18

ИНСТИТУТ ЯДЕРНОЙ ФИЗИКИ СО АН СССР

Ya.I. Kolesnichenko, V.P. Nagornyj

A NEUTRON GENERATOR BASED ON A THERMONUCLEAR DEVICE WITH A HIGH ALBEDO BLANKET

PREPRINT 91-23

НОВОСИБИРСК

A Neutron Generator Based on a Thermonuclear Device With a High Albedo Blanket

Ya. I. Kolesnichenko, V. P. Nagornyj

Institute for Nuclear Research

Academy of Sciences of the Ukr.SSR, Kiev, USSR

ABSTRACT

The paper presents the concept of a neutron generator based on a thermonuclear device with a blanket acting as a neutron reflector. It is shown that in such a generator the neutron flux to the test sample of a material is much higher (by a factor of several tens) than the flux in an ordinary thermonuclear device. Neutron generator of this type based on a small two-component tokamak, is considered. The neutron flux in an ITER-type tokamak reactor with a neutron reflector is estimated.

1. INTRODUCTION

Neutron generators are needed for many practical purposes. One such purpose is radiation testing of materials, which is necessitated by the problem of the first wall of a fusion reactor. A neutron generator serving this purpose should supply a neutron flux to the test sample which over the time of the experiment would give a neutron fluence equal to 10 MW y/m^2 for the wall of the demonstration reactor (DEMO) and 40 MW y/m^2 for a commercial reactor [1]. The generator should produce neutrons with energies $E =$ = 14.1 MeV and also with lower energies (the neutron flux to the first wall can be essentially higher than the initial flux of particles with an energy of 14.1 MeV).

The concept of a neutron generator with such properties is presented in this paper. The proposed generator is a thermonuclear device in which neutrons are accumulated in a special way. The thermonuclear device can be virtually any steady-state (quasi-steady-state) system with magnetic plasma confinement (tokamak, stellarator, open trap, etc.) or even a pulsed system. However, the calculations made in this paper do not apply to the latter.

The characteristics of our neutron generator based on a thermonuclear device differ essentially from those of earlier proposed neutron sources [2-5]. These have an almost monoenergetic spectrum of neutrons with $E = 14.1$ MeV and differ from a fusion reactor of the relevant type only by their smaller dimensions and the energy gain factor ($Q \ll 1$), thus having correspondingly lower neutron fluxes than the reactor. In particular, Vasil'ev et al. [4] described a neutron source based on a centrifugal trap, while Hendel and Jassby [5] analysed the potential of tokamaks as neutron sources. Only Mirnov et al. [2] and Berk and Ryutov [3] considered a special neutron generator (based on an open trap) where the neutron flux density could be increased locally by a factor of up to 6.

Section 2 describes our method of amplifying the neutron flux in a thermonuclear device. In the same section we have performed evaluations for a neutron source with neutron accumulation based on an ITER-type tokamak reactor. In Section 3 we give the results of calculations for a neutron source based on a small two-component tokamak.

2. INCREASING THE THERMONUCLEAR NEUTRON FLUX

Let us assume that the plasma which serves as the source of 14 MeV neutrons is surrounded by a blanket of nonfissile material with a high neutron albedo (β_n). Then, if we neglect neutron multiplication in the blanket material, the following neutron balance equation holds:

$$\frac{\partial N}{\partial t} = \dot{N}_+ - \Gamma S + \Gamma\beta_n (S - S_0), \qquad (1)$$

where N is the number of neutrons in the volume bounded by the inner surface of the blanket, $\dot{N}_+$ the rate of neutron production by nuclear fusion reactions, Γ the density of the neutron flux from the plasma to the blanket inner surface, S the area of this surface, and S_0 the cross-sectional area of the openings in the blanket associated with the structural features of the device. Equation (1) is written on the assumption that neutrons which go through the openings in the blanket do not return to the plasma. For simplicity's sake it is further assumed that the plasma and the blanket have circular cross-sections so that $\Gamma \simeq$ const on the surface and we can represent the neutron flux in the form $\Gamma \cdot S$. In the steady state we obtain from (1):

$$\Gamma = \frac{\Gamma_0}{1-\beta_n(1-S_0/S)}, \qquad (2)$$

where $\Gamma = \dot{N}_+/S$ is the density of the neutron flux to the inner surface of the absorbing blanket or in the absence of the blanket (at the same points).

Hence it follows that the flux density is higher than Γ_0 by a factor of $K_* = [1 - \beta_n(1 - S_0/S)]^{-1}$. If the thickness of the blanket is small in comparison with the radius of its inner surface ($L_b \ll a_b$, see Fig. 1), the quantity K_* characterizes the increase of the neutron flux to a sample located outside the blanket near the openings.

If however, the sample is placed between the first wall

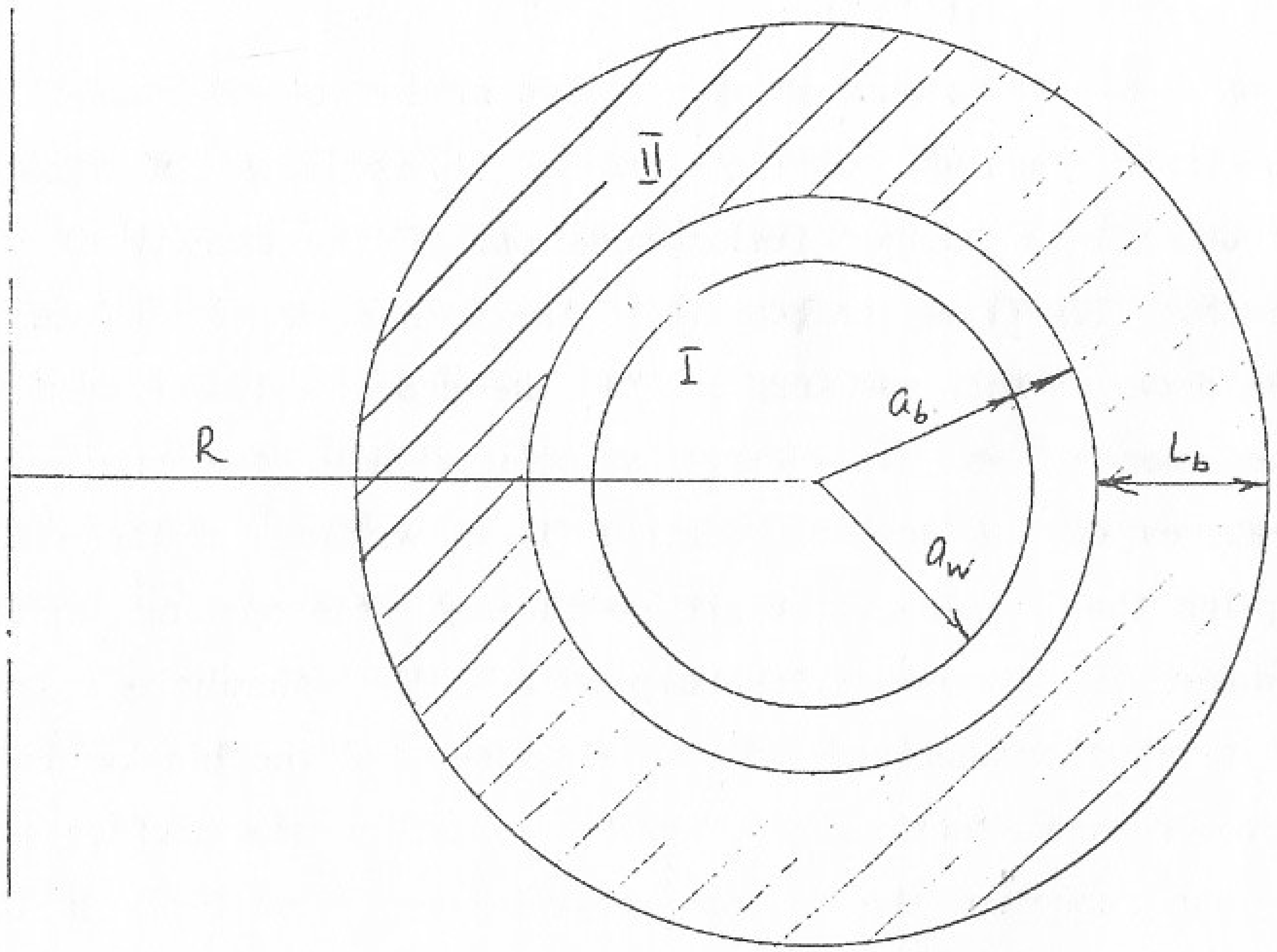

Fig. 1. Section of the chamber through the Φ = const plane, where is the toroidal angle. I and II denote the regions with the plasma and the blanket, respectively.

and the blanket, the flux to it will be higher than $K_*\Gamma_0$ owing to neutrons moving towards the plasma. For simplicity, we shall assume that the gap between the first wall and the blanket inner surface is small in comparison with the latter's radius, i.e. $a_b - a_w \ll a_b$. In this case, we can neglect the gradient of the neutron flux over the sample dimensions, and the total density of the neutron flux to the sample will be

$$\Gamma_\Sigma = \Gamma + \Gamma_1, \tag{3}$$

where Γ_1 is the density of the additional neutron flux from the blanket and the total amplification factor of neutron flux (K) is

$$K = \frac{\Gamma_\Sigma}{\Gamma_0}. \tag{4}$$

From (3) and (4) it follows that

$$K = K_*\left(1 + \frac{\Gamma_1}{\Gamma}\right) = K_* K_1, \tag{5}$$

where $K_1 \equiv (1 + \Gamma_*/\Gamma)$ is the factor of additional increase of flux due to neutrons moving towards the plasma. It is easy to see that $K_1 \leq 2$, while $K_1 = 2$ for $K_* = \infty$, $S_0 = 0$. In fact, $K_* = \infty$ for $\beta_n = 1$, $S_0 = 0$, therefore for $K_* = \infty$ the neutron flux to the plasma is the highest and equals the flux from the plasma; consequently $K_1(K_*=\infty) = K_{1max} = 2$. In order to estimate how high K should be to ensure that K_1 is close to K_{1max}, we assume that $K_* = n$, where n is integer. Then, if we assume that this value is attained owing to an n-fold escape of neutrons from the plasma, we come to the conclusion that neutrons should return to the plasma $n - 1$ times. Therefore, $\Gamma_1/\Gamma = (n-1)/n$. If we require that $K = 2$ with not more than 10% error, we obtain the condition $n > 5$. We shall see below that this condition can be easy satisfied.

Taking into account the foregoing, we shall characterize the increase of the flux to a sample located between the first wall (confining the plasma) and the blanket by a flux amplification factor

$$K = \frac{2\alpha}{1-\beta_n(1-S_0/S)} . \qquad (6)$$

Here we have introduced the factor $\alpha > 1$ to take into account the neutron multiplication reactions in the blanket (it is assumed that the magnitude of the albedo β_n is a slowly varying function of neutron energy); in the absence of multiplication $\alpha = 1$. In order to estimate the β_n in expression (6), we can use the well-known relationship (see, for example, Ref. [6]):

$$\beta_n = \frac{1+2D \left.\dfrac{d \ln \Phi(r)}{dr}\right|_{a_b}}{1-2D \left.\dfrac{d \ln \Phi(r)}{dr}\right|_{a_b}} , \qquad (7)$$

where D is the neutron diffusion coefficient in the blanket material (having the dimension of length) and $\Phi = n\,|\vec{v}|$, where n and $\vec{v}$ are, respectively, the density and velocity of neutrons in the blanket. $\Phi(r)$ in the simplest approximation

[6] satisfies the diffusion equation

$$\frac{D}{r}\frac{\partial}{\partial r} r \frac{\partial \Phi}{\partial r} - \Sigma_a \Phi = 0, \qquad (8)$$

where Σ_a is the macroscopic neutron absorption cross-section. Considering that $\Sigma_a D \ll 1$, with the help of (7) and (8) we find

$$\beta_n \approx 1 - 4\kappa D \frac{K_1(\kappa a_b)}{K_0(\kappa a_b)} , \qquad (9)$$

where $\kappa^2=\Sigma_a/D$, and $K_0(x)$ and $K_1(x)$ are modified Bessel functions. Hence, substituting β_n into Eq. (6), for $S_0/S = 0$ we obtain*)

$$K = \frac{\alpha K_0(\kappa a_b)}{2\kappa D K_1(\kappa a_b)} . \qquad (10)$$

In Fig. 2 the solid lines represent values for the flux amplification factor versus the inner radius of the blanket (a_b) for $\alpha = 1$ which were obtained on the basis of Eq. (10) and the neutron cross-section data [7] for $E_n = 14$ MeV (for neutron energies in the 3 - 14 MeV range the scattering and absorption cross-sections change little). It is seen that even at $a_b \sim 40$ cm amplification factor $K \sim 10$-30 is possible, and at $a_b > 100$ cm the amplification factor can attain 60-70 (when D_2O is used). Neutron multiplication as a result of

*) Formulae (9) and (10) are valid for a very thick blanket ($\kappa L_b \gg 1$). For smaller L_b the values of β_n and K are, of course, lower. For example, for $\kappa L_b=1$ the value of K is ~10% lower than that obtained by formula (10).

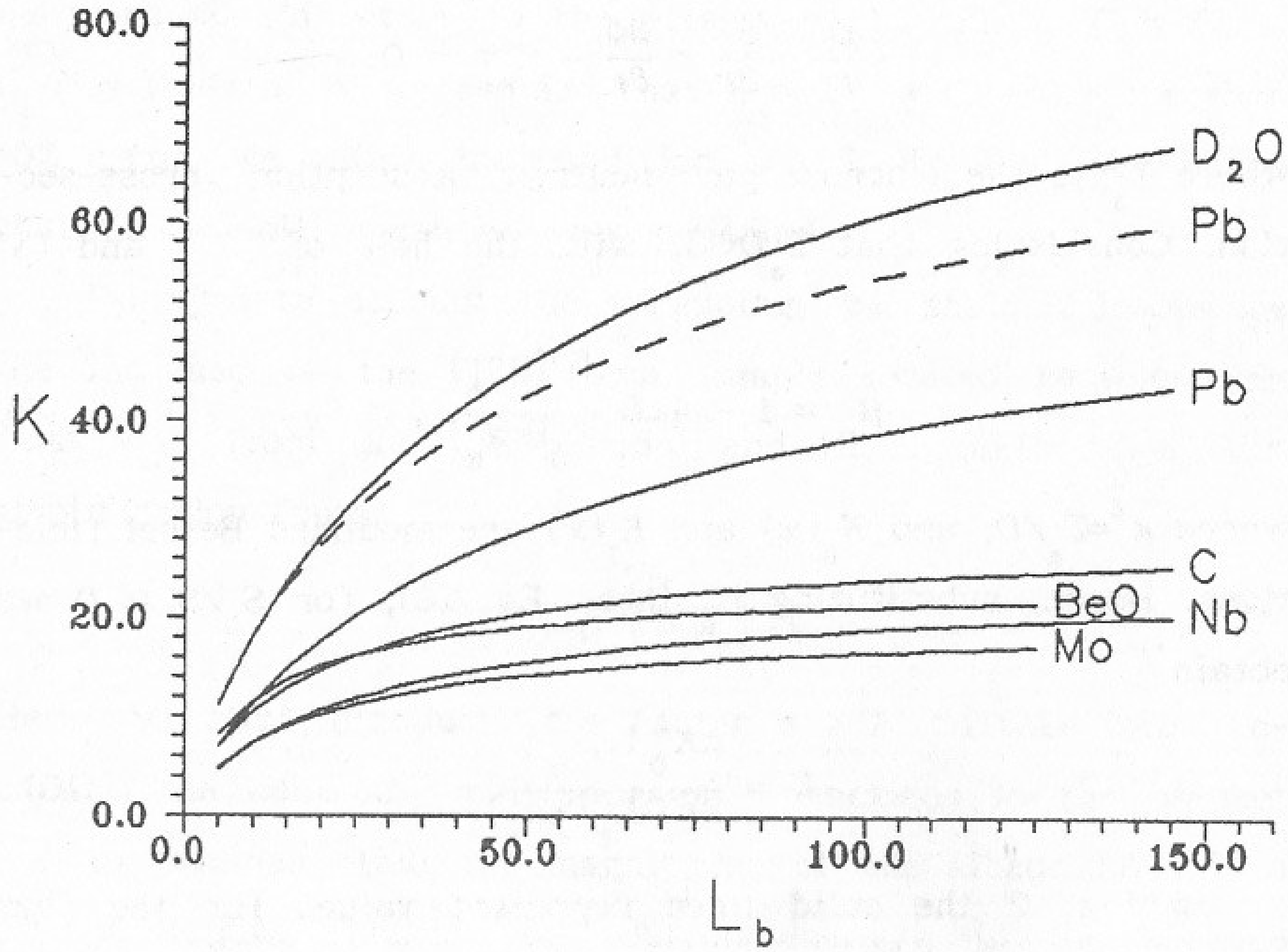

Fig. 2. Flux amplification factor $K = 2/(1-\beta_n)$ as a function of inner radius of blanket for several materials (solid lines). The following values of $\kappa - 1$ (cm) and D (cm), respectively, were assumed in the calculation: Mo (54, 1.52), Nb (76, 1.5), C (54, 0.903), BeO (29, 0.76), Pb (170, 1.32), D_2O (160, 0.8). The dashed line indicates the value of $(2\alpha/(1-\beta_n)$ for lead taking into account the n–$2n$ reaction; here we used values of $\alpha = 2$ and $\kappa^{-1} = 117$ cm, $D = 1.43$ cm corresponding to a secondary neutron energy $E_n = 7$ MeV.

reactions of the type $n \to 2n$, $n \to 3n$ in some materials leads to a still higher increase of K (α>1), although this is accompanied by a further decrease in neutron energy. In Fig.2, the dashed line shows the value of K for lead, in which the mean path length for the $n \to 2n$ reaction is only 5 cm, i. e. α=2; however, the albedo has to be determined for secondary neutrons with E_n = 7 MeV.

The proposed method of increasing thermonuclear neutron flux is a versatile one - it can be used in thermonuclear devices of different types (tokamak, stellarator, open traps, etc.). Note that since the thermal load on the first wall is associated with the plasma and alpha particle fluxes, it is determined by the value of the initial flux Γ_0 and can be very moderate even for a high value of the neutron flux Γ. Its efficiency increases with the dimensions of the device up to values at which $K_0(\kappa a_b) \simeq K_1(\kappa a_b)$, corresponding to the condition $a_b > 2\kappa^{-1}$ (in order to evaluate the possibility of satisfying this condition, let us note that, for example, when carbon is used in the blanket κ^{-1} = = 54 cm, but when D_2O is used κ^{-1} = 160 cm). Given that the initial thermonuclear neutron flux increases with the size of the device, the highest neutron flux to the test sample will clearly be achieved in large-scale devices. For example, in an ITER-type tokamak reactor with a lead blanket K = 60 and $\Gamma \simeq 3\cdot 10^{15}$ n/cm$^2\cdot$s. However, even small devices can have sufficiently large Γ, several times larger than the Γ_0 of an ITER reactor without a reflecting blanket, and consequently be of interest for the radiation testing of materials.

3. NEUTRON GENERATOR BASED ON A TWO-COMPONENT TOKAMAK

In our opinion, the most attractive neutron source at the present stage of a development is a tokamak as it is the best-tested system. For this reason below we consider the neutron source based on the two-component tokamak. We show that even in a small tokamak with a moderate magnetic field and an injected power of only 5 MW, a neutron flux density in the sample about $3\cdot10^{14}$ n/cm$^2\cdot$s can be obtained, which is sufficient for a fluence level of ~10^{22} n/cm^2 in less than a year. In certain sense such a device may be regarded as the extremely small one.

The features of this device which enable such results to be achieved are as follows:

I- It uses a blanket with a high neutron albedo β_n, which gives a considerable increase of the neutron flux;

II- Not only neutrons with E_n = 14.1 MeV but also neutrons with a wide energy spectrum formed by the blanket are used for testing samples;

III- The test sample is placed between the first wall and the blanket;

IV- The 14.1 MeV neutrons are produced in the DT beam-plasma reactions, deuterium beam is being proposed to use.

Let us look at the possible characteristics of such a

neutron generator. In all the «practical» formulae used below the radius R is expressed in meters, the temperature T in kiloelectronvolts, the magnetic field B in teslas, the density n in 10^{20} m^{-3}, the current I in the plasma in mega-amperes, the power P in megawatts and the time in milliseconds.

Parameter	Calculated value	Explanation
(a) Geometry		
Major radius of plasma, R	0.9 m	
Radius of first wall, a_w	0.15 m	
Radius of blanket inner surface, a_b	0.2 m	
Blanket thickness, L_b	0.35 - 0.4 m	
Aspect ratio of coils, A	1.4 - 1.5	
Plasma chamber volume, V	0.4 m^3	
First wall surface area, S	5.3 m^2	
(b) Plasma part		
Plasma aspect ratio, A	6	
Magnetic field on axis, B	4.2 T	Maximum field on coil 12-15 T
Current in plasma, I	0.25 MA	
Safety factor, q	2.1	$q = 5\dfrac{RB}{A^2 I}$
Average temperature of electrons and ions, $\bar{T}$	2 keV	
Average density of ions, $\bar{n}$	$1.2\cdot10^{20}$m^{-3}	Murakami parameter Mu $= 10\dfrac{\bar{n}R}{B}$

Temperature at plasma center, $T(0)$	3 keV	Tangential injection provides main heating at center heating at center
Plasma density at center, $n(0)$	$2.4 \cdot 10^{20} m^{-3}$	Parabolic profile
Effective charge number, Z_{eff}	1	
Required energy confinement time, τ_E	11 ms	$\tau_{E1}^{DIII} < \tau_E \equiv \frac{3\bar{n}\bar{T}V_p}{P} < \tau_{E2}^{DIII}$, where $\tau_{E1}^{DIII} = I(25+58/P)$ $\tau_{E2}^{DIII} = I(58+61/P)$ P_i is the injected power in MW; $P=0.8P_i$ $P_i = 5$ MW;
Toroidal, $\bar{\beta}$	1.1%	Troyon limit $\beta_{max} = \frac{20}{qA} = 1.6\%$
Poloidal $\bar{\beta}_I$	1.75	$\beta_I/A = 0.29$

(c) Neutral injection (tangential)

Injected power, P_i	5 MW	
Density of injected power, $\mathcal{P}_i$	12.5 MW/m	
Energy of injected deuterons, ε_o	140 keV	
Deuteron current, I_o	35.7 A	
Deuteron gyroradius, ρ_L	1.8 cm	
Neutron beam ionization length, $\bar{\lambda}$	32 cm	
Fast ion slowing down time, $\tau_\varepsilon(\varepsilon_o)$	28 ms	$\tau_\varepsilon^{-1} = -\frac{1}{\varepsilon^*}\frac{d\varepsilon^*}{dt}$, where ε^* is the fast ion energy

Charge-exchange time, τ_{cx}	12 ms	$\tau_{cx} = \tau_n \frac{\langle\sigma v\rangle_e}{\sigma_{cx}(v^*)v^*}$, $\tau_n = \tau_E$, and v^* is the fast ion velocity
Transverse pressure of fast particles, $\beta_\perp^*$	0.4%	We assume that the fast particle confinement time in the plasma is twice the charge-exchange time $\tau^* = 2\tau_{cx}$, $\beta_\perp^*(\%) = \frac{\overline{\sin^2\vartheta^*}}{4}\frac{\mathcal{P}_i}{B^2}\tau^* f(\tau^*/\tau_\varepsilon)$ or $\tau^* = 24$ ms, $\tau = 28$ ms $(\tau^*/\tau) = 0.6$
Energy efficiency, Q	0.64	During time τ^* the ion loses 75% of its energy, escaping from plasma with energy < 40 keV. Thus, $Q > Q(140) - Q(40) = 0.64$ $Q(\varepsilon_o) = P_{fus}/P_i = (\varepsilon_{fus}/\varepsilon_o)f$; $P_{fus} = \varepsilon_{fus} I_o f$, $\varepsilon_{fus} = 17.6$ MeV, f - being the reaction propability
Beam area, S_b	500-1000 cm	$S_b = P_i/p_{max}$, where $p_{max} = 5 - 10$ kW/cm^2
Ratio of beam area to first wall surface area, S_b/S_w	0.009-0.019	

(d) Tritium system

Tritium burnup in one year of operation	141 g/year	
Tritium drawing through the system (not exceeding)	78 g/hour	for $\tau_n = \tau_E$

(e) First wall

Thermal wall load, W_{th}	1.06 MW/m	$W_{th}=P_i\left(1+Q\,\frac{\varepsilon_\alpha}{\varepsilon_{fus}}\right)/S_w$ $\varepsilon_\alpha = 3.5$ MeV
Neutron load, $\Gamma_0 E_n$	0.48 MW/m	
Density of thermonuclear neutron flux with $E_n = 14.1$ MeV to the first wall, Γ_0	$2.13\cdot10^{13}$ n/cm$^2\cdot$s	

(f) Blanket

Flux amplification factor, K	15	Lead (35 cm) + + absorber(5 cm)

The absorber can be lithium-containing material which will ensure both neutron shielding of the coils and partial tritium breeding. The blanket should contain channels through which the test sample can be inserted into the slit for testing. The thermal load in the blanket is small - 0.55 W/cm^3.

The total neutron flux to the sample is about $3\cdot10^{14}$ n/cm$^2\cdot$s.

4. CONCLUSION

In this paper we have described a method of applying to the test sample of a material the maximum flux of neutrons formed in a plasma system by nuclear fusion reactions (for

example, DT reactions). The increase of neutron flux is achieved by accumulation of neutrons in the plasma confinement volume an their rational use. The neutron energy spectrum lies in the region of E_n < 14.1 MeV. The shape of the spectrum depends on the material of the blanket, its structure and geometrical factors.

Our consideration showed that the flux of thermonuclear neutrons could be increased considerably - by a factor of several tens. But this consideration is qualitative rather than quantitative. Detailed calculations of specific systems need to be performed in order to obtain a complete picture of the characteristics of the neutron source considered.

In our opinion, it is of greatest interest at the present stage to calculate and develop the design of the neutron source based on a tokamak. Its possible parameters are presented in Sec. 3. However, the neutron source, proposed in Sec. 3 can not operate in steady-state regime because of the large energy release in the toroidal magnetic field coils at the inner circumference of the torus. To avoid the mentioned difficulty and, in addition, to provide coils with the neutron absorber protecting them against the neutron flux one need to increase the tokamak major radius (and aspect ratio of the torus). Then the plasma cross-section should be taken elongated and, perhaps, bean-shaped. For example, one can take $A = 12$, $a = 15$ cm, $k = 2$ (k is a plasma elongation).Also the toroidal field maybe decreased ($B \sim 3 \div 3.5$ T).

At last, we note that the mirrors, especially Cas-Dynamic Trap [2], seems to be good candidates for the use as a neutron source with high albedo blanket because they have technological advantages in comparison with toroidal systems.

The authors are grateful to D.D. Ryutov for fruitful discussion.

REFERENCES

1. *S.Ishino, P.Schiller, A.Rowcliffe.* Journal of Fusion Energy, v.8 (1989) N. 3/4, p.147.
2. *V.V. Mirnov, V.P.Nagornyj, D.D. Ryutov.* A gas-dynamic trap with a two component plasma. Preprint 84-40, Institute of Nuclear Physics, Novosibirsk (1984);
   I.A. Kotel'nikov, V.V. Mirnov, V.P. Nagornyj, D.D. Ryutov. In: Plasma Physics and Controlled Nuclear Fusion Research (Proc. Conf. London, 1984), v. 2, IAEA, Vienna (1985) p. 309.
3. *H.L. Berk, D.D. Ryutov.* Comments Plasma Phys. Controlled Fusion, v. 13 (1990) N.4, p. 173.
4. *V.V. Vasil'ev, V.Eh. Lukash, A.A. Bekhtenev et al.* Vopr. Nauki i Tekh. Ser. Termoyad. Sintez 2 (1985) p. 41.
5. *H.W. Hendel, D.L. Jassby.* The Tokamak as a Neutron Source. Preprint PPPL-2656 (1989).
6. *S. Glasstone, M.C. Edlund.* The Elements of Nuclear Reactor Theory. D. Van Nostrand Co. Inc., 1952.
7. *D.I. Garber, R.R. Kinsey.* Neutron Cross Sections, BNL-325, Third Edition, v. II, 1976.

Ya.I. Kolesnichenko, V.P. Nagornyj

A Neutron Generator Based on a Thermonuclear Device with a High Albedo Blanket

Я.И. Колесниченко, В.П. Нагорный

Генератор нейтронов на основе термоядерной установки с высокоальбедным бланкетом

Ответственный за выпуск: С.Г.Попов

Работа поступила - 18 марта 1991 г.
Подписано к печати 21.03 1991 г.
Формат бумаги 60×90 1/16
Объем 1,3 п.л., 1,0 уч.-изд.л.
Тираж 180 экз. Бесплатно. Заказ 23.

Ротапринт ИЯФ СО АН СССР,
г. Новосибирск, 91.