NEUTRONIC MODEL OF A MIRROR BASED FUSION-FISSION HYBRID FOR THE INCINERATION OF SPENT NUCLEAR FUEL AND WITH POTENTIAL FOR POWER GENERATION

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In the last decade the Georgia Institute of Technology (Georgia Tech) published several design concepts of tokamak based fusion-fission hybrids which use solid fuel consisting of the transuranic elements of spent nuclear fuel from Light-Water-Reactors. The objectives of the hybrids are the incineration of the transuranic elements and additional net energy production.

The paper presents a rough scientific design of the blanket of a mirror hybrid which was derived from the results of neutron transport calculations. The main operation parameters of two hybrid options were specified. One is the analog to Georgia Techs first version of a "fusion transmutation of waste reactor" (FTWR) and the other is a possible near-term option which requires minimal fusion power.

I. INTRODUCTION

Already in the eighties various concepts of fusion-fission hybrids have been intensively studied by the fusion community. Following proposals of Bethe[1], these studies aimed at two objectives:

- To gain energy more efficiently than by means of a fusion reactor alone.
- To breed additionally new fissile material either U-233 from Th-232 or Pu-239 from U-238.

A revival of such research activities started in the middle of the nineties after Bowman et al.[2] proposed to use so-called accelerator driven subcritical systems (ADS) for the incineration of nuclear waste from fission reactors. Since then, newly developed concepts of fusion-fission hybrids include the objective of reducing nuclear waste from fission reactors too.

Presently two lines of design concepts which differ in the type of fuel are pursued. One development line is based on the use of the molten salt Flibe as both coolant and fuel carrier[3-5]. The other line uses solid fuel together with liquid metal or gas cooling[6-8]. Especially this direction was advanced at Georgia Tech by several tokamak based concepts. The first version, denoted as "FTWR", which uses lithium-lead eutectic as coolant and for tritium breeding, is designed for a nominal criticality $k_{eff} \leq 0.95$ and a fission power of 3 GW$_{th}$. It fulfills self-sufficiency in both tritium and electricity. An improved version "FTWR-SC" with superconducting magnets[7] would achieve a total energy efficiency of Q_e=4.9.

Mirror machines can have steady state operation and the open geometry is essentially simpler than for a torroidal device. Therefore, mirror based hybrids promise substantial cost savings. On the other hand, their plasma confinement is worse. Hence, it is of interest to clarify whether a mirror could serve as neutron source for a hybrid and how it would compare to tokamak based hybrids. Early studies in this direction were made by Taczanowski et al.[9]. Recently, Noack et al.[10] studied a possible use of the GDT based fusion neutron source as driver of a minor actinide burner and Moiseenko et al.[11] considered a stellarator-mirror based hybrid.

On the base of results of neutron transport calculations carried out with the MCNP5 Monte Carlo code[12] and with cross-sections from the JEFF-3.1 data library[13] a rough axisymmetric blanket model was devised. It follows the proposal of a hybrid based on a straight field line mirror[14]. It is foreseen to use superconducting magnets[15]. Up to now, their shielding was not yet considered. This paper describes the model in the present state, points to important features, explains several reactor safety aspects and presents the most important operation parameters. Though the full set of transport calculations was made only for the design with the same nominal criticality of the FTWR, a second option with increased criticality $k_{eff} \leq 0.97$ and reduced fission power of 1.5 GW$_{th}$ was identified by means of appropriate scaling. Since it requires substantially less fusion power from the mirror, it could be considered as a near-term option.

II. MODEL OF THE BLANKET

II.A. Geometry, Materials and Neutron Source

The radial structure of the model is shown in Fig. 1. The sizes, materials and temperatures, which were assigned to the materials in accordance with the cross-section sets available in Jeff-3.1, are as follows.

- Vacuum chamber: Radius=90 cm, vacuum.

- First wall: Outer radius R_o=93 cm, HT-9 steel, temperature T=600°K.
- Buffer: R_o=108 cm, LBE (44.5Pb/55.5Bi-eutectic), T= 700°K.
- Core: R_o=129.3 cm for k_{eff}=0.95 (R_o=130.3 cm for k_{eff}=0.97), homogenized mixture. Fuel: (TRU-10Zr)-alloy dispersed in Zr-matrix in a mass ratio as specified for the ADS-project HYPER[16], fertile-free composition of transuranic isotopes (TRU) as given in Ref. 8, T=900°K. Structure/cladding: HT-9, T=800°K. Coolant: LBE, T=700°K.
- Core expansion zone: R_o=144.3 cm, LBE, T=700°K.
- Radial reflector: R_o=194.3 cm, homogenized mixture of HT-9 (70 vol%) and Li17Pb83 (30 vol%), T=700°K, 20% enrichment of Li-6.
- Axial reflector: Inner radius R_i=93 cm, R_o=144.3, homogenized mixture of HT-9 (70 vol%) and LBE (30 vol%), T=700°K.

The core, buffer and expansion zone are 25 m long. At their front faces these zones are closed by axial reflectors with a thickness of 0.5 m. All four zones belong to the same LBE-coolant circuit. The radial reflector is assumed to be cooled by Li17Pb83-eutectic in a separate loop. The first wall and the radial reflector are 26 m long.

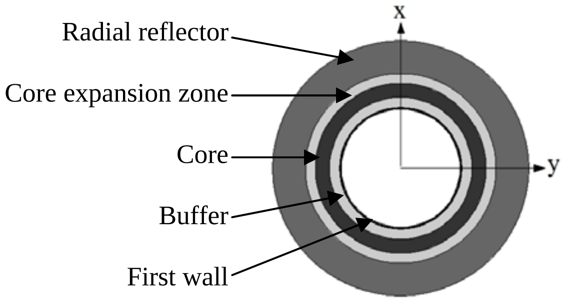

Fig. 1. Radial structure of the blanket model.

The fundamental feature of the design is that the core confines a high-energy (fast) neutron spectrum which is well suited for the incineration of TRU-isotopes. The function of the buffer is to shield the first wall against the high-energy fission neutrons produced in the core and to reflect the core at its inside. The expansion zone reflects the core at its outside and gives variability for larger core loadings or for the installation of special assemblies which should be irradiated by high-energy neutrons.

The external fusion neutron source is modeled as a cylindrical volume with a diameter of 20 cm and a length of 25 m. The neutrons are emitted with the kinetic energy of 14.1 MeV in isotropically selected flight directions. Radial uniformity is assumed and the axial variation of the emission density is shown in Fig. 2. It is representative for a mirror with sloshing ions density peaks. They can be produced by ion cyclotron resonance heating[19].

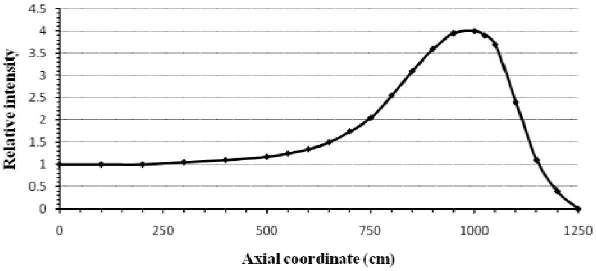

Fig. 2. Axial dependency of the fusion neutron emission density.

II. IMPORTANT CALCULATION RESULTS

II.A. Criticality Calculations and Safety Aspects

The thickness of the core D_c was iteratively determined by means of standard criticality calculations for the fresh loading of the core. The results are:

- D_c=21.3 cm for the criticality k_{eff}=0.95,
- D_c=22.3 cm for the criticality k_{eff}=0.97.

The first version is the analog mirror hybrid of the FTWR and the second can be considered as a near-term option of a mirror hybrid (see section III.).

From liquid-metal cooled fast reactors it is well known that accidents in course of which coolant is lost or boiling of coolant occurs in the core can result in a considerable increase of the criticality. Therefore, two series of criticality calculations were carried out in which certain representative configurations were modeled with appropriate geometry. Concerning the loss-of-coolant-accidents several calculations were carried out with different LBE-levels. In no case an increase Δk_{eff} of the criticality was observed. Compared to the fast reactors, this finding is a remarkable positive specific feature. In the other series several rings without the coolant were modeled within the core extending over the whole length. The results are shown in Fig. 3. One can see that the criticality increases if central radial regions are voided. However, even the maximal effect is so limited that the blanket in any case remains in deep sub-criticality. The

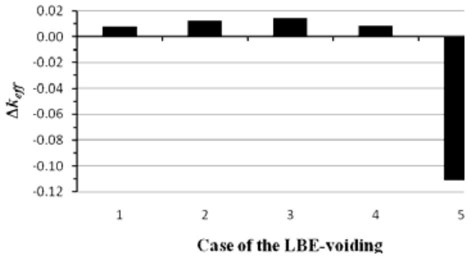

Fig. 3. Criticality effects of different cases of partial coolant voiding. Voided radial regions (cm): 1 – [115,122], 2 – [113,124], 3 – [111,126], 4 – entire core, 5 – buffer, core and expansion zone.

thin radial extension of the core and the LBE-coolant which practically does not absorb neutrons are the reasons for this positive safety aspect. Though the calculations were made for the analog hybrid model with k_{eff}=0.95 it seems to be justified to conclude that the same fact will be valid in the case with nominal criticality k_{eff}=0.97 too.

II.B. Calculations with Fusion Neutron Source

The calculation results of important integral parameters for fresh fuel loading of the analog mirror hybrid are given in TABLE I.

TABLE I. Calculation results of integral parameters (all quantities are normalized “per fusion source neutron”)

Parameter	Definition	Result
M_S	Average number of fission neutrons created	23.6
N_{fis}	Average number of fission reactions	8.0
$N_{n,2n}$	Average number of n,2n-reactions	0.69
h_{fis}	Average fission energy (MeV) deposited in the core	1513
TBR	Tritium breeding ratio – average number of T-nuclei produced	2.95

Further secondary parameters can be derived:

- The neutron multiplicity of a fission neutron

$$M_{eff}=k_{eff}/(1-k_{eff}). \tag{1}$$

- The so-called source importance

$$\varphi^{*}=M_S/M_{eff}\,. \tag{2}$$

  It amounts to 1.24 for the present hybrid model.

- The power amplification factor of a fusion-fission hybrid which is the ratio between fission power generated and fusion power delivered by the driver

$$PAF_{fis}=P_{fis}/P_{fus}=h_{fis}/E_{fus} \tag{3}$$

  where E_{fus}=17.6 MeV.

Moreover, the power amplification factor of a hybrid scales approximately with the criticality of the blanket in proportion with M_{eff}[19] hence, the relationship

$$PAF_{fis}=4.53\cdot M_{eff} \tag{4}$$

is valid and gives for both hybrid options: PAF_{fis}=86.0 and 146.4, respectively.

The local dependencies of several integrals of the neutron field within the core were studied. To this end, the core volume of the analog hybrid was divided in seven cylindrical rings of equal thickness and they were divided in axial direction in cells of 50 cm height. The volume averaged quantities were calculated for this grid of cells. As an example Fig. 4 shows the axial dependencies of the fission heating density (per fusion neutron) along the inner and outer core rings. From this diagram, two facts should be pointed out. The axial profile of the source intensity (see Fig. 2.) does not appear but a long plateau instead. Moreover, the drop of the heating density in outward direction is about 20% only. Both aspects are positive regarding the incineration of the fuel.

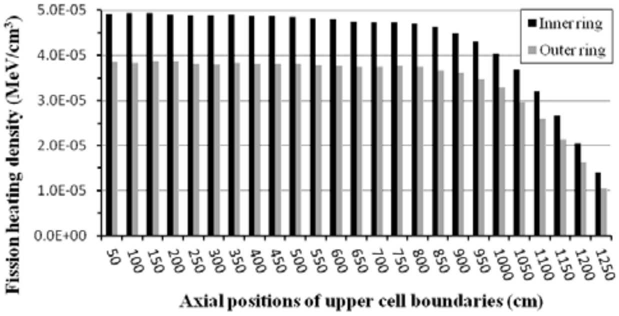

Fig. 4. Axial dependencies of the fission heating.

The calculated cell averaged flux spectra showed that the neutron spectrum varies very weakly and this only close to the core edges. Because of this, the ratios between the incineration rates of the TRU-isotopes are practically constant over the whole core volume.

III. RESULTS OF EXTRAPOLATIONS TO TWO POWER HYBRID OPTIONS

During operation of the designed hybrids at high power the fuel burns itself down. This burn-up results in the decrease of the amount of fissile isotopes and in the increase of neutron capturing fission products. Thus, the change of its isotopic composition causes reductions of the criticality k_{eff} and consequently of the power amplification factor in accordance with Eq. (4). Likewise as for fission reactors, the operation of a hybrid is divided in cycles of certain duration. For each cycle the core is loaded with fresh fuel so that the operation starts again with the nominal criticality. In addition, in order to keep the power output during the cycle constant the burn-up of the fuel during the cycle must be compensated. The FTWR and the present mirror hybrid suppose that this is done by increasing the source intensity. In order to hold the fusion power requirement for the mirror minimal, the cycle length was fixed to 311 effective full power days (EFPDs) which is just half of the FTWR-cycle. The criticality drop-offs between the beginning and the end of cycle (BOC and EOC, respectively) for both hybrid options were interpolated from burn-up data given for the FTWR[6]. The results are given in TABLE II. With the values of the criticalities at BOC and EOC determined in this way, one gets the fusion powers required at these moments by means of Eqs. (3) and (4). With an assumed value of 0.15 for the plasma-*Q* $Q_p=P_{fus}/P_{heat}$ the maximal heating powers that have to be launched into the plasma are P_{heat}=494, and 130 MW for the analog and for the near-term option, respectively. The latter value should be achievable by ion cyclotron resonance heating.

The critical issues for the first wall of a fusion device are the heat load from the plasma and the material damage caused by the high-energy neutrons. Both effects work in different areas – the neutron damage in the range of the confinement region where the neutron source and the blanket are located and the heat load at the plasma dump plates in both end tanks. Estimates show that dump plates of radius between 4-5 meters or with W-shaped surfaces would solve the heat load problem for the near-term hybrid. The neutron damage of the first wall is caused both by the fusion source neutrons and by the fission neutrons produced in the core. The latter contribution turned out substantially bigger. However, the fusion neutrons cause the slight maximum of the damage rate in the axial range between $750\leq z/\mathrm{cm}\leq 900$. By taking into account that the fission power remains constant during the cycle and that the fusion power increases $\sim 1/M_{eff}$ in accordance with Eqs. (1) and (3), one obtains the axial profile of the DPA accumulated per cycle which is shown in Fig. 5. The lifetimes given in TABLE II were calculated with the maximal DPA-values and with the assumption that the HT-9 steel can withstand 150-200 DPA as proposed for DEMO.

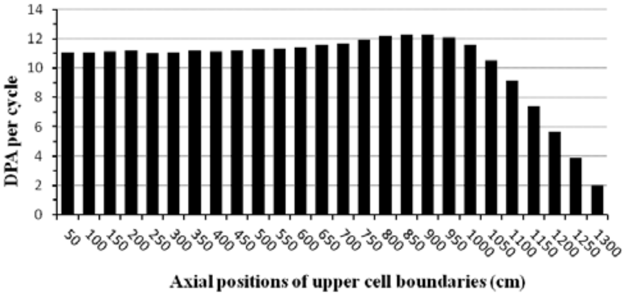

Fig. 5. DPA per cycle in the 1. wall of the analog hybrid.

Since the fusion power increases during the cycle, the tritium breeding ratio (TBR) was calculated as the ratio of cycle integrated tritium production to consumption. The T-production in the radial reflector is assumed to be proportional to the fission power which is constant and the consumption varies with the fusion power $\sim 1/M_{eff}$. The results obtained for both hybrids are given in TABLE II.

The "engineering *Q*" of a fusion-fission hybrid is defined as ratio of the gross electric power produced to the electric power inputted[6]. Likewise as for the TBR, the ratio of the produced to the consumed energies per cycle is the appropriate energy performance parameter for our cycle-wise operating hybrids. The Q_e-values given in TABLE II were approximately calculated using data quoted for the FTWR-SC. They reveal a potential of the mirror hybrids for power generation in addition to the incineration of spent nuclear fuel, even with the low Q_p=0.15.

TABLE II. Key-parameters of the mirror hybrids

Parameter	**Mirror hybrid option**			
	"Analog"		**"Near-term"**	
P_{fis}, GW$_{th}$	3		1.5	
Cycle length, EFPDs	311		311	
	Criticality, required fusion power			
	BOC	EOC	BOC	EOC
k_{eff}	0.95	0.90	0.97	0.945
P_{fus}, MW	35	74	11	20
	Neutron damage of 1. wall			
DPA, per cycle	12.3		5.7	
Lifetime, cycles	12-16		26-35	
	Tritium reproduction			
TBR, per cycle	1.9		3.5	
	Energy performance			
Q_e, per cycle	2.1		3.1	

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