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Stationary Compact VNS Tokamak for Transmutation

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Abstract. The concept of stationary VNS tokamak with aspect ratio A=2 based on moderate physical and technical assumptions is presented. Non-inductive stationary mode can be achieved by using of tangential NBI. Using of the confinement multiplier $H_{IPB98(y,2)}$=1.6 was shown to be enough to getting a stationary mode with P_{NB} ~ 45 MW. The dependence of NB absorption profile on neutral energy requires to apply neutral beams with energies 140 and 500 keV. The carried out calculations have shown possibility of tangential NBI to provide fully non-inductive current drive. The portion of bootstrap current f_{bs} is ~ 0.4. Efficiency of CD is ~ 0.06 A/W. The beam-target interactions during the fast ion slowing down are taken into account. An averaged neutron loading is about 0.3 MW/m^2. As specific Ohmic heat in toroidal coils does not exceed 12 W/cm^3, the heat can be removed from coils by usual ways. The divertor problem is suggested to solve by using of a design similar ITER with 24 reception cartridges. The blanket with neutron multiplication factor k_{eff} ~ 0.95 and lithium heat-transfer agent was chosen. The volumetric blanket power is ~ 15 MW/m^3.

1. Introduction

Effective utilisation of waste nuclear fuel with the aim to use its energy potential and to achieve radiation equivalence of buried fission products is one of the most important tasks in 21 century's nuclear energy. One of the actively discussed ways to solve this task is to make use of a volumetric thermonuclear neutron source for transmutation of long-living high activity wastes of nuclear power stations [1-4]. Tokamak is considered as such neutron source because it is the most physically and technically developed thermonuclear device. Beginning from the end of 80-s, several concepts of tokamak based volumetric neutron source have been proposed [4-7] both for material research and transmutation. At the present time, the most detailed considerations of tokamak transmutator can be found in papers by Stacey et al., Qiu et al., Cheng et al. and in papers by scientists from TRINITI, NIIEFA and NIKIET (Russia).

2. The Physical Characteristics of VNS

The Russian concept of VNS for transmutation utilises the advantages of low aspect ratio tokamaks enabling one to create compact volumetric source with warm coils and effective blanket. This concept is based on the following physical and technical backgrounds:

- Aspect ratio A = 2, on the boundary between spherical and classical tokamaks;
- Moderate parameters of plasma column (R=2 m, a=1 m) with elongation $k_{95} \approx 1.7$ and single null configuration;
- Total fusion gain factor Q ≈ 2;
- Beta normalised $\beta_N \leq \max(3, 5 \cdot l_i)$;
- Fusion power $P_{fus,\Sigma} \approx 50 - 60$ MW;
- Confinement time corresponds to the ITER scaling IPB98(y,2) at the confinement enhancement factor H≤ 1.6;
- The main heating method is neutral deuterium beam injection with two particle energies (140 and 400-500 keV). Neutral beams are used for plasma heating and current drive. Besides D-T reaction between beam and plasma target contributes to total neutron

production.
- Combined current drive scenario;
- The blanket and protection layer are located between plasma and the toroidal coil on the outboard part of the vacuum vessel; the solid angle filling by the blanket $k_{BL.} \geq 0.7$;
- Total electric power supply $P_{site} \leq 500$ MW.

Choice of aspect ratio A=2 is dictated by a number of simple considerations. First, it is possible at A=2 to use combined method of current initiation and ramp up without significant technical problems. Second, configuration with A=2 enables us to use positive features both low aspect and classical tokamaks. For latter ones, the fundamental database has been created necessary for developing projects with strong technical backgrounds. Simplified parametric analysis shows that the main advantages of A=2 configurations are: more optimum scenario of current ramp up and maintenance, more efficient utilisation of neutron flux, minimum cost of both tokamak itself and its exploitation. Numerical modelling of scenarios and characteristics of VNS were carried out by means of the 1.5D DINA code [8]. The main parameters of the VNS are summarised in Table 1. They were calculated at various energies of injected neutral beams and fixed value of confinement factor $H_{IPB98(y,2)}$.

TABLE 1: MAIN PARAMETERS OF VNS TOKAMAK.

	I	II	III	IV	V
I_p, MA	5.00	5.3	5.7	5.0	5.2
β_p	1.2	1.3	1.3	1.3	1.4
R, m	2	2	2	2	2
τ_E, ms	412	427	453	415	423
n_{e20}, m^{-3}	1.0	1.0	1.0	1.0	1.0
κ	1.7	1.7	1.7	1.7	1.7
P_{NBI},MW	45.	45.	45.	20/25	20/25
T_e, keV	6.8	7.1	7.6	6.8	7.0
T_i, keV	7.1	7.5	8.0	7.1	7.1
q_b	7.1	6.8	6.3	7.1	7.0
q_0	2.9	1.4	1.1	1.3	1.0
n_e/n_{GW}	0.638	0.604	0.563	0.633	0.619
I_{bs}, MA	2.542	2.181	1.998	2.181	2.027
W_p, MJ	20.900	22.153	23.709	20.935	21.319
Γ_n, MW/m^2	0.311	0.352	0.373	0.297	0.298
Q	1.217	1.374	1.462	1.164	1.169
β_N	3.189	3.550	3.909	3.531	3.752
P_{B-TI}, MW	25.429	28.204	28.416	23.002	22.391
P_n, MW	46.587	52.732	56.015	44.576	44.737
I_{NB}, MA	2.459	3.104	3.750	2.847	3.102
f_{bs}	0.5089	0.4142	0.3411	0.4433	0.3922
$B_t(0)$, T	3.9	3.9	3.9	3.9	3.9
$H_{IPB98(y,2)}$	1.6	1.6	1.6	1.6	1.6
E_{NB}, keV	200.	300.	400.	140/400	140/500
T:D	0.7: 0.3	0.7: 0.3	0.7: 0.3	0.7: 0.3	0.7 :0.3

It was shown that more than 30% of plasma current can be introduced by inductive means without inductor current reversal. This makes it possible to remove the central solenoid from

VNS at the steady state and thus to exclude long radiation load on CS materials and to lower neutron loss. Further current ramp up is created by means of current drive due to neutral beam injection and bootstrap current. Time to reach a steady state is defined by skin processes and is ~ 60 minutes. Stationary current is maintained by current drive from 140 and 400 keV neutral beams and bootstrap current. CD efficiency is about (0.07-0.1) A/W, that agrees with experimental data. Thermonuclear gain factor Q is about ~ 1.4 and increases with increase in neutral beam power. Fraction of fast particles in plasma pressure from beam injection is not higher than 10%. Changing the deuterium to tritium concentration ratio enables one to change in some limits the neutron output and hence the neutron flux to the blanket.
Plasma column with elongation k_{95}= 1.7 is vertically unstable demanding to use a control system. It is assumed in the JUST-T concept that there can be disruptions including vertical disruptions. Vertical disruption modelling including halo currents has been performed by means of the DINA code. Disruption was considered at the stationary state of discharge. Disruption process consists of two stages: slow VDE (Δt ~150 ms) when the main displacement of the plasma column takes place and quick VDE (thermal quench with Δt ~1ms), during which halo currents are induced in the vacuum vessel. Toroidal and poloidal components of halo currents are similar in value (about 1.5 MA). These currents must be taken into account when calculating the JUST-T vacuum vessel strength.

3. Design of VNS

The geometrical configuration of VNS with blanket arranged outside of vacuum vessel is shown in Fig.1.

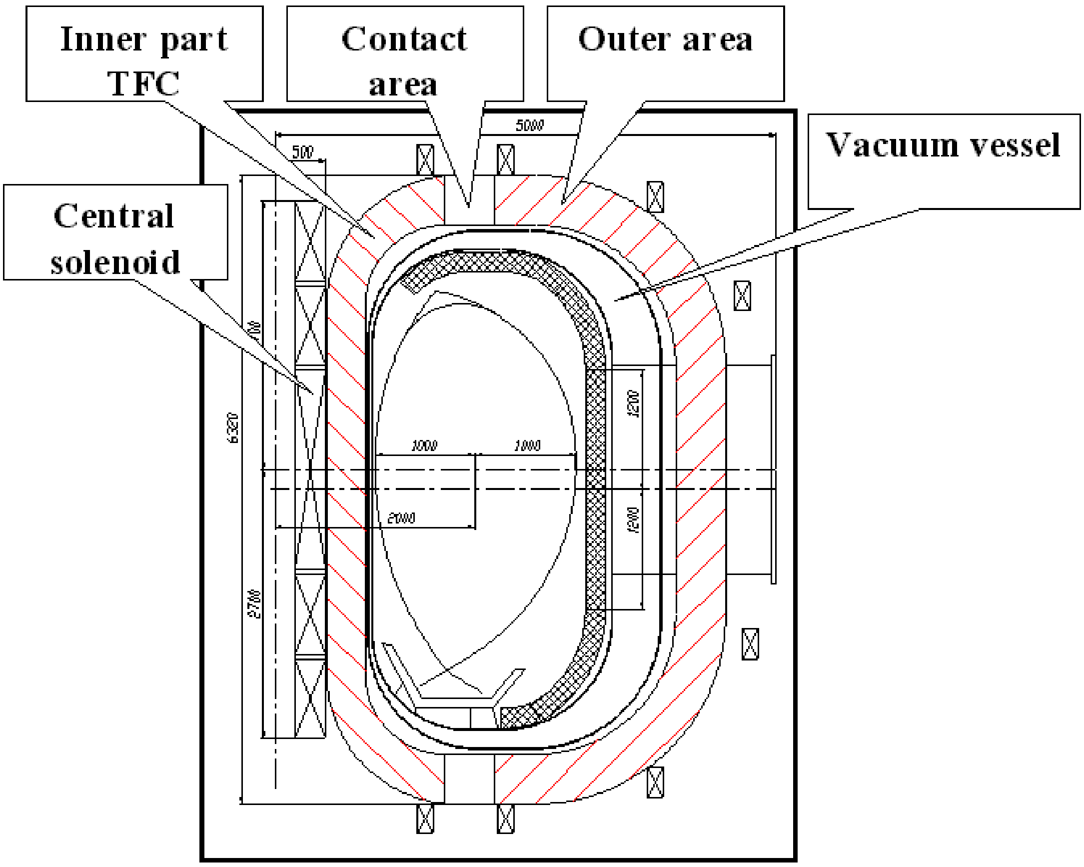

FIG. 1. The structure of VNS tokamak for transmutation

3.1. Parameters and Main Characteristics of Magnetic System, Vacuum Vessel and Divertor

Toroidal magnetic system consists of 20 coils. Ripples due to finite number of toroidal coils are about 0.2%. Current in toroidal coil is about 2 MA. The coil is multiturn with turn number being equal to 50 and coil current 40 kA.

The toroidal coil especially its inboard surface is under powerful radiation fluxes. Copper is used as toroidal coil material. The toroidal coil will be manufactured from buses with ceramic isolation.

Two variants of construction of the toroidal coil have been considered: detachable and permanent. In the first case, the separate central part must has possibility to be replaced during exploitation of VNS. It is connected with that the blanket containing minor actinides is located only on the outboard, upper and partly on lower part of vacuum vessel (excluding divertor). On inboard part, there is no protection, and resource of the central part of the toroidal coil is much less than that of its other parts which leads to much number of replacements. But there can be difficulties to provide detachable electric contact and its exploitation.

In the case of joint, the inner part of the toroidal coil is assembled in the form of separate unit and can be replaced. In this case, the vacuum vessel is detachable and consists of 2 parts. External parts of the toroidal coil is attached to the inner parts and packed out in contact zones after installation of the vacuum vessel. The contact zone is a slot connection compressed with wedge connectors from both sides. This provides good electric contact and strength.

Let us consider a possibility of creation of stationary toroidal magnetic field in the VNS tokamak. Assume the following geometry of the toroidal coil:

- Plasma major radius $R_0 = 2$ m
- Toroidal field at the radius R_o $B_t = 3.9$ T
- Internal radius of the central column $R_1 = 0.54$ m
- External radius of the central column $R_2 = 0.94$ m
- Central column height $H = 6.3$ m
- Magnetic field rippling on the plasma boundary 2%

The main parameters of toroidal field coils are presented in Table 2.

TABLE 2: PARAMETERS OF TOROIDAL FIELD COILS.

Parameter	Value	Unit
Total current	39 000 000	A
Cross section of the conductor of the central cylinder	1.91	m^2
Minimum current density	2.04×10^7	A/m^2
Copper mass of the central cylinder (maximum)	108	ton
Length of inner circle	3.2	m
Length of outer circle of the cylinder	5.9	m
Minimum electric power for current generation at given dimensions (copper at 20 ^{o}C)	80	MW
Number of sections (legs) of toroidal coil (not less)	20	

Water cooling of the poloidal coil is provided through the hole in the conductor center. Assuming that water velocity is V and channel length is not more than 6 m, one can obtain preliminary data of cooling regime.

The some additional characteristics of toroidal field coils are presented in Table 3.

TABLE 3: ADDITIONAL CHARACTERISTICS OF TF COILS.

Name, units	Nominal
Square cross section bus dimension, cm	2
Isolation thickness, mm	1
Hole diameter, cm	1
Number of conductors in the bus	14
Number of turns in 1 section	14
Total number of turns	280
Power source current, A	139 000
Current density, A/m^2	$3.1 \cdot 10^8$
Source power, MW	122
Power source current voltage, V	900
Total cross section of cooling holes, m^2	0.31

Let us adopt that convective heat transfer from the wall is 10000 W/m^2 degree, water speed is 5 m/s, current starts at T=0 with density J_0- 31 A/ mm^2. A set of equations for 1D model of non-stationary heat transfer consists of equation of thermal balance in the wall with density of internal heat sources $q_s = J_0^2 \times \rho(T)$, equation of energy conservation for incompressible liquid. Numerical solution of this set of equations provides graphs for distribution of wall and water temperature against time and channel length. From these graphs one can see that stationary temperature of the poloidal coil conductors is not higher than 90 °C. The central solenoid (CS) must provide current formation and ramp up to $I_p \approx 2.5$ MA by means of inductive method. Necessary reserve of the magnetic flux in solenoid (≈ 6 B·c) can be provided by means of solenoid discharge from maximum current to zero current. Solenoid with zero current is switched off or removed from the radiation zone. The central solenoid is placed behind the toroidal coil in a zone where it can move in vertical direction. The central solenoid is sectionalised. The sections are mounted on a single column. The total height of the central solenoid is 5.5 m. Magnetic field on the central solenoid axis is 16 T. Total current is ≈ 70 MA·turns. Evaluations show that stresses in central solenoid conductor will be ≈ 150 MPa. It is assumed to use silver containing bronze with isolation from poliamid film as material for the central solenoid conductors.

Poloidal fields system used to provide necessary form and position of the plasma column and single null configuration consists of 8 turns and is located outside the toroidal system.

Two variants of blanket location have been considered. In the first one, blanket with protection is located inside vacuum vessel and plays role of passive coils. In the second variant, blanket is located between the vessel and the toroidal coil. In this case passive coils must be placed into the vessel. Both variants have positive and negative features, so the final decision must be made after more elaborate development of the design. It is assumed in calculations that electromagnetic pressure on the upper side of the vessel from halo currents is not more than 1.35 MPa and the total vertical force on the whole vessel from halo currents is not more than 5 MN. Calculation results show that the most stressed vessel elements are flat bottoms and internal toroidal transitions. For both variants, the maximum deformation from mechanical loads is mainly determined by vertical displacement of the central cylinder and reaches value of 2.4 mm for one-layer vessel and 1.8 mm for two-layer vessel. Calculations enable us to conclude that conditions of static strength are satisfied under normal exploitation for both variants of vacuum vessel. At equal safety factor, the two-layer vacuum vessel is lighter than one-layer one by 14%. Geometrical characteristics of the vacuum vessel are determined by the blanket dimensions. It is adopted that the blanket with thickness ≈ 200 mm

containing minor actinides is located only on the outboard, upper and partly on lower part of the vacuum vessel (excluding divertor). Inner vessel surface has no protection. The vacuum vessel must provide vacuum not less than 10^{-5} Pa, wall heating up to 400°C. One- and two-layer variants of vacuum vessel have been considered.
In one-layer variant, the vacuum vessel consists of central internal cylinder with radius 935 mm, flat upper and lower bottoms and external cylinder with radii 3100 and 3700 mm. The height of the transverse cross section of the vessel is 4600 mm, width – 2165 mm. In the equatorial plane, 10 rectangular cross-section 2000x510 mm horizontal pipes are connected to the external cylinder of the vessel (4 of them is tangent to the plasma column axis).
In the second variant, the wall is two-layer, thickness of the layers of the internal cylinder is 10 mm, the thickness of the layers of the other parts of the vessel, poloidal and toroidal stiffening ribs is 15 mm. Distance between wall layers is 60 mm. Vacuum vessel and in-vessel elements mass is about 50 tons, material is stainless steel SS 316 LN.
In case of permanent toroidal coil, construction and service works are much easier. In this case the toroidal coil is mounted on each half of vacuum vessel from both sides, positioned and fixed. Disadvantage of such a construction is that the whole electromagnetic system should be replaces when resource of the central part is exhausted. In detachable variant, the current transition from turn to turn is assumed to organise on the outboard part by means of separation of output ends along the surface. Connection of coils will be created by means of diagonal current jumpers.
Preliminary evaluation of forces acting on the toroidal coil has been done. Total compressing force is ≈ 80000 tons. Total disruption force is ≈ 30000 tons. Total tilting force is ≈ 300 tons. Tilting moment acting on the toroidal coil is ≈ 1200 tm. Total weight of the toroidal coil is 300 tons.

3.2. Divertor Heat Loads

It is assumed that 70% of the plasma heat power P_{out} gets through to the divertor and the power distribution between the outer and the inner legs is 3:1. In this case the power load on the outer divertor plate is $P_{out,e} \approx 30$ MW. The major radius of the outer plates $R_{div.\,e} \approx 2$ m.
The SOL width estimation is $\Delta_{SOL,PK} \sim 2$ cm. If the flux tube width is 3 times larger at the divertor plates as compared to the midplane (the flux widening factor is 3) then the energy deposition width is $\Delta \sim 6$ cm and the heat load at normal incidence is $P_{div.e} \approx P_{out,e}/2\pi R_{div,e}\Delta \approx 40$ MW/m^2. At the tilt angle of the divertor plates ~ 15÷20^o , the heat load drops to 10÷12 MW/m^2. The radiation losses in the divertor volume reduce this load.

3.3. Estimation of Efficiency of Minor Actinides Transmutation in Blanket

Calculations were made for a spherical blanket 20 cm in thickness with 50 cm- thick biological protection layer being behind it. Neutron load was taken to be 0.423 MW/m^2. Table 4 presents the contents of the blanket zones.

TABLE 4. CONTENTS OF BLANKET ZONES

Zone 1	Width – 1.5 cm, 50% steel, 50% heat transfer agent,
Zone 2	Width – 5 cm, 15% steel, 50% heat transfer agent, 35% MA,
Zone 3	Width – 15 cm, 15% steel, 40% heat transfer agent, 45%MA,
Zone 4	Width – 50 cm, 75% steel, 25% water

MA were presented by dioxide with density of 9 g/cm^3, isotope contents: Np^{237} – 30.8%,

Am^{241} – 65.4%, Am^{242m} – 0.02336%, Am^{243} – 3.8% (corresponds to MA contents in irradiated fuel of the LWR-1000 reactor after 40 years holding time). Three heat transfer agents were considered: liquid lead (case1), liquid lithium with natural isotope content (case 2), and water (case 3). It is also possible to use liquid salt blanket.
The fission rates of actinides and integral characteristics of the JUST-T blanket are presented in Table 5. It is supposed that 82.8% of plasma neutrons impinge on the blanket. Location of MA inside the blanket makes it possible to obtain high fission power or transmutation of MA. Blanket without water has much higher K_{eff} value and, correspondingly, higher fission power. The table data are the upper estimation of possible blanket parameters showing the potential capabilities of the thermonuclear reactor. Engineering development of the blanket design in cases 1 and 2 taking into account the real first wall coverage by the blanket, restrictions due to heat exhaust, coil protection, etc, will lead to 3-5 times lower fission power inside the blanket. But even in this case, one reactor-transmutator with plasma power 56 MW can provide transmutation of MA from 10-15 LWR-1000 type reactors. The following geometry and contents of the blanket were considered.

TABLE 5: Integral characteristics of the blanket.

Fission events per one DT neutron	Variant 1	Variant 2	Variant 3
Zone 2	2.553	1.868	0.659
Zone 3	6.844	5.172	1.233
Total	9.397	7.04	1.892
K_{eff}	0.966	0.955	0.863
Fission heat power, MW	56.3	42.2	11.3
Average fission power, MW/m^3	282	211	56.7
Mass of transmuted MA, kg/year	22.2	16.6	4.44
Fission heat power, MW	6176	4627	1243
Amount of supported LWR-1000	68	51	13.6

4. Preliminary VNS Cost Evaluation

The main units are electromagnetic system (EMS), auxiliary heating system (AUX), blanket (BL), divertor (DIV). The main contribution to the total cost of the EMS is from toroidal coils (TF). It is assumed $C_{EMS} \approx 1.5\ C_{TF}$. The unit cost of normally conducting magnetic system is assumed to be 0.1 M$/ton. Toroidal coils weight - 400 tons, so EMS cost is about 60 M$.
For auxiliary heating system the conventional unit cost is about 2 M$/MW, so the 45 MW auxiliary heating system costs about 90 M$.
The ITER blanket unit cost ~0.4 M$/m^2 gives ~ 40 M$ cost for surface ~100 m^2. The blanket of the VNS for transmutation of minor actinides is substantially more powerful and complicated and should provide tritium reproduction, so the above value should be increased up to $C_{BL,\Sigma} \approx 100$ M$.
The VNS divertor is similar to ITER and 3 times as smaller which gives $C_{DIV} \approx 40$ M$..
Besides these subsystems it is necessary to add the electric power supply system (POWS) and the system of energy transfer (HEATR).
Similar to ITER, electric power supply system costs $C_{POWS} \approx 40$ M$. The system of energy transfer from the blanket seems to be a substantial part of the total cost. Expert evaluation of such system gives $C_{HEATR} \approx 150\div 200$ M$. As a result the total cost of the VNS for transmutation of minor actinides can be $C_{\Sigma} \approx 400\text{-}500$ M$.

Conclusion

On basic of the moderate physical and technical assumptions and achieved experimental parameters the concept of compact tokamak as a volumetric neutron source for transmutation was developed. Tokamak can also be used for physical studies of fusion plasma in stationary conditions and for material investigations and tests for support of future fusion energetic.

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