NEUTRONIC MODEL OF A STELLARATOR-MIRROR FUSION-FISSION HYBRID

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The MCNPX numerical code has been used to model a compact concept for a fusion-fission reactor based on a combined stellarator-mirror trap. Calculation results for the radial leakage of neutrons through the mantle surface of the fission reactor are presented.

PACS: 52.55.Hc

INTRODUCTION

Utilization of spent nuclear fuel is a current global problem. Up to now, this problem is not solved in a sustainable way. Most of the spent nuclear fuel from PWR is stored in geological nuclear waste repositories. Because of the slow decrease of radioactivity, the repository term is incredibly long, about 300000 years. Another option is separation of transuranic elements and burning them in fast reactors. The waste without transuranic content becomes non-radioactive much faster. Transuranic elements included into the fuel cause deficit of delayed neutrons, that decrease the reactor controllability. Thus, an attractive idea is the development of a subcritical reactor, the main purpose of which will be a more safe burning of transuranic elements from the spent nuclear fuel.

STELLARATOR-MIRROR HYBRID

In Ref. [1] a stellarator-mirror hybrid reactor is proposed. It consists of a magnetic trap for plasma confinement in which fusion neutrons are generated and a sub-critical fast reactor driven by these neutrons. The magnetic trap is of a combined type: it is a toroidal stellarator with an embedded magnetic mirror with lower magnetic field. The stellarator part is for confinement of warm dense deuterium target plasma. Hot tritium sloshing tritium ions are confined at the mirror part of the device. At this part the plasma column is straight. It is surrounded by a cylindrically symmetrical fission mantle.

The hot minority tritium ions are sustained in the plasma by neutral beam injection (NBI). The NBI is normal to the magnetic field and targets plasma just near the fission mantle border. The sloshing ions bounce inside the magnetic mirror between the injection point and midplane of the mirror. Some fusion neutrons are generated outside the reactor core near the injection point. Besides, the fission neutron flux has a high value near the ends of the reactor central through-hole. There is a need of protection from these neutrons.

The purpose of this paper is to calculate the leakage of neutrons through the mantle surface of the fission reactor.

CALCULATION MODEL

The model developed has a cylindrical symmetry with a horizontal axis. Its radial and axial structure is shown in Fig. 1. The vacuum chamber contains the D-T plasmas which supplies the fusion neutrons. The inner radius of the vacuum chamber is 0.5 m. The first wall thickness is 3 cm.

The reactor core thickness was determined from the results of critically calculations. A thickness of 27.8 cm was chosen to make the effective multiplication factor $k_{eff} \approx 0.95$. The length of the core is 3 m. It has axial reflectors on both sides. The thickness of each axial reflector is 57.8 cm and length is 0.5 m. The fission blanket (core) is surrounded by the buffer (LBE). The thickness of the buffer is 15 cm. The thickness of the radial reflector is 30 cm and the shield thickness is 25 cm. The radial reflector was made from homogeneous mixture of HT-9 steel and Li17Pb83 (20% enriched Li-6) with the volume fractions 70% and 30%, respectively. This mixture is used for tritium breeding. The shield contains a 60:40 vol.% mixture of the stainless steel alloy S30467 type 304B7 [2,3] with water. The steel contains 1.75 wt.% of natural boron. All the materials, as well as their temperatures, which are included in the design were taken from Ref.[4].

The fission blanket was represented in the model as a homogenized mixture of fuel, structure/cladding and coolant. HT-9 and LBE were used as structure/cladding and coolant materials, respectively.

The actual fuel material is the zirconium alloy (TRU-10Zr) which consists of the transuranic elements with 10 wt.% of zirconium. The alloy has a mass density of 18.37 g/cm^3. The isotopic composition of the TRU in wt.% are: U-235 – 0.0039, U-236 – 0.0018, U - 238 – 0.4234, Np-237 – 4.313, Pu-239 – 53.901, Pu-240 – 21.231, Pu-241 – 3.870, Pu-242 – 4.677, Am-241 – 9.184, Am-242m – 0.0067, Am-243 – 1.021, Cm-243 – 0.0018, Cm-244 – 0.1158, Cm-245 – 0.0125, Cm-246 – 0.0010.

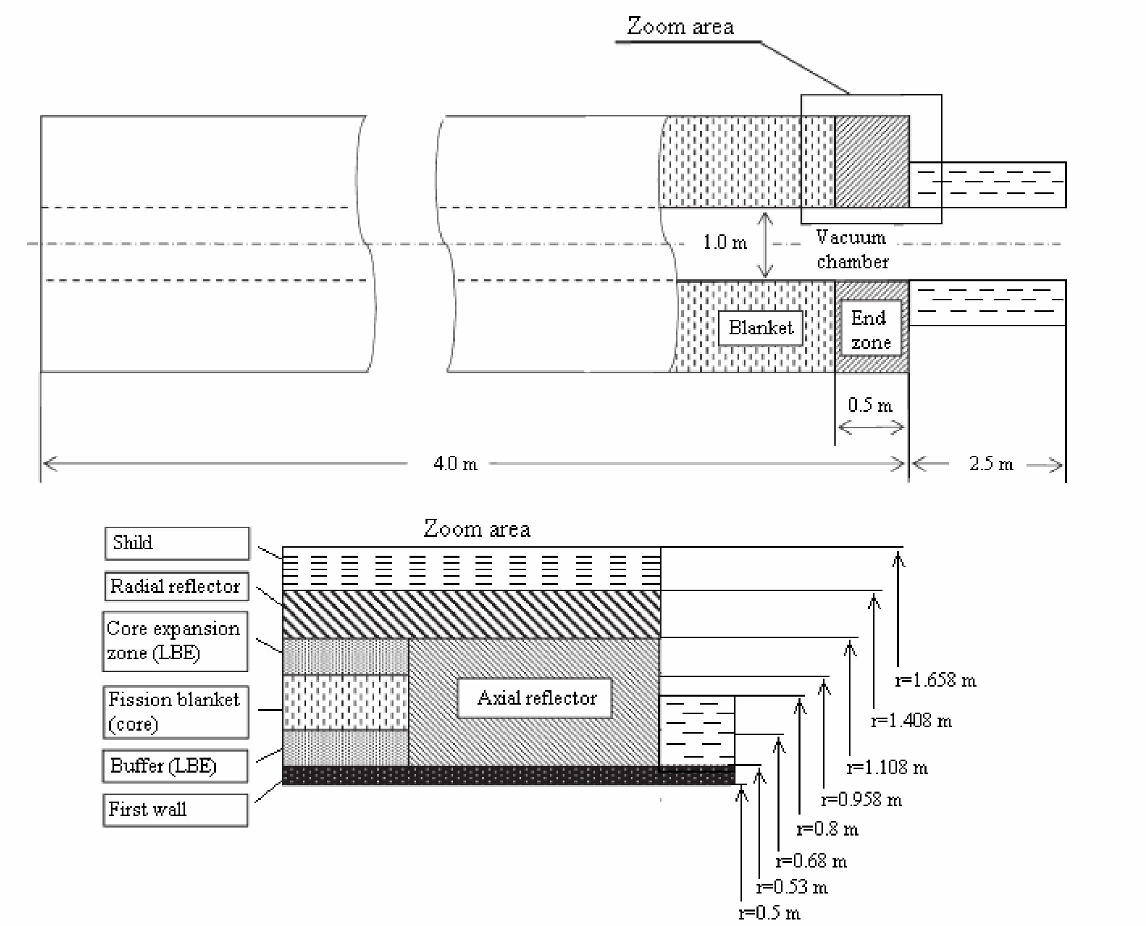

Fig. 1. Radial and axial structures of the mirror based fusion-fission hybrid model

This isotopic composition corresponds to the composition of the spent nuclear fuel from PWR after removal from it of uranium-238. The following volume fraction was used for the homogenized fission blanket: fuel slug material – 0.14, structure/cladding – 0.103, coolant – 0.695. The LBE was assumed to be a mixture of 44.5 wt.% lead and 55.5 wt.% bismuth. The material which has been used for the axial reflectors: a homogeneous mixture of HT-9 steel and LBE-coolant with the volume fractions 70 and 30%, respectively.

The total length of the main part of the model is 4 m.

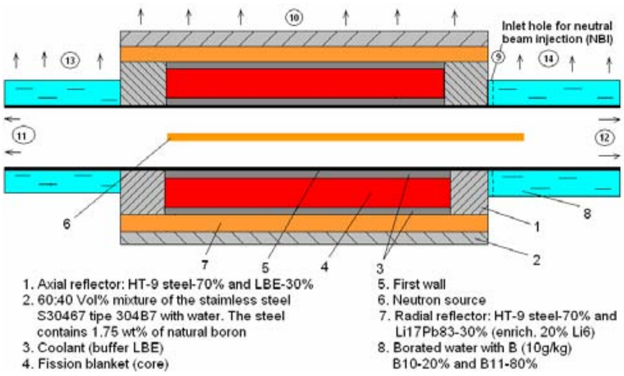

Fig. 2. Scheme of the reactor part of the fusion-fission hybrid

Since the neutron generation zone extends slightly beyond the fission reactor core, as shown in Fig. 2, this part of the plasma column is surrounded by a vessel filled with borated water to absorb the outcoming neutrons, because the water slows the neutrons and boron absorbs slow neutrons.

The concentration of boron in the water was taken 10 g/kg. The isotopical content is B_{10} – 20% and B_{11} – 80%. The part with borated water has a length of 2.5 m at both sides of the main part and a thickness is of 27 cm. At the right side of the reactor, openings are made to provide access to the plasma for the neutral beam (see Fig. 2, inlet hole for neutral beam injection).

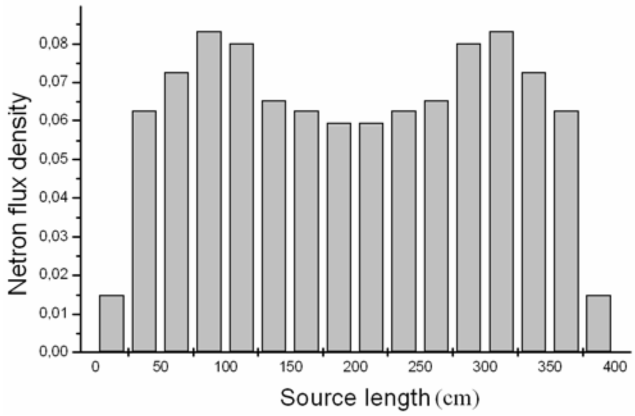

Fig. 3. Intensity distribution of the source neutrons for Einj=300 keV

In the calculation model, a D-T fusion neutron source was used. In the calculation model, the emission density was distributed within a number of cylindrical volumes of radius 10 cm and with a length of 4 m. At every source point, the fusion neutrons were emitted with a fixed kinetic energy of 14.1 MeV and isotropic velocity distribution. The intensity distribution along the length of the neutron source [5], which was used in the MCNPX model, is shown in Fig.3. The total number of the particles emitted by the source is normalized to unity.

RESULTS OF CALCULATIONS

The MCNPX numerical code has been used to model the neutron transport of a mirror based fusion-fission reactor. The calculated effective multiplication factor is k_{eff} = 0.95087±0.00017. A core volume of 4.3 m^3 containing 2.3 tones of the fuel (TRU-isotopes [3]).

For this calculation, average fission energy (MeV) deposited in the core per fusion source neutron: is 1136±1% MeV, which corresponds to an energy multiplication of 65.

Calculation results for the leakage of neutrons through the mantle surface of the fission reactor are presented below. Neutron leakage, normalized per fusion neutron, through the separate surfaces (see Fig. 2) is

- Surface 9: 3.61×10^{-3} ±1%
- Surface 10: 1.7×10^{-3} ±1%
- Surface 11: 0.0881± 0.3%
- Surface 12: 0.0985 ± 0.3%
- Surface 13: 0.116 ±1%
- Surface 14: 0.286 ± 0.3%

For comparison, calculation results for leakage of neutrons without borated water shielding are presented:

- Surface 9: 4.77×10^{-3} ±1%
- Surface 13: 1.3 ± 0.4%
- Surface 14: 3.13 ± 0.2%

As seen from the above results, radial leakage of neutrons with the borated water shielding on both sides of the reactor is an order of magnitude lower.

The estimates predict that the power released with neutrons from the reactor to outer space is 9.4×10^{-16} Ws per fusion source neutron. Thus, the total power delivered into the magnetic coils which surround the reactor mantle will not exceed the value of 5.7 kW for a neutron source with intensity 6×10^{18} neutrons per second. This power loading should be considered in the calculation of the cooling requirements for the magnetic coils.

In the considered scheme with NBI some fusion neutrons are generated under the axial reflector and even outside the reactor. Such a wasting of neutrons decreases the hybrid reactor efficiency. To estimate the loss of the efficiency, the calculation is made with a better neutron source: it is of the same shape, but 1 m shorter and placed symmetrically with respect to the midplane. In this case, the average fission energy deposited in the core per fusion source neutron is higher by 23%. Thus, the power losses in NBI case could be estimated by this number.

CONCLUSIONS

By means of neutron transport calculations a principal design for fission blanket of a mirror fusion–fission hybrid has been devised. The calculations were carried out with the Monte Carlo code MCNPX.

Neutron outflux of the simulated reactor was calculated.

Since part of the fusion neutrons are generated outside of the fission reactor core, this part of the plasma column was surrounded by a vessel filled with borated water to absorb the outcoming neutrons.

The neutron fluxes that are emitted into the stellarator vacuum chamber at both sides of the reactor have also been calculated.

ACKNOWLEDGEMENT

This work is supported in part by a grant from Swedish Institute.

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Article received 30.10.12

НЕЙТРОННАЯ МОДЕЛЬ ГИБРИДНОГО РЕАКТОРА НА ОСНОВЕ СТЕЛЛАРАТОРА И ОТКРЫТОЙ ЛОВУШКИ

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С помощью программы MCNPX разработана концепция контролированного гибридного реактора не больших размеров на основе открытой ловушки. В работе представлены результаты радиальной утечки нейтронных потоков за пределы моделируемой системы.

НЕЙТРОНА МОДЕЛЬ ГІБРИДНОГО РЕАКТОРА НА ОСНОВІ СТЕЛАРАТОРА ТА ВІДКРИТОЇ ПАСТКИ

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За допомогою програми MCNPX розроблена концепція контрольованого гібридного реактора не великих розмірів на основі відкритої пастки. В роботі представлено результати розрахунків виходу нейтронних потоків за межі модельованої системи.