERNATIONAL ATOMIC ENERGY AGENCY

SIXTEENTH IAEA FUSION ENERGY CONFERENCE

Montréal, Canada, 7-11 October 1996

IAEA-CN-64/G1-5

NATIONAL INSTITUTE FOR FUSION SCIENCE

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(Received - Sep. 6, 1996)

NIFS-447 Sep. 1996

RESEARCH REPORT
NIFS Series

NAGOYA, JAPAN

16th IAEA Fusion Energy Conference
Montréal, Canada
October 7-11, 1996

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keywords:
helical system, heliotron reactor, modular coil, force-free coil, system optimization, cost assessment, blanket design

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Abstract

DESIGN ASSESSMENT OF HELIOTRON REACTOR

Helical reactor designs are studied based on the physics and engineering concept of the Large Helical Device (LHD) which is characterized by two advantages; efficient closed helical divertor and simplified continuous-coil system.

Firstly, optimization studies of ℓ =2 conventional LHD-type Reactors (LHD-R) have been carried out. One-point plasma modelling in addition to 3D-equilibrium/1D-transport analysis clarified the D-T ignition condition. An accessible design window for reactor parameters is found using physics and engineering constraints. The cost estimation suggests the importance of the compact design to reduce the cost of electricity.

Secondly, a new reactor design candidate, Modular Heliotron Reactor (MHR) is proposed focusing on the advantage of efficient helical divertor compatible with modular helical coil system. The special coil winding system permits the appropriate coil gap for reactor module maintenance, and leads to the compatibility between the good plasma confinement and the efficient helical divertor configuration. Two MHR design options are selected based on the LHD-R system analysis.

Thirdly, based on the advantage of the simplified continuous-coil design, a high-field Force-Free Helical Reactor (FFHR), is proposed for the reduction of the electro-magnetic force by adopting ℓ =3 force-free-like continuous-coil system. The molten-salt FLiBe, LiF-BeF_2, is selected in FFHR as the self-cooling tritium breeder from the view-point of safety and the compatibility with the high magnetic field design.

1. Introduction

In helical reactor design the plasma and the helical coil systems are strongly coupled with each other, and detailed optimal plasma-coil design studies are required. At present two large optimized experimental helical machines, the Large Helical Device (LHD) and the Wendelstein 7-X (W7-X), are under design and construction[1]. The LHD concept, different from the W7-X, is characterized by two advantages; (1) efficient closed helical divertor and (2) simplified superconducting (SC) continuous-coil system.

The present LHD SC machine had been optimized based on physics criteria[2] and engineering system analysis[3]. Here, we apply the same system analysis to LHD-type continuous-coil Helical Reactors (LHD-R), and show the accessible design window. Based on this LHD-R system analysis, the Modular Heliotron Reactor (MHR) design is proposed with new modular coil system, keeping the good features of helical divertor configurations of the conventional continuous-coil design. In addition, the ℓ =3 helical reactor is studied to reduce the electromagnetic force on the continuous helical coil as the Force-Free Helical Reactor (FFHR).

Schematic drawings of these coil designs are shown in Fig. 1.

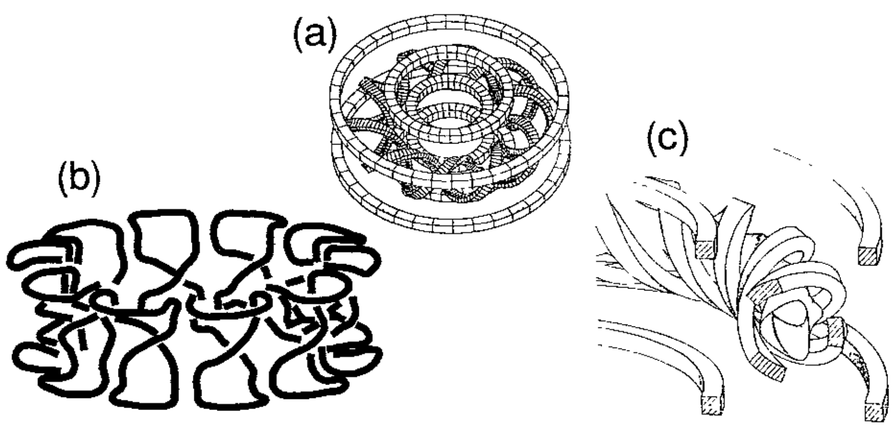


FIG. 1. Schematic coil configurations of (a)LHD, (b)MHR and (c)FFHR.

2. Optimization of LHD-type Helical Reactors (LHD-R)

The system assessment of heliotron reactors has been done starting with the LHD-type ℓ =2, m=10 coil system and the magnetic configurations. Physics optimization has been carried out by equilibrium/stability beta-limit analysis and transport model projecting the D-T ignition condition. Engineering analysis has also been performed taking into account of maximum permissible magnetic field, coil stress, wall neutron loading etc. Finally, the cost estimation has been done for the optimization of the reactor system. This design flow chart is shown in Fig. 2.

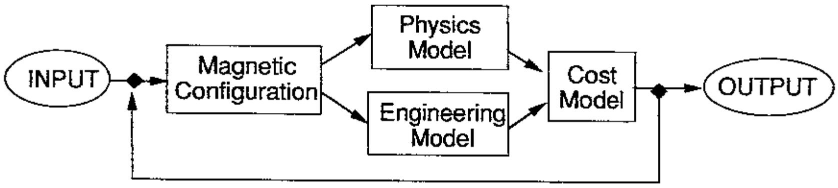


FIG. 2. Flow Chart of Helical System Optimization.

At first, we will concentrate on the ℓ =2 continuous coil reactor design. The physics properties of the LHD configuration have been widely investigated, and beta achievement (~5%) and plasma transport projections have already been clarified[2].

Ignition conditions of D-T burning plasmas in LHD-R are studied using zero-dimensional power balance equations with profile corrections based on several empirical confinement scalings (LHD, gyro-reduced Bohm, Lackner-Gottardi or International Stellarator scalings) and one-point neo-classical ripple loss model (combined model of $1/\nu$, $\nu^{1/2}$ and ν regime). The typical POPCON plot is shown in Fig. 3, which is bounded by anomalous plasma loss, neo-classical ripple loss, density limit and beta limits. Confinement improvement factor f_H (H-factor) of 2 is assumed for anomalous transport and 1.5 times larger density limit is required. The ripple transport loss is dominant in the high temperature regime, and the effective helical magnetic ripple should be smaller than ~5%.

The effect of H-factor and helical magnetic ripple on the reactor size is shown in Fig. 4. Without improvement of anomalous transport (LHD scaling in this left-hand figure), 20 meter major radius machine is required in the case of 6T. The effect of other confinement scaling laws without neoclassical ripple loss are shown in the right-hand figure. LG(Lackner-Gottardi) scaling gives rise to the most optimistic results and the LHD scaling suggests the rather high-field (high-density) compact machine. The same dependence of IS(International Stellarator) and GRB(gyro-reduced Bohm) scalings are obtained.

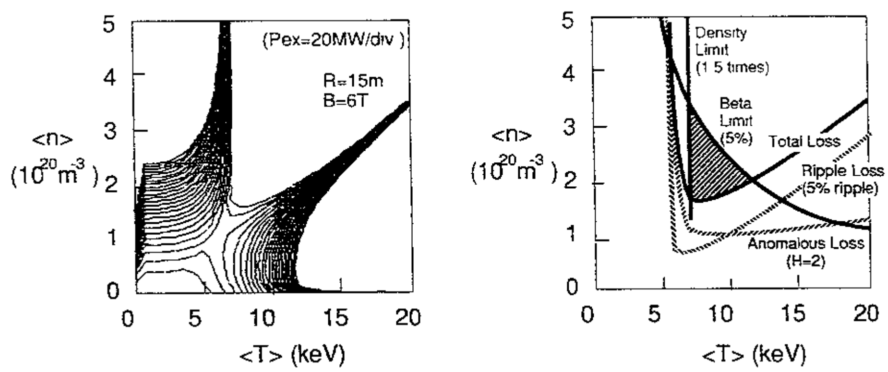


FIG. 3. POPCON Plot for LHD-R.

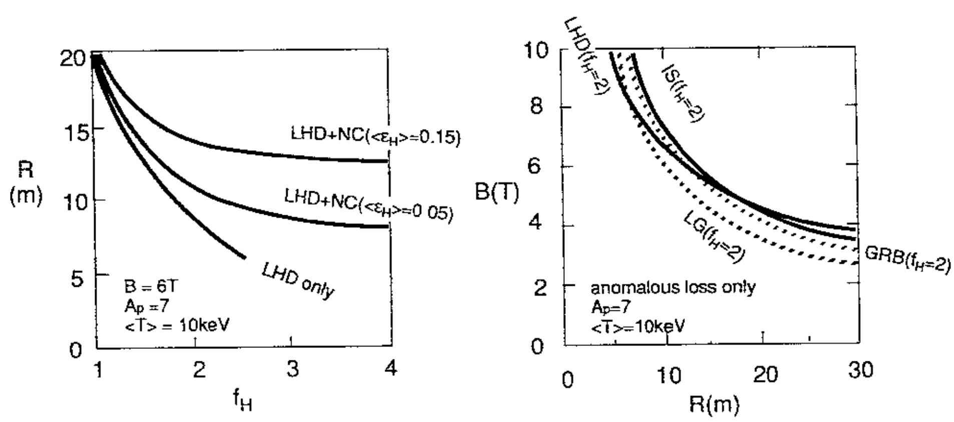


FIG. 4. Effecs of magnetic field ripple and H-factor (left) and effects of several confinement scalings (right) on ignition machine size parameters.

The detailed transport analysis using 3-dimensional equilibrium / 1-dimensional transport code[3] has been performed, and the achievement of the D-T ignition has been confirmed. Time evolution and typical radial profiles of LHD-R plasma are shown in Fig. 5. Plasma density and external heating power, P_{ex} (initially 200 MW), are feedback-controlled to produce and sustain 500 MW alpha power, P_{alpha}. Direct alpha loss power, P_{direct}, and helium ash accumulation effects are also included in this simulation. Ripple ion transport is dominant in the central region and the self-consistent negative radial electric filed E_r reduces ion heat loss in the outer region (see lower right figure).

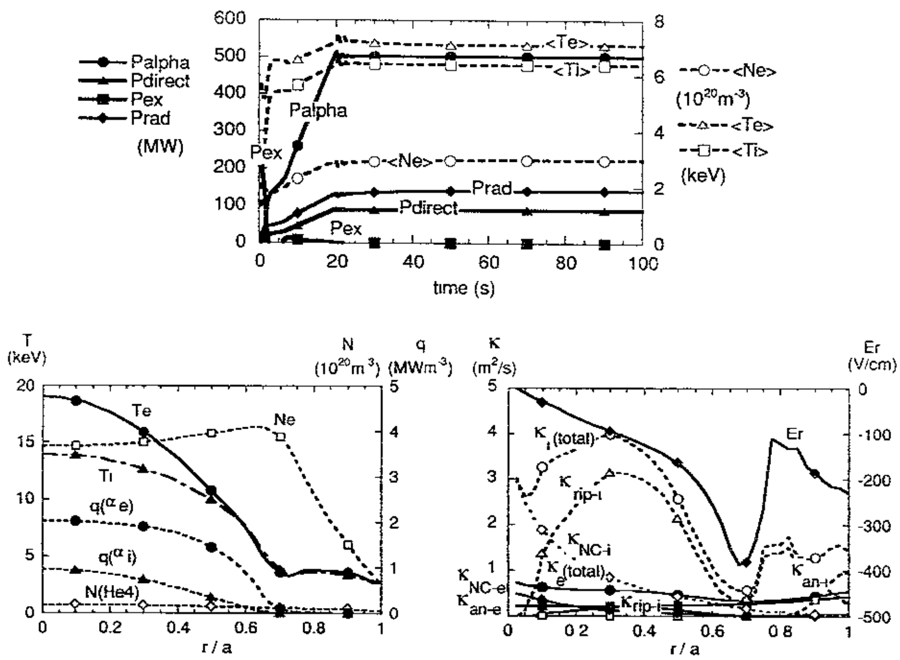


FIG. 5. Example of 3D Equilibrium / 1D Transport Simulation of LHD-R Ignition. Upper graph shows time evolution of plasma parameter for R = 12.5 m, B = 6.5 T reactor. Lower graphs show radial profiles of temperature T, electron and helium density N, alpha heating power density q(α), thermal conductivity K and radial electric field E_r at t = 100s.

The magnetic configuration and the coil-divertor clearance are evaluated by using simplified systematic scalings[4]. The system studies including engineering design criteria for Nb_3Sn SC helical coils are carried out to fit the ignition plasma condition. The reference magnetic configuration adopted here is m=10, γ_c=1.2 system with closed helical divertor. The maximum magnetic field (B_{max} < 16 T) with coil current density (~30A/mm^2) is allowed for Nb_3Sn superconducting coil systems. Neutron wall loading (L_n < 3 MW/m^2), the coil-divertor clearance (Δ_{cd} > 1 m for standard blanket space, or , Δ_{cd} > 0.4 m for compact design) and the coil stress limit (σ_{coil} < 250 MPa) should be required. The conditions of total fusion thermal power (P_{fusion} > 1 GW) and coil magnetic energy (W_{mag} < 500 GJ) should be included in the criteria for the cost-effective system. The confinement enhancement factor f_H of 2 and density limit factor of 2 (during startup phase only) are assumed in LHD scaling. To reduce neoclassical ripple loss, the averaged effective helical ripple should be less than 5 %

which is attainable in LHD-R by means of inward shift of the plasma column and/or the coil pitch modulation to the so-called transport-optimized heliotron. Among these criteria we found a design window for LHD-R as shown in Fig. 6. This analysis is also applicable to the Modular Heliotron Reactor (MHR) mentioned later.

For D-T reactors, the Standard design with major radius of 16.5 m, and magnetic field strength of 5T is obtained. The Compact system with major radius of 10.5 m and magnetic field strength of 6.5T is evaluated by sacrificing the divertor clearance. The D-^{3}He reactor analysis is also carried out and the scale requirements of these reactors are clarified with ~3 confinement enhancement factor and < 2% helical ripple configuration.

The cost estimation has been done using similar model as Ref.[7], and it is clarified that the compact model is more cost-effective than the standard design as shown in Fig. 7.

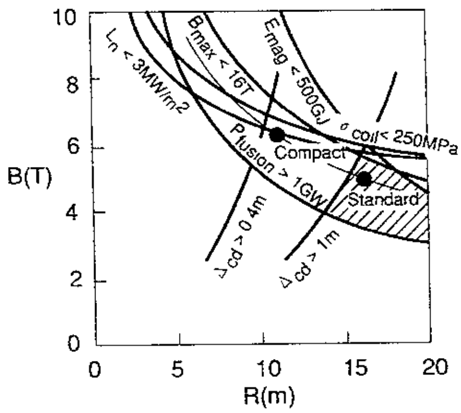


FIG. 6. Design Window of LHD-R.

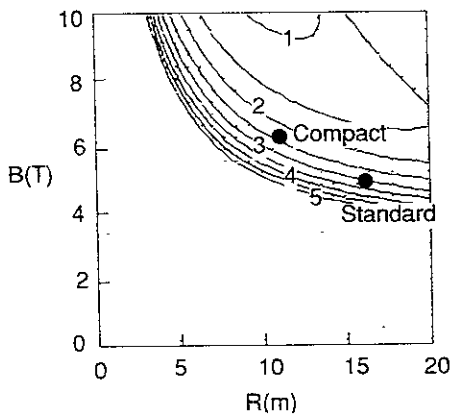


FIG. 7. Cost estimation of LHD-R. (Contours of normalized cost of electricity)

3. Modular Heliotron Reactor (MHR) Based on LHD-R

Helical system with continuous helical coils is usually difficult to make the system modular for easy reactor maintenance. On the other hand, the conventional modular stellarator is designed mainly to optimize the core magnetic confinement, and it is not easy to keep enough divertor space for heat load reduction and helium ash exhaust. It is important issue to search for a new divertor configuration compatible with modular coil system.

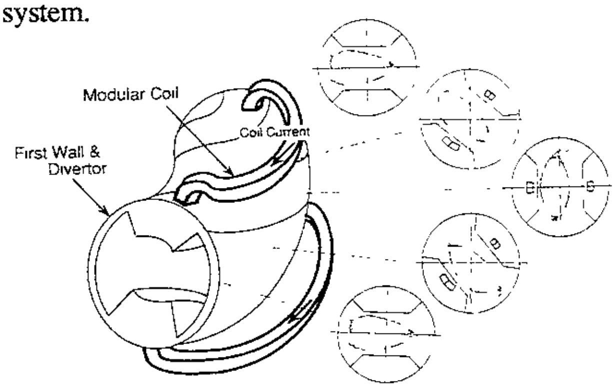


FIG. 8. Magnetic Divertor Configuration of MHR.

Based on LHD physics concept we proposed a Modular Heliotron (Ref.[5,6]). The coil system of the reference Modular Heliotron without one-turn poloidal-field coils was constructed based on the conventional Heliotron by combining the sectored helical field coils with the sectored returning poloidal field coils. The connection current feeders were arranged to avoid the destroy of the divertor layer and to keep large space for the divertor chamber. Figure 8 denotes the one coil module and the magnetic divertor structure.

The LHD design study had been carried out based on the physics concept with respect to equilibrium/stability beta limit, particle orbit confinement, divertor layer clearance, energy transport and so on[2,3]. In a similar way, physics optimization for MHR system has been done with main three criteria; the gap angle between adjacent modular coils (index of coil modularity), the branching-off of divertor separatrix layers (index of closed divertor) and the magnetic properties such as plasma radius, rotational transform, beta limit, particle confinement etc. (index of good plasma confinement). The stand-point of our proposal is that the closed helical divertor configuration with tolerable neoclassical ripple transport loss might be important in the reactor design to the improve edge confinement leading to H-mode transition in addition to the helium ash exhaust.

Good magnetic configurations are found to be produced by adopting optimum coil winding modulation (using outside-plus / inside-minus pitch modulation parameter α [3]) as a function of coil gap Δ_{gap}, even in the case of a large increase in gap angle. Figure 9 shows the equilibrium beta limit and confinement fraction defined by minimum-B contours for LHD-R($\alpha = 0$ and $\Delta_{gap} = 0^{\circ}$ (continuous coil)) and MHR(α = +/-0.15~0.3 at $\Delta_{gap} = 4\sim8^{\circ}$). The Mercier mode analysis also suggests that the stability properties of MHR is comparable with LHD-R. More optimized configurations is being searched by adjusting coil shaping parameters.

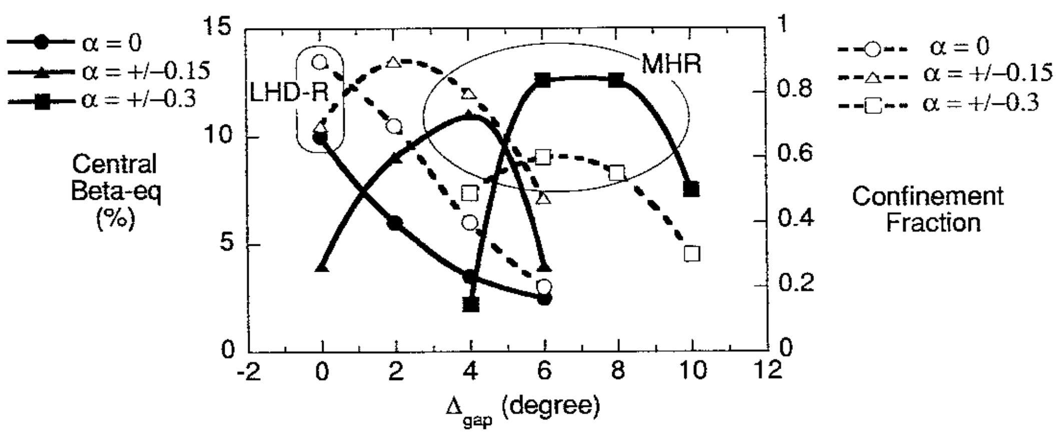


FIG. 9. Physics optimization for MHR as a function of coil gap Δ_{gap}.

The MHR system evaluation has been done using the above-mentioned continuous-coil optimization results, and typical two design candidates, MHR-C (compact design without high-field inboard blanket) and MHR-S (standard design), are selected (Table I). For ~10 meter major radius, ~6 Tesla magnetic field machine with coil current density of ~30A/mm^2, the required coil gap is larger than 4 degree (0.7m distance). The construction and maintenance concept of this reactor design is based on the 10-period good coil modularity without poloidal coils. Moreover, large ports are available to repair the helical divertor plates by the remote handling. The detailed engineering design studies of MHR are under way.

TABLE I LHD and Three Reactor Design Parameters

	LHD	MHR-C	MHR-S	FFHR-1
Plasma parameters				
number of pole : ℓ	2	2	2	3
toroidal pitch number m	10	10	10	18
major radius : R (m)	3.9	10.5	16.5	20
av. plasma raduins: $<a_p>$(m)	<0.65	1.5	2.36	2
fusion power : Pf (GW)	-	2.8	3.8	3
external heating power : P_{ex} (MW)	< 20	55	80	100
toroidal field on axis : B_0 (T)	4	6.5	5	12
average beta : $< \beta >$ (%)	> 5	5	5	0.7
enhancement factor of τ_{LHD}		2	2	1.5
plasma density : ne(0) (m^{-3})	$1x10^{20}$	$7.8 x10^{20}$	$4 x10^{20}$	$2x10^{20}$
plasma temperature : Te(0) (keV)	> 10	15.6	15.6	22
Engineering parameters				
av. helical coil radius : $<a_c>$ (m)	0.975	2.52	3.56	3.33
pitch parameter : $\gamma_c=m<a_c>/(\ell R)$	1.25	1.2	1.2	1
coil to plasma clearance . Δ (m)	0.03	0.43	1.14	1.1
coil current . I_H (MA/coil)	7.8	34.1	41.25	66 6
coil current density : J (A/mm^2)	(53)	30	30	27
max. field on coils : B_{max} (T)	(9.2)	14.7	14.9	16
stored energy with poloidal coils (GJ)	1.64	210	221	1290
neutron wall loading : P_n (MW/m^2)	-	3.4	1.9	1.5
SC material	NbTi	Nb_3Sn	Nb_3Sn	Nb_3Al or $(NbTi)_3Sn$

4. Force-Free Helical Reactor (FFHR)

Based on the SC continuous-coil system of LHD, a conceptual design of force-free helical reactor (FFHR) has been performed to make clear key issues required for power-plant engineering including materials development and to introduce innovative concepts expected to be available in a coming few decades. As the first stage for concept definition of FFHR, the reference reactor FFHR-1 has been designed [8, 9, 10, 11]. Cost estimation and design optimization are planned in the second stage in the present study of Phase-I.

Main specifications of FFHR-1 are listed in Table 1. The ℓ =3 system is adopted to obtain a force-free-like coil configuration compatible with having a sufficient space for plasma confinement as shown in Fig. 1. From the design window shown in Fig. 10, where the coil cross section is a rectangular shape, the ignition case A at $B_0 \sim 12$ T and $R \sim 20$ m is almost optimum as far as the following three parameters are concerned: the $B_{\perp max}$ in helical coils below 15T, the coil-to-plasma clearance Δ over 1m needed for blanket and shield, and the enhancement factor f_H for τ_{LHD} less than 2. Under this condition Nb_3Al or $(NbTi)_3Sn$ is chosen as a primary candidate for the SC material.

FFHR has two main features. The first is to reduce the electromagnetic force between continuously winding SC helical coils by reducing the helical pitch parameter $\gamma_c = (m/\ell)(a_c/R)$ as shown in Fig. 11, where the averaged minor-radius hoop force on helical coils $<f_a>$ normalized by B_0 and coil current I_H in FFHR-1 is reduced to 35% of the value in LHD. Since the ratio a_p/a_c decreases with γ_c, the reduction of γ_c is also needed to make a wide coil-to-plasma clearance for the blanket and shield space. In FFHR-1 the ratio a_p/a_c is about 0.6 and the coil-to-plasma clearance of about 1m is obtained. In case of the ℓ =2 system, on the other hand, since the closed magnetic surface is not formed ($a_p = 0$) at $\gamma_c < 0.9$, much reduction of the hoop force at $\gamma_c < 0.9$ is of no use for plasma confinement.

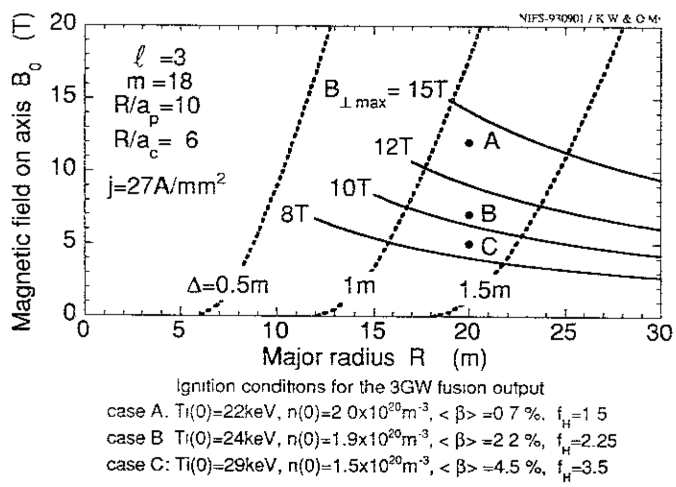
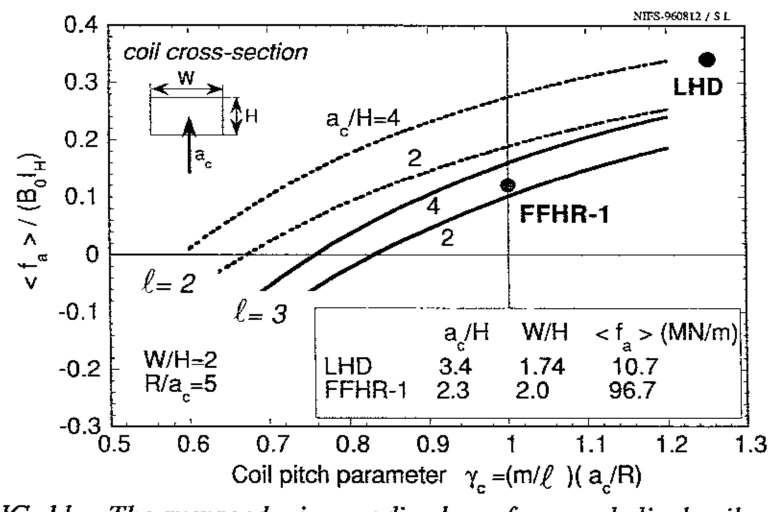


FIG. 10. The design window for B_0 and R under constraints of the maximum perpendicular field in helical coils $B_{\perp max}$, the coil-to-plasma clearance Δ, and ignition conditions.

The ideal force-free condition on helical coils is possible under the combination with vertical field coils to reduce the toroidal hoop force, but requires the decrease in a_c or increase in the helical coil current I_H to obtain a fixed value of the toroidal field B_0, because of reducing γ_c. In the FFHR design, however, the ideal force-free condition is not adopted from viewpoints of engineering safety on magnetoelastic stability and an excessive shrinkage of the plasma minor radius a_p. Even with a moderate reduction of the magnetic force, there are two attractive merits : one is simplification of coil supporting structures which gives a wide open area for the maintenance of in-vessel components, and the other is the use of high magnetic fields leading to a some margin in the plasma beta, $<\beta>$, for self-ignition with an allowable amount of He ash, and requiring a less-severe enhancement factor for the energy confinement time. In fact, as shown in Fig. 10, the case A with B_0 = 12T requires $<\beta>$ of only 0.7% , but the case C with B_0 = 5T requires $<\beta>$ of as much as 4.5% and f_H of as high as 3.5. Therefore, in the next stage of concept optimization, there is a wide flexibility to come to a compromise between two merits of simple structures and high magnetic fields.



	a_c/H	W/H	< f_a > (MN/m)
LHD	3.4	1.74	10.7
FFHR-1	2.3	2.0	96.7

FIG. 11. The averaged minor-radius hoop force on helical coils $<f_a>$ normalized by $(B_0 I_H)$ as a function of coil pitch parameter γ_c.

The second feature in FFHR is to select the molten-salt FLiBe, LiF-BeF_2, as the self-cooling tritium breeder from the main reason of safety: low tritium inventory, low reactivity with air and water, low pressure operation, and low MHD resistance which is quite compatible with the high magnetic field in FFHR-1. Due to the extremely low tritium solubility, which is about 8 orders lower than that of liquid Li [12], the T_2 gas separation system becomes quite simple. The total flow amount of FLiBe is about 7 m^3/s, and the operation pressure with a Reynolds number over 10^5 is estimated to be less than 0.5 MPa with the total pump power of only 0.8% of the fusion output P_f.

Due to the large reactor size, the neutron wall loading is reduced down to 1.5 MW/m^2, which allows to use the in-vessel components for the full lifetime of 30 years without replacing them if structural materials reliable up to 450 dpa at temperatures above 600°C are developed. Here the low activation ferritic steel JLF-1(Fe_9Cr_2W) is selected, which is hopefully used at the temperature of 550°C for up to 100 dpa, and ODS steel and vanadium alloys are the second options[11]. Fig. 12 shows the blanket and shielding structure in FFHR[8]. The Mo-TiC alloy, which has high resistance against neutron irradiation, is used for the first wall. The W-TiC alloy, which is currently under development by replacing Mo with W in view of the induced radioactivity[11], is the second option. The double walled blanket and transfer tube are covered with He gas to sweep out the permeated T_2 and to monitor drain events. The neutron multiplier Be is also used as the metal scavenger (Be + 2TF —> BeF_2 + T_2) to reduce the amount of severely corrosive TF molecules. The total tritium breeding ratio TBR is over 1.1, and the first neutron flux (>0.1 MeV) is successfully reduced by more than 5 orders at the SC coils. The volumetric nuclear heating in FLiBe is more than 60% of the fusion output P_f.

Replacement of the blanket might be required at least every 10 years. The basic design for this procedure is to use blanket units, which are replaced through maintenance ports by sliding along the continuous helical coils. At this time, since the total mass of 400 ton of FLiBe is moved to a drain tank, the weight of each blanket unit can be below 5 ton. Radioactive wastes in each replacement are 800 ton of JLF-1, 160 ton of Mo-TiC or 300 ton of W-TiC, which is only 16 m^3 in volume and can be managed, and 350 ton of Be which is the mass of recycling use.

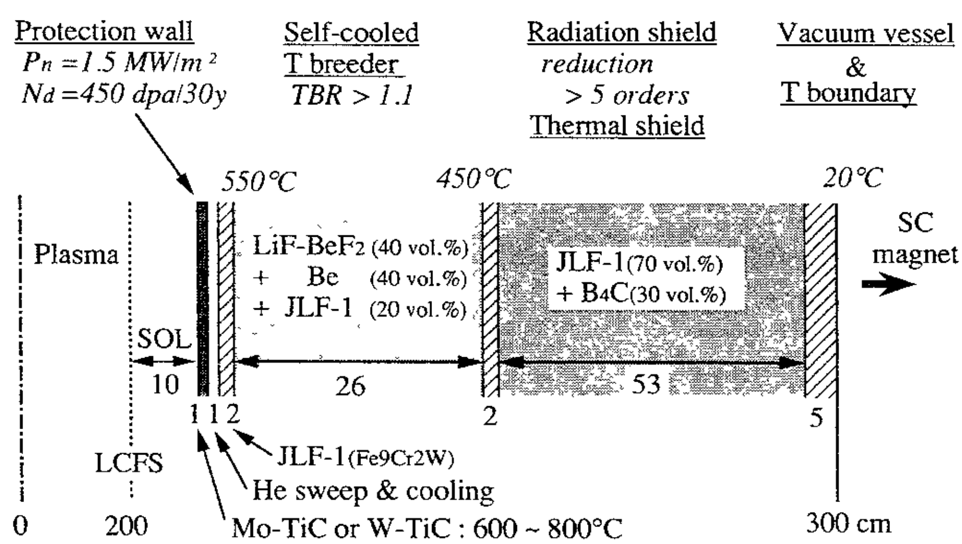


FIG. 12. The blanket and shielding structure in FFHR.

Important subjects in FFHR are under investigation : (1) optimization of the magnetic field configuration to improve MHD stability, (2) design of divertor pumping systems for He ash, (3) scenario of fabrication of large scale SC helical coils, (4) control of materials corrosion in the FLiBe system, (5) maintenance and repair techniques.

5. Summary

In this paper, we assessed heliotron reactors based on LHD physics and engineering design, and clarified the following points:

(1) Optimization studies of ℓ =2 conventional LHD-type Reactors (LHD-R) have been carried out by physics, engineering and cost analyses. One-point plasma modelling in addition to 3D-equilibrium/ 1D-transport analysis clarified the condition for DT ignition. The typical reactor design window was derived using physics and engineering constraints, and two design options were selected within the limits of the permissible blanket space, neutron wall loading, coil stress and so on. The cost estimation suggested the importance of compact designs to reduce the cost of electricity.

(2) The new Modular Heliotron Reactor (MHR) compatible with well-defined and efficient closed helical divertor was studied. The physics optimization of this system has been carried out based on the LHD (Large Helical Device) physics concept by means of vacuum magnetic surface analyses and finite-beta calculations. The effectiveness of this new coil system and its magnetic configuration was clarified. Engineering design of this Modular Heliotron Reactor (MHR) was also carried out based on this physics concept, and the required machine parameters for DT ignition were clarified.

(3) The high-field Force-Free Helical Reactor (FFHR), was proposed for the reduction of the electromagnetic force by adopting ℓ =3 force-free-like continuous-coil system, and the engineering issues for power-plant reactor were clarified. The molten-salt FLiBe, LiF-BeF_2, as the self-cooling tritium breeder was selected with the in-vessel structural material of low radioactive ferritic steel JLF-1 from the main reason of safety: low tritium inventory, low reactivity with air and water, low pressure operation, and low MHD resistance which is compatible with the high magnetic field design.

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