Advanced Study of a Tokamak Transmutation System

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Abstract

An advanced tokamak transmutation system is proposed as an alternative application of fusion energy based on a review of the previous studies. This system includes: (1) a low aspect ratio tokamak as fusion neutron driver, (2) a Radioactivity Clean Nuclear Power System (RCNPS) as blanket, (3) a novel concept of liquid metal center conductor post (CCP) as part of toroidal field coils. A preliminary feasibility study has been carried out for the system, which included the aspects of core plasma physics, blanket neutronics and design. A driver of 100MW fusion power under 1MW/m^2 neutron wall loading can transmute the amount of High Level Waste (including minor actinides and fission products) produced by 10 standard pressurized water reactors of 1 GW electrical power output. Meanwhile, the system can produce tritium and output about 2 GW of electrical energy. After 30 years of operation , the biological hazard potential (BHP) level of the whole system will decrease by 2 orders of magnitude.

1. Introduction

It is widely considered that fusion energy as a commercial energy source is still decades away. The near-term usage of fusion energy as a volumetric neutron source (VNS) for transmutation of nuclear High Level Wastes (HLW) in which the core plasma parameters and fusion technology requirements are far less stringent, would be advantageous to the eventual development of fusion power as well. For the fusion project to prosper, recently it has been proposed that fusion may be reorient itself toward a shorter-term goal, a fusion/fission hybrid [1,2,3,4]. Underlying this proposed strategy are three basic assumptions: first, nuclear (fission) power will again become important; second, natural nuclear fuel is limited in supply, necessitating breeding and the transmutation of nuclear wastes; and third, reducing proliferation dangers is important in any nuclear option. For this purpose such a fusion/fission hybrid system requires: (1)A fusion neutron driver must be operating in steady state and reliably. (2) After "burning" high level wastes and "breeding" fissile fuel in the blanket the total BHP level will be reduced, and the environmental case will be acceptable to the public. (3)The whole structure of such a device should be very simple and easy to the replace. The proposed conceptual design shown in Fig.1. would satisfy the above requirements.

2. Fusion Neutron Driver

The recent success of the START experiment has spurred great interest in the concept of the spherical tokamak (ST)[5] as a power plant (STPP) and a volumetric neutron source (VNS) because such a concept will have attractive features such as very high plasma b (~50%), low toroidal magnetic field (~2T) on axis, high self-driven current fraction f_b (>95%), and a compact fusion power core. For the ST-based fusion power plant to be economical, several engineering and physics issues must be addressed. Because of the lack of space for substantial shielding on the inboard side, a copper TF coil and also center post need to be used, resulting in large resistive losses and large recirculating power. The negative effect on the plant cost can be partially offset by operating at equilibria with intrinsically large plasma b. On the other hand, the high b also means high neutron wall loading, e.g. up to 8MW/m^2 [7], which results in severe neutron irradiation effects on the first wall (FW), especially on the center conductor post (CCP), which is used as part of the toroidal coil and requires replacement at regular intervals. We would rather increase the blanket energy multiplication than increase the power density from the core plasma to avoid severe radiation effects on the CCP as well as on the FW. Therefore it is very conservative and reliable to reduce the plasma parameter level (neutron wall loading) as low as possible in such a device.

Table 1. Two sets of core parameters of the STPP and ST-VNS

Parameters	STPP	VNS	Comment
Total fusion power [P_f][GW]	**3.2**	**0.1**	
Major radius a[m]	**3.4**	**1.4**	**Set by wall loading and fusion power**
Neutron wall loading P_w[MWm^{-2}]	**3.7**	**1**	**Balance between breeding and neutron damage**
Plasma current Ip[MA]	**31**	**9.2**	**Determined by fusion power**
Center post current[MA]	**31**	**9.0**	**~Ip to minimize recirculating power**
Aspect ratio A	**1.4**	**1.4**	**Minimum permitted by dissipation in the Center post**
Elongation k	**3.0**	**2.5**	**To maximize f_b**
Triangularity d	**0.45**	**0.45**	**Compromise between MHD stability and Center column ‘ tapering’**
b_N	**8.2**	**6.5**	**10% below MHD ballooning limit**
b_t **%**	**58**	**35**	**Low recirculating power, high P_f**
l_i	**0.1**	**0.2**	**Hollow J profile, high k**
Poloidal b_p[%]		**0.95**	
Bootstrap current fraction [%]	**0.88**	**0.81**	
Toroidal field B_t[T]	**2-2.5**	**2.5**	
Plasma edge q	**3.1**	**5.5**	**Monotonic q profile with 2nd stability access**
Average density $\langle n_e \rangle$[$10^{20}m^{-3}$]	**1.1**	**1.1**	
Average temperature <T>[keV]	**19.2**	**9.5**	
Plasma volume[m^3]		**50**	
Auxiliary power P_{aux} [MW]	**27**	**19**	

The most significant differences between the STPP[5] and ST-VNS optimization are presently under consideration. Table 1. gives two sets of possible core parameters which were analyzed.

3. The Multifunctional Blanket Concept

Studies on transmutation blankets were previously performed by Qiu L.J. and Xiao B.J.[2] In this contribution, a Th-U fuel cycle multifunction blanket is proposed, that could be used to achieve the following functions: HLW transmutation, tritium self-sustainment, fissile material breeding and energy amplification. The HLW (MA: minor actinides) from the spent fuel of fission power plants is initially loaded into the blanket in the form of pebble beds. The fission product (FP) waste (^{135}Cs, ^{129}I and ^{99}Tc) can be efficiently transmuted in the outer zone due to their large capture cross sections for low energy neutrons. In order to keep the neutron flux at a constant level with increasing in operation time, the neutron multiplier (^{233}U) should be added into the blanket after an optimized operation time, which can be easily achieved by adding ^{233}U pebbles and taking out the burned ^{233}U fuel pebbles. Natural thorium (^{232}Th) is arranged to breed fissile ^{233}U in the zone next to the MA transmutation zone for the purpose of producing nuclear fuel for fission power plants as well as ^{233}U self-sustaining in this system. At the end of each operation year, the enrichment of ^{233}U is checked so that once the enrichment is over 3%, the thorium pebbles containing ^{233}U are all unloaded and the zone is replenished with fresh thorium.

Table 2. Remaining HLW and ^{233}U producedin the blanket after 30 years

Nuclides	^{241}Am	^{243}Am	^{244}Cm	^{237}Np	^{235}Cs	^{129}I	^{99}Tc	^{233}U
Initial	**11080**	**1959**	**353**	**9825**	**4886**	**13494**	**54541**	**1**
After 30years	**3**	**23.8**	**25.7**	**3.67**	**57.6**	**18.7**	**0.007**	**249000**

(unit: 10^{24} atoms)

4. Liquid Metal Center Post

The exposed CCP. in an ST reactor will receive severe neutron damage, and resistive and nuclear heating power, and it requires replacement at regular intervals. A novel concept of liquid metal CCP is proposed[1], where flowing liquid metal Li inside a replaceable tube made of low activation materials (alloy, stainless steel etc.) coated by SiC is used as the medium carrying toroidal field coil current and removing Ohmic heat and nuclear heat. The detailed study on this novel concept has been comparing it with the conventional concept using copper or copper alloy. The novel concept has the following advantages:

(1) no resistivity increase after a long period of operation;
(2) less transmutation wastes produced, owing to regular replacement;
(3) easy removal of resistive and nuclear heat;
(4) tolerable neutron structure damage since various materials may be selected for the structure;
(5) enhancement in tritium breeding ratio.

Table 3. Comparison of DPA values for various reference reactors at P_w=1 MW/m^2

Component	**N-CCP (liquid)**	**N-CCP (tube)**	**C-CCP (whole)**	**C-CCP (FW)**	**RT-FW (FW)**	**ITER (FW)**	**ITER (FW)**
Material	**Li**	**ODS***	**Cu**	**Cu**	**Cu**	**Cu**	**316SS***
DPA/year	**No limit**	**5.8**	**3.0**	**12.1**	**14.0**	**10.1**	**9.3**

* ODS means ODS ferritic stanless steel; 316SS means 316-type stainless steel.

It is noted that a pinch effect may influence the feasibility of this kind of novel concept due to adoption of liquid metal in a strong magnetic field. Therefore, it should be seriously considered in the design. In this case, the estimated pressure in the liquid CCP deriving from the pinch effect has a 10 MN/m^2 order of magnitude.

5. Natural Divertor

Free boundary MHD equilibrium calculations show that an elongation occurs naturally in a low aspect ratio tokamak when only a simple vertical field is applied. Given significant plasma temperature (>0.1keV) and limited atomic processes, the outboard SOL can be unstable to the MHD ballooning instability with line-tying at the ends. A MHD instability model suggests that the ST tokamak plasma would have a thick SOL which contains a diverted SOL fraction of more than 90% the thickness of the SOL, three times thickness of the pressure e-folding, Δ~0.12a, but a high-A conventional tokamak would lead to a relatively narrow SOL, Δ~0.05m by CHEN Y.P.[6]. So, a wider SOL (SOL thickness $\geq$ 15cm) and a natural divertor are necessary in an ST tokamak transmutation reactor without using divertor coils, making the divertor design different from the divertor design in conventional tokamak. Diverted and thick SOL can disperse and exhaust heat over large surface areas, which can reduce the heat flux on the target plate of the divertor. On the other hand, the natural divertor target plate must occupy a small space and handle a large heat flux, which makes the structural design of a natural divertor more difficult.

Modeling results show the plasma parameter distribution in the computational SOL region, especially on the target plate of a natural divertor.

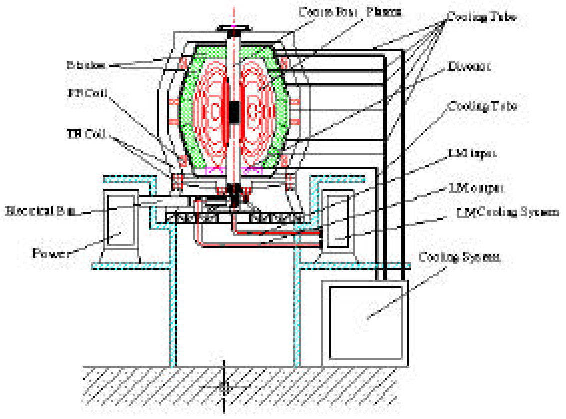

Fig.1 Overview of ST-VNS.

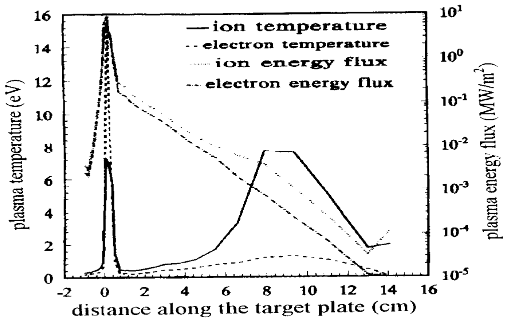

Fig.2. Plasma parameter distribution

The simulation results for edge plasma parameter distribution near the natural divertor plate are shown in Fig.2. When radiation gas (cold He gas, energy E_0=0.1eV,flux $\phi = 1.3\times10^{23}/m^2.s$) is puffed into the divertor room from the bottom of the natural divertor, simulation results show the total energy flux peak value P_{max} is reduced to 3.861MW/m^2. He gas puffing is applied to the design. Some simulations are done for SOL thickness Δ=5 cm in a conventional tokamak reactor. In the simulation, the other plasma input parameters are the same except for the SOL thickness Δ.

Table 4. Main difference of plasma parameters near divertor target plate

Peak temperature [eV]	SOL thickness Δ=15cm	gas puffing Δ=15cm	SOL thickness Δ=5 cm
Ion temperature	7.692	7.166	7.853
Electron temperature	15.55	0.9557	78.85
Ion energy flux [W/m^2]	5.222	2.241	4.084
Electron energy flux [MW/m^2]	8.965	1.620	25.05
Ion density[10^{22}/m^3]	0.5361	0.946	0.1868
Pressure [Pa]	2576	2678	2723
total energy flux [MW/m^2]	13.37	3.861	27.92

When the SOL thickness Δ=5cm, high plasma temperature and large energy flux value exist near the divertor target plate. But , when Δ =15cm, the plasma temperature and energy flux valules near the divertor target plate are reduced, and temperature and energy flux distribution rangles are expanded along the divertor target plate. So a wider SOL can reduce damage of the divertor target plate, which is a benefit for a low aspect ratio tokamak. According to simulation results, the main difference of plasma parameters near the divertor target plate among Δ=15cm, Δ=15cm (gas puffing) and Δ=5cm is shown in table 4.

6. The Novel Design Method and Structure

The design process of a nuclear fusion reactor is very complex and often changing, even the AutoCAD (Auto-Compute Aided Design Software Package) has been used very widely in the field of Fusion Engineering Design. To contrast performance and parameters for nuclear fusion device design in projects (by X.P.LIU), system design supported by variation and visualization has been necessary. For a fusion reactor to be used as research device, a novel design method guided by parametric component design and variation assembly logic relations is presented. Parametric component design includes center conductor post design, PF coil design, TF coil design, divertor cassette design, blanket design, shielding design, vacuum pumping design and so on. Solid modeling technology is used for all design objects, and 3D representation and simulation functions are open to user. Some parts of the design are shown in Figs.3 and4.

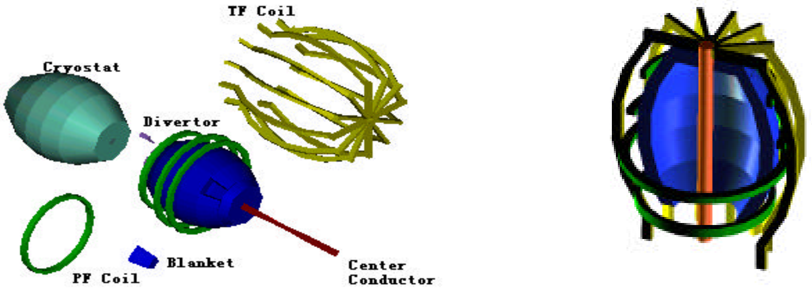

Fig.3 Overview of ST design results

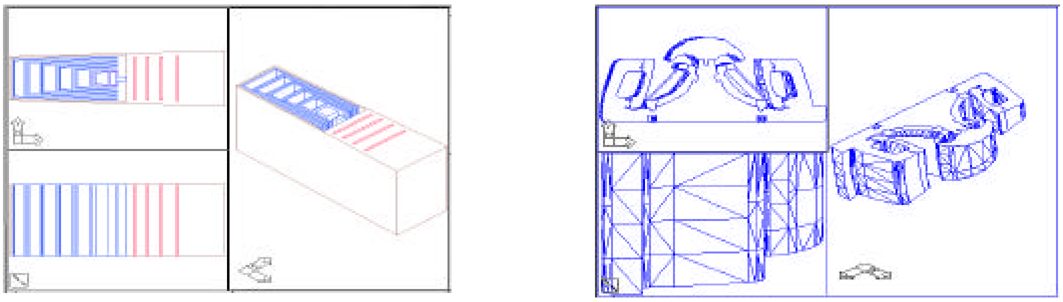

Fig.4. Blanket and Divertor of a transmutation reactor

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