INFCE

International
Nuclear
Fuel
Cycle
Evaluation

INFCE/DEP./WG.8/94

An Optimized Symbiotic Fusion and Molten-Salt Fission Reactor System

Attention Microfiche User,

The original document from which this microfiche was made was found to contain some imperfection or imperfections that reduce full comprehension of some of the text despite the good technical quality of the microfiche itself. The imperfections may be:

- missing or illegible pages/figures
- wrong pagination
- poor overall printing quality, etc.

We normally refuse to microfiche such a document and request a replacement document (or pages) from the National INIS Centre concerned. However, our experience shows that many months pass before such documents are replaced. Sometimes the Centre is not able to supply a better copy or, in some cases, the pages that were supposed to be missing correspond to a wrong pagination only. We feel that it is better to proceed with distributing the microfiche made of these documents than to withhold them till the imperfections are removed. If the removals are subsequestly made then replacement microfiche can be issued. In line with this approach then, our specific practice for microfiching documents with imperfections is as follows:

1. A microfiche of an imperfect document will be marked with a special symbol (black circle) on the left of the title. This symbol will appear on all masters and copies of the document (1st fiche and trailer fiches) even if the imperfection is on one fiche of the report only.

2. If imperfection is not too general the reason will be specified on a sheet such as this, in the space below.

3. The microfiche will be considered as temporary, but sold at the normal price. Replacements, if they can be issued, will be available for purchase at the regular price.

4. A new document will be requested from the supplying Centre.

5. If the Centre can supply the necessary pages/document a new master fiche will be made to permit production of any replacement microfiche that may be requested.

The original document from which this microfiche has been prepared has these imperfections:

☐ missing pages/figures numbered: _____

☐ wrong pagination

☒ poor [redacted] printing quality : Page 18 not readable.

☐ combinations of the above

☐ other

INIS Clearinghouse
IAEA
P. O. Box 100
A-1400, Vienna, Austria

79-119
Translated from Russian

INFCE WG8B

IAEh-2819

I.V. KURCHATOV INSTITUTE OF ATOMIC ENERGY

V.L. Blinkin and V.M. Novikov

AN OPTIMIZED SYMBIOTIC FUSION AND MOLTEN-SALT FISSION REACTOR SYSTEM

Moscow 1977

Key words: Molten-Salt Reactor (MSR), molten salts, tritium, blanket, uranium-233, uranium-thorium cycle, fusion reactor

ABSTRACT

A symbiotic fusion-fission reactor system which breeds nuclear fuel is discussed. In the blanket of the controlled thermonuclear reactor (CTR)*/ uranium-233 is generated from thorium, which circulates in the form of ThF_4 mixed with molten sodium and beryllium fluorides. The molten-salt fission reactor (MSR) burns up the uranium-233 and generates tritium for the fusion reactor from lithium, which circulates in the form of LiF mixed with BeF_2 and $^{233}UF_4$ through the MSR core. With a CTR-MSR thermal power ratio of 1:11 the system can produce electrical energy and breed fuel with a doubling time of 4-5 years. The system has the following special features: (1) Fuel reprocessing is much simpler and cheaper than for contemporary fission reactors; reprocessing consists simply in continuous removal of ^{233}U from the salt circulating in the CTR blanket by the fluorination method and removal of xenon from the MSR fuel salt by gas scavenging; the MSR fuel salt is periodically exchanged for fresh salt and the ^{233}U is then removed from it; (2) Tritium is produced in the fission reactor, which is a much simpler system than the fusion reactor; (3) The CTR blanket is almost "clean"; no tritium is produced in it and fission fragment activity does not exceed the activity induced in the structural materials; (4) Almost all the thorium introduced into the CTR blanket can be used for producing ^{233}U.

*/ Note by Languages Division: As will be seen, this is the term used in the Soviet version of the abstract in English.

1. Nuclear power can only develop if fuel can be bred [1]. For thermonuclear power (at least in the development stage in which a fusion reaction is produced in a deuterium-tritium plasma) it will be necessary to produce tritium. Thus, for nuclear power from both fission and fusion two of the key problems to be solved are those of breeding and fuel reprocessing.

Where fission power is concerned, an approach to the fuel problem is being sought in the construction of breeder reactors. At present, efforts are being concentrated mainly on the development of uranium-plutonium liquid-metal fast breeders with solid fuel rods. One of the serious problems presented by this system is the very high cost of fuel reprocessing; as fuel reprocessing technology develops and approaches industrial scale its cost increases continually [2]. This is to a large extent caused by problems encountered when working with highly irradiated fuel and by the even more stringent safety requirements in connection with the transport of irradiated fuel elements and radiochemical production operations.

A second problem, common to all nuclear reactors operating on the uranium-plutonium fuel cycle, is that of the disposal of actinides. These transuranium elements have lifetimes commensurate with geological eras, and no solution has as yet been found for this problem.

For nuclear power to develop at the required rate, with a doubling time of about ten years, breeder reactors will have to have doubling times of five years [1]. However, it is by no means certain that liquid-metal or any other breeder reactors will prove capable of such a doubling time.

A number of problems arising in connection with fuel reprocessing and disposal of long-lived actinides can be solved much more easily in molten-salt thermal uranium-thorium breeders with circulating nuclear fuel (see, for example, Ref. [3]). The absence of fuel rods and the fact that fuel can be reprocessed continuously make the fuel cycle for these reactors much cheaper [4], while use of the uranium-thorium fuel cycle lowers the formation rate of actinides by six orders of magnitude.

However, with the molten-salt thermal reactor it is difficult to achieve a doubling time of less than 15-20 years. Also, complete fuel

reprocessing with continuous removal of protactinium and rare-earth elements, which is the only way to obtain breeding, has not yet been demonstrated in plants on an industrial scale. The greatest difficulties with this type of reprocessing occur in connection with the extraction of rare-earth elements [5]. Thus, molten-salt breeders have so far not been fully proven to be economic or competitive.

Another possible way in which nuclear power could be developed - by the construction of controlled thermonuclear reactors (CTR) - is also beset with a number of complex technical problems. For fusion reactors to be supplied with fuel, the problem of tritium production must be solved. This is even by itself a complex enough task and is rendered even more complex by the fact that tritium can easily diffuse through structural materials at high temperatures in the blanket of the CTR. The fusion reactor itself is an extremely complex system full of special equipment, such as a cryogenic cooling system for the super-conducting coils of the electromagnets surrounding the blanket. This makes access to the blanket, where tritium is produced and thermal energy is generated, extremely difficult. The problems of tritium production and removal from the CTR blanket and of preventing tritium releases to the environment therefore remain today among the most difficult tasks to be faced.

Apart from "pure" fusion reactors, the possibility of constructing so-called "hybrid" fission-fusion reactors has also been widely discussed recently [6]. The blanket of a hybrid reactor contains such heavy elements as ^{238}U, thorium or the fissionable isotopes of uranium and plutonium, which can undergo fission by interacting with thermonuclear neutrons. Thanks to the fission reaction, much more energy can be generated in the blanket than would be generated by thermonuclear fusion. Part of this energy can be used to sustain the fusion reaction, for example, by injection of high-energy neutrals into plasma.

However, a considerable disadvantage of hybrid reactors is the contamination of the CTR blanket by fission products. This is a much more serious problem than with fission reactors, since the fusion reactor is a far more complex system. Thus, although in principle hybrid reactors could have distinct advantages as far as the production of both energy and fissionable materials for fission reactors is concerned, the high radioactivity of their blankets makes it necessary to treat them with a certain caution.

The main problems arising in connection with the independent development of fission and fusion power systems can thus be summarized as follows:

(1) Contamination of the blankets of hybrid reactors by fission products;

(2) Problems of tritium production in CTR blankets and its removal;

(3) Complexity and high cost of reprocessing fuel for solid-fuel fission breeder reactors;

(4) Accumulation of actinides in fission reactors or in the blankets of hybrid reactors operating on the uranium-plutonium fuel cycle;

(5) Long fuel doubling time (of the order of 20 years) with molten-salt thermal breeders.

2. These problems can be avoided or alleviated by building "symbiotic" systems, i.e. systems combining fission and fusion reactors. Symbiotic systems need to be organized in such a way that the advantages of each type of reactor are used to the maximum and their disadvantages reduced to a minimum.

The idea of a symbiotic system appears first to have been put forward by Lidsky [6]. Lidsky's system involves the combination of a molten-salt converter reactor (MSCR) [7] with a fusion reactor; in the blanket of the fusion reactor tritium is produced from liquid metallic lithium and ^{233}U is produced from thorium circulating in the form of ThF_4 in the mixture of molten salts of lithium and beryllium. The converter fission reactor produces most of the system's energy output, part of which can be used for sustaining the fusion reaction in the CTR. With each fusion reaction 1.08 atoms of tritium and 0.325 atoms of ^{233}U are generated in the CTR blanket. The total doubling time of fuel in this symbiotic system is seven years, with a fusion-fission reactor power ratio of 1:15 and a ^{233}U production coefficient in the MSCR of 0.96.

However, Lidsky's symbiotic system is not completely self-consistent, since it does not extract all the potential advantages inherent in the actual idea of symbiosis. The system is basically a simple combination

of a molten-salt converter with a thermonuclear reactor, with the latter reactor fulfilling the function of ^{233}U production. In this sense Lidsky's system is not an optimal one, since it embodies the disadvantages of each of the reactors constituting it.

In fact, the technology of tritium production and removal from liquid metallic lithium encounters the same problems as in a pure fusion reactor. Also, the maintenance of a high conversion efficiency in the MSCR requires, apart from continuous removal of gaseous fission products such as Xe and Kr from the reactor, fairly frequent changing of the fuel salt which is contaminated by fission products, unless protactinium and rare-earth elements are continuously removed from it. If the fuel is not continuously reprocessed in this way, the amount of thorium used in the fission reactor will be small and fuel cycle costs will increase accordingly [4].

Finally, because a single-salt MSCR has a localized production zone, where the neutron spectrum is harder and the fissionable fuel is used inefficiently, the specific fuel loading will be rather high. Thus, in order to achieve the necessary doubling time in this symbiotic system, it would be necessary to increase the contribution of the CTR to the overall power to an unjustifiably high level.

3. The question of how to find the most rational distribution of functions between the elements of a symbiotic system can be approached with a view to making problems of fuel materials production easier and simplifying them as far as possible. This approach can be seen as a search for an optimized symbiotic system. We believe that the answer can to a large extent be found in a symbiotic system whose general layout is shown in Fig. 1.

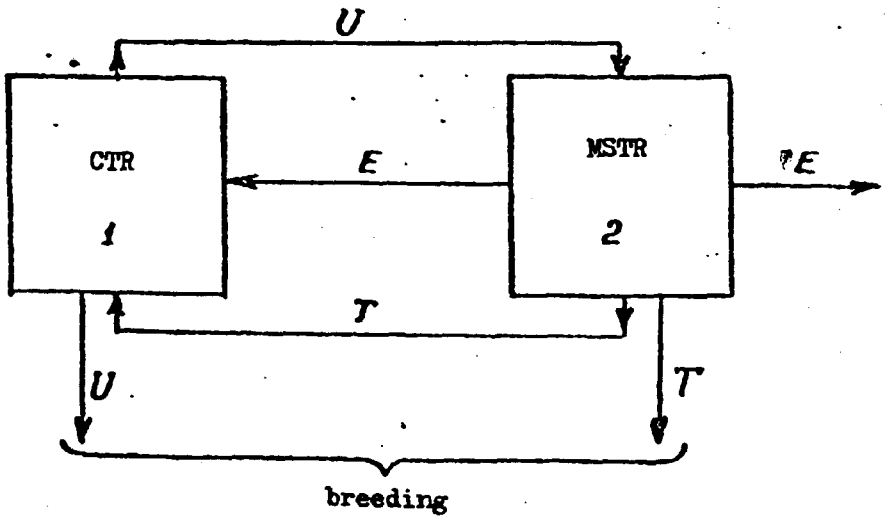

Fig. 1. Layout of a symbiotic system:

1 - Controlled thermonuclear reactor (CTR); 2 - Molten-salt thermal (fission) reactor (MSTR); E - energy generated by the MSTR; T - tritium channels; U - uranium-233 channels.

In this system the CTR fulfils the function merely of a producer of fissionable fuel (^{233}U), while tritium and most of the thermal energy are produced in the MSTR.

A mixture of liquid metal fluorides, including ThF_4, circulates through the CTR blanket. With neutron capture thorium is transmuted to protactinium which, with a half-life of 27 days, decays to ^{233}U. The protactinium and ^{233}U that are formed in the salt can be removed from the system through a bypass circuit [5]. This ensures that the equilibrium concentration of ^{233}U in the salt is sufficiently small and thus keeps the fission rate of uranium nuclei sufficiently low. Part of the ^{233}U produced in the CTR blanket passes into the fission reactor operating in association with the CTR; the rest represents the breeding gain.

The fission reactor, which together with the fusion reactor makes up the symbiotic system, is a molten-salt thermal reactor (MSTR) with a graphite moderator and reflector. It has a mixture of liquid fluorides of ^{233}U, lithium and beryllium circulating through its core. Part of the energy generated in the MSTR can be passed on to the CTR for sustaining the D + T fusion reaction. Xenon and krypton are continuously removed from the fuel salt circulating in the MSTR by gas scavenging [8].

A large proportion of the other fission products, rare-earth elements in particular, can be removed, for example, by the extraction method [5].

It is also possible for a simpler fuel reprocessing technique to be used, whereby the fuel salt is replaced periodically as the fission products accumulate; ^{233}U is recovered from the discarded salt by fluorination. In this case, only gaseous fission products are continuously removed from the fuel salt. If the salt is replaced every 4-6 years, a tritium conversion coefficient (K_T) of the order of 0.8-0.9 is ensured with the MSTR.

The fuel reprocessing operations performed in the symbiotic system under discussion, in both their complete and simplified forms, are shown in Table 1. With the performance of these operations the thorium circulating in the CTR blanket can be utilized to the maximum for conversion to ^{233}U.

Let us now consider the parameters of our symbiotic system. The fuel balance equations for the reactors constituting the system are as follows:

$$\frac{dN_1}{dt} = k_T \alpha R_2 - R_1, \qquad (1)$$

$$\frac{dN_2}{dt} = k_U R_1 - \alpha R_2 . \qquad (2)$$

where the subscripts 1 and 2 relate to fusion and fission reactions, respectively; N_i is the total number of tritium atoms in the CTR plasma or of ^{233}U atoms in the MSTR; R_i is the rate of the i-th reaction; K_T is the tritium conversion coefficient in the MSTR (the number of tritium atoms which are formed for each ^{233}U atom burnt); K_U is the ^{233}U conversion coefficient in the CTR (the number of ^{233}U atoms which are formed with each fusion reaction); α is the number of ^{233}U atoms burnt up in relation to the number of ^{233}U atoms undergoing fission.

Table 1

Fuel reprocessing operations with an optimized symbiotic system

System component	Operation	
	Complete fuel reprocessing	Simplest fuel reprocessing
CTR blanket	Continuous recovery of ^{233}U by fluorination and of protactinium by extraction	
MSTR	Continuous recovery of xenon and krypton by gas scavenging	
	Continuous recovery of tritium by gas scavenging	
	Maintenance of ^{233}U level in fuel salt	
	Periodic replacement of graphite moderator in core	
	Continuous extraction of rare-earth elements and other fission products	Periodic replacement of fuel salt and recovery of ^{233}U from it

If the symbiotic system operates according to a self-consistent regime in such a way that $\frac{1}{N_1} \cdot \frac{dN_1}{dt} = \frac{1}{N_2} \cdot \frac{dN_2}{dt} = \frac{1}{\tau} = \text{const.}$, then from Eq. (1) and Eq. (2) it follows that:

$$\frac{P_1}{P_2} = \frac{\alpha}{2Ak_u} \cdot \frac{W_1}{W_2} \left[\sqrt{(A+B)^2 + 4AB(k_u k_r - 1)} + A - B \right], \qquad (3)$$

$$\tau = \frac{AB}{A+B} \cdot \frac{2}{\sqrt{1 + 4(k_u k_r - 1)\frac{A+B}{(A+B)^2}} - 1}, \qquad (4)$$

where W_i is the energy generated by the i-th reaction; $P_i = R_i W_i$ which is the thermal capacity of the i-th reactor; $A = \frac{W_1 G_{10}}{m_1}$; $B = \frac{W_2 G_{20}}{\alpha m_2}$; m_1 and m_2 are the masses of tritium and ^{233}U, respectively; $G_{io} = \frac{m_i N_i}{R_i W_i}$ which is the specific fuel inventory of the i-th reactor.

If $\beta = \frac{A}{B} = \frac{W_1 G_{10}}{m_1} \cdot \frac{\alpha m_2}{W_2 G_{20}} \ll 1$, Eq. (3) and Eq. (4) will have the form:

$$\frac{P_1}{P_2} = \alpha k_T \frac{W_1}{W_2} \left[1 - \beta (k_U k_T - 1) + O(\beta^2) \right], \qquad (5)$$

$$T_2 = \frac{\varepsilon \ln 2}{L} = \frac{\varepsilon G_{20} \ln 2}{L (k_U k_T - 1)} \left[1 + \beta k_U k_T + O(\beta^2) \right], \qquad (6)$$

where $\varepsilon = \frac{W_2}{\alpha m_2} = 2.38$ MW · a/kg, the energy generated by the burn-up of a unit mass of ^{233}U; α is the load factor of the symbiotic system; T_2 is the fuel doubling time for the system as a whole.

The specific tritium inventory of a fusion reactor of the tokamak type is of the order of G_{10} = 1–6 g/MW(th) and the specific inventory of the MSTR is of the order of G_{20} = 0.3–0.7 kg/MW(th). Since W_1 = 20 MeV, W_2 = 200 MeV and α = 1.1 for ^{233}U with a thermal neutron spectrum, then

$$\beta = \alpha \frac{W_1}{W_2} \cdot \frac{G_{10}}{G_{20}} \cdot \frac{m_2}{m_1} \leq 1{,}1 \frac{20}{200} \cdot \frac{0{,}006}{0{,}3} \cdot \frac{233}{3} = 0{,}17 \ll 1$$

and, consequently, for purposes of calculations Eq. (5) and Eq. (6) may be used; for the symbiotic system under consideration they have the following forms:

$$\frac{P_1}{P_2} = 0{,}11 k_T \left[1 - \beta (k_U k_T - 1) + O(\beta^2) \right], \qquad (7)$$

$$T_2 = 2{,}063 \frac{G_{20}}{k_U k_T - 1} \left[1 + \beta k_U k_T + O(\beta^2) \right] \text{ years}, \qquad (8)$$

where G_{20} is measured in kg/MW(th) and the load factor L is taken to be 0.8.

From Eq. (7) and Eq. (8) it will be seen that the symbiotic system can operate under a self-consistent regime when the power ratio between the fusion and fission reactors if of the order of 1:10. In this case, the fuel doubling time T_2 scarcely depends on the specific tritium inventory of the fusion reactor. The system will be breeding fuel (where $T_2 > 0$) if $K_U K_T > 1$.

In order to perform a numerical evaluation of the symbiotic system parameters let us take the thermal capacity of the molten-salt reactor P_1 to be 2250 MW, which, with an efficiency of 0.44, gives an electrical power of 1000 MW. The layout of an MSTR-1000 is shown in Fig. 2.

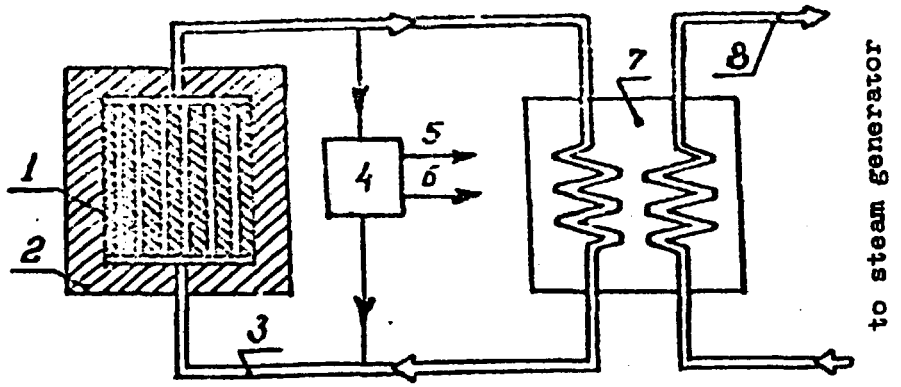

Fig. 2. Layout of MSTR

1 – core; 2 – reflector; 3 – fuel salt circuit; 4 – fuel reprocessing facility; 5 – tritium removal channel; 6 – fission product removal channel; 7 – heat exchanger; 8 – salt coolant circuit.

The neutron physics parameters of MSTR-1000 were calculated using the "9M" program in a 26-group P_1 approximation. With the dimensions of the reactor as given in Table 2, the volume of fuel salt in a core filled with graphite moderator, the concentration of uranium in the fuel salt and the isotopic composition of the lithium incorporated in the fuel salt in the form of LiF were varied. The parameters of the reactor were chosen on the basis of these calculations in such a way that the fuel doubling time of the symbiotic system for each of the values given for the uranium conversion coefficient K_U in the CTR blanket would be kept to a minimum. The fuel salt inventory in the circuit (see Fig. 2) was found from the calculated proportion by volume of the core occupied by fuel salt and from the salt inventory in the outer part of the circuit, which was taken to be the same as in the MSBR-1000 reactor [9]. The equilibrium concentration of fission products and transthorium elements in the fuel salt was taken to be the same as in the equilibrium fuel cycle for the MSBR [3].

The main thermal parameters are given in Table 2 and the physical parameters, optimized for a minimum doubling time, are given in Table 3. Table 3 shows that, even with rather a low uranium conversion coefficient in the CTR blanket (K_U = 1.3), the doubling time does not exceed 10 years. This is explained by the very small specific inventory of the MSTR, which can be even smaller than that of a reactor of the MSBR type; this is due to the fact that in the MSTR tritium is produced mainly in the thermal zone of the neutron energy spectrum and thus, unlike the MSBR, with the MSTR it is not necessary to arrange a separate breeding zone, where isotopes undergoing fission are utilized inefficiently.

Table 2

Thermal parameters of MSTR-1000

Parameters	Values
Thermal capacity (MW)	2250
Net electric capacity (MW)	1000
Full thermal efficienty (%)	44.0
Reactor dimensions (m):	
Core diameter	4.0
Core height	4.0
Thickness of graphite reflector	1.0
Flow rate of fuel salt in circuit (m^3/s)	2.63
Fuel salt inventory (m^3):	
In circulation circuit	31.9
In reprocessing facility	13.6
Mean velocity of fuel salt in core (m/s)	1.6
Proportion by volume of fuel salt in core (%)	20.0
Average power density in fuel salt in core (W/cm^3)	225.0
Composition of fuel salt (mol.%)	
LiF	50.0
BeF_2	50.0
$^{233}UF_4$	< 0.1
Properties of fuel salt:	
Density at 650°C (g/cm^3)	2.20
Specific heat (J/kg.°C)	2810
Melting point (°C)	356

Table 3

Physical parameters of MSTR-1000 and parameters of a symbiotic system optimized for minimum doubling time

K_u	C_{20} (kg/MW(th))	K_T	$^{233}UF_4$ in salt (mol.%)	^6Li in Li (mol.%)	$T_{2'}$ (years)	P_1/P_2
1,3	0,70	0,912	0,121	0,162	8,6	0,100
1,4	0,50	0,847	0,087	0,008	6,3	0,004
1,5	0,475	0,837	0,082	0,080	4,4	0,0025
1,6	0,45	0,827	0,078	0,084	3,3	0,001

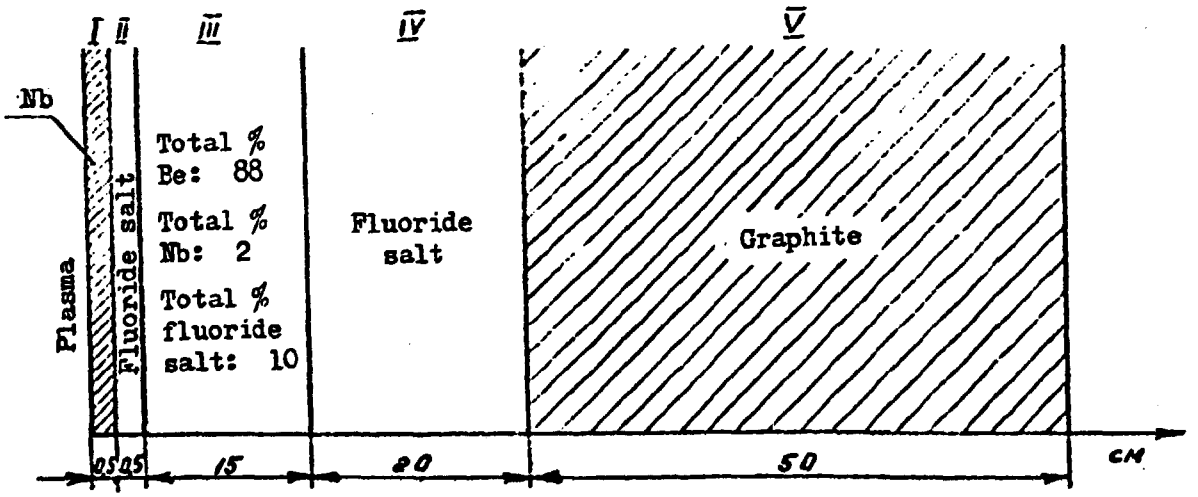

Fig. 3. Diagram of CTR blanket

The parameters for MSTR-1000 given in Table 3 relate to complete fuel reprocessing.

It is important to note that the extraction of rare-earth elements from the MSTR fuel salt is much easier than a similar process would be with the MSBR fuel salt, since the MSTR salt does not contain protactinium and thorium, which means that the problem of separating the rare-earth elements from them does not arise.

From Table 3 it can be seen that, for the doubling time not to exceed five years, the value of K_U in the CTR blanket must be not less than 1.47. This value can be achieved in a blanket in which there is no fission of heavy elements only if elements with a sufficiently high reaction cross-section (n,2n) are included in the composition of the blanket. One of the best materials from this point of view is beryllium. Since metallic beryllium dissolves in a fluoride salt, it needs to be placed in the blanket with a protective sheath. Niobium, molybdenum, tungsten or a nickel-base alloy, for example, Hastelloy-N [9], can be used for this protective sheath.

A possible CTR blanket composition, which ensures the necessary K_U value, is shown in Fig. 3. The first wall is made of niobium and zone II is a channel 0.5 cm wide, through which a mixture of molten fluorides circulates and cools the first wall. Zone III contains a lattice of beryllium rods sheathed in niobium and cooled by the flow of salt, which occupies 10% of the volume of this zone. Zone IV is the ^{233}U conversion zone, through which salt containing thorium circulates. The graphite in zone V serves as a neutron reflector.

Two salt compositions were considered. The first composition (^7LiF: 71 mol.%, BeF_2: 2 mol.% and ThF_4: 27 mol.%) is the same as that considered by Lidsky [6]. The melting point of this salt is approximately 560°C, and its density at 650°C is 4.5 g/cm^3. The lithium contained in this salt needs to be highly enriched in lithium-7 (up to 99.999%), so as to limit to the utmost any formation of tritium in the blanket. The second salt composition has the same amounts of the components as the first, but uses NaF instead of ^7LiF. The thermophysical properties of the second salt do not differ very substantially from the salt containing LiF, but since sodium has a larger neutron capture cross-section than lithium-7, the K_U will be somewhat lower. However, the absence of lithium in the salt completely prevents the formation of tritium in the blanket and thereby ensures a very "clean" system.

The blanket parameters were calculated by the Monte Carlo method using the one-dimensional BLANK program [10]. Calculations for blankets with the geometries of zones III and IV optimized for a maximum K_U value are given in Table 4. It will be seen from this table that in a blanket with lithium salt (version No. 1) a K_U value of 1.59 can be achieved, which gives a doubling time for the symbiotic system of T_2 = 3.3 years

with a CTR-MSTR power ratio P_1/P_2 = 1:11. Where sodium salt is used (version No. 2), K_U = 1.51 and T_2 = 4.2 years when P_1/P_2 = 1:10.8.

Table 4

CTR blanket parameters

Version No.	Salt composition (mol.%)	No. of (n,γ) reactions for Th*/ (K_U)	No. of Li(n;n',α)T reactions (K_T(')*/)	No. of Th nuclei undergoing fission*/
1	^7LiF - 71 BeF_2 - 2,0 ThF_4 - 27	1,59	0,035	0,0085
2	NaF - 71 BeF_2 - 2.0 ThF_4 - 27	1,51	0,00	0,0110

*/ Normalized for one thermonuclear neutron.

Thus it is possible to obtain an entirely satisfactory T_2 value without any tritium at all being produced in the CTR blanket.

The fission rate of thorium nuclei in the blanket represents 1.1% of the fusion reaction rate. The fission rate of ^{233}U nuclei depends on the intervals between removals of protactinium and uranium from the blanket. The maximum permissible interval between removals of these elements can be found if the fission rate of ^{233}U does not exceed the fission rate of thorium. If uranium and protactinium are removed from the salt at intervals of ten days, the fission rate of ^{233}U will represent 30% of the fission rate of thorium.

Obviously, the achievement of minimum radiation contamination of the blanket in any thermonuclear reactor is governed by the activity induced in the structural materials. This induced activity does not

depend greatly on the types of material used to fabricate the blanket and amounts to a few curies per watt of thermal power produced by the reactor [11].

In the CTR blanket considered by us the fission-fragment activity in the salt, which occurs as a result of the fission of heavy nuclei, is several times lower than the activity induced in the niobium used to fabricate the first wall and the sheathing of the beryllium rods (see Table 5). The specific activity of the salt is two orders of magnitude lower than that of the niobium.

Table 5

Activity of salt and niobium in CTR blanket

Material	Activity	
	Specific (Ci/W)	Per unit volume (Ci/cm^3)
Nb	1.0 - 2.5 [11]	200 - 500 [11]
Fluoride salt	0.4	1.8

In order to provide conditions of turbulent flow for the salt cooling the first wall of the blanket, the velocity of the salt must be at least 1.5 m/s. With an energy flux through the wall of 1 MW/m^2 the salt in zone II will be heated by 3.5°C for every running metre of channel. The temperature drop occuring between the niobium wall and the coolant in this case reaches 30°C. In zone III salt with the same velocity is heated 7.9°C/m, while in zone IV, where for turbulent flow to be achieved a velocity of 0.11 m/s is sufficient, the salt is heated by 1.7°C/m. With the coolant heated to these levels and with a blanket inlet temperature of 630°C, the circulation of the salt can be arranged in such a way that at the outlet from each section of the blanket the salt temperature does not exceed 700°C and the temperature of the first wall does not exceed 750-800°C.

A special feature of the symbiotic system under consideration is the use of beryllium as a neutron multiplier; the weight of beryllium in the blanket of a thermonuclear reactor of the tokamak type is 560 t with

torus dimensions R = 11.4 m and r = 5 m. For the symbiotic system as a whole, with an energy flux through the first wall of the CTR of 1 MW/m^2, the total weight of beryllium, including the beryllium contained in the mixture of salts circulating in the CTR blanket and the MSTR core, is 25 t/GW of thermal power produced by the symbiotic system. The consumption of beryllium due to burn-up in the neutron flux is 0.26 t/GW·a.

With world reserves of about 600 000 t [12], the quantity of beryllium available would be enough to generate a thermal energy of 540 TW·a in symbiotic devices of the type under consideration. This means that if symbiotic devices with a total capacity of 18 TW(th) were constructed they would have enough beryllium to operate for 30 years.

4. With the symbiotic system of fusion and fission reactors considered by us fuel can be bred with a doubling time of 3.5-4.5 years and a thermal power ratio between the fusion and fission reactors of the order of 1:10. This system has the following characteristics:

(1) The fuel reprocessing technology is considerably simpler and cheaper than in the case of the fission breeder reactors currently being developed. Complete reprocessing of the fuel in the MSTR fission reactor requires continuous removal of the gaseous fission products from the circulating fuel salt by gas scavenging and extraction of other fission products, in particular, the rare-earth elements.

It is possible to arrange the fuel cycle in such a way that the only operations consist of the continuous removal of ^{233}U and protactinium from the CTR blanket and of gaseous fission products from the MSTR fuel salt. The fuel salt in this case is exchanged periodically as fission products accumulate, and ^{233}U is then recovered from it;

(2) Tritium for fusion reactors is produced in the fission reactor, which is a simpler system than the fusion reactor. The formation of tritium atoms in the fluoride medium of the MSTR fuel salt causes the tritium to be oxydized to TF, which greatly decreases tritium diffusion through structural materials. Continuous removal of tritium fluoride from the fuel salt can be performed by gas scavenging;

(3) The blanket of the thermonuclear reactor is comparatively "clean" in terms of radioactive contamination. No tritium is produced in it, and the fragment activity of heavy nuclei fission products does not increase the activity induced in the structural materials;

(4) Practically all the raw material for the production of fissionable materials is used up in the system; in principle, all the thorium introduced into the CTR blanket can be used for producing ^{233}U.

The advantages of the symbiotic system discussed are achieved by a more rational distribution of functions between its two main constituent parts. In this sense this symbiotic system is an optimal one.

The authors are grateful to G.E. Shatalov for enabling them to use the BLANK program for the CTR blanket calculations and to Yu.N. Brikov and S.A. Konakov for performing the calculations for the fission and fusion reactors.

REFERENCES

[1] ALEKSANDROV, A.P., Atomnaya ehnergetika i nauchno-tekhnicheskij progress (Nuclear power and progress in science and technology), in: "Atomnoj ehnergetike XX let" (20 years of nuclear power), Atomizdat, Moscow (1974) 205-213 [in Russian].

2. Sesnikoff N., Expensive enrichment.- "Environment", 1975, v.17, N 5, p. 28-35.

[3] BLINKIN, V.L., Teplovye ehnergeticheskije bridery s tsirkuliruyushchim yadernym goryuchim (obzor) (Review paper on thermal power-producing breeders with circulating nuclear fuel), IAEh preprint 2589, Moscow (1975) [in Russian].

4. Shimazu Y., Fuel cycle cost analysis on molten-salt reactors.- "Nucl. Sci. Thecnol.", 1976, v.13, N 4, p.190-198.

5. Engineering development of MSBR fuel recycle.- "Nucl. Appl. Technol.", 1970, v.8, p. 170-178. Auth.: Whatley M.E., McNeese L.E., Carter T.L. e. a.

6. Lidsky L.M., Fission-fusion systems: hybrid, symbiotic and rugean.- "Nucl. Fusion", 1975, v.15, N 1, p.151-174.

7. Perry A.M., Molten-salt converter reactors.- "Appl. Nucl. Energy", 1975, v.2, p. 809-815.

8. Scott D., Matherly W.F., Graphite and xenon behavior and their influence on molten-salt reactor design.- "Nucl. Appl. Technol.", 1970, v.8, p. 179-189.

9. Bettis E.S., Robertson R.C., Design and performance features of a single-fluid molten-salt breeder reactor.- "Nucl. Appl. Technol.", 1970, v.8, p.190-270.

[10] MARIN, S.V., MARKOVSKIJ, D.V., SHATALOV, B.E., Raschet potokov nejtronov i ehnergovydelenij v blankete termoyadernogo reaktora (Calculation of neutron flux and energy production in the blanket of a thermonuclear reactor) in: "Trudy Vsesoyuznogo soveshchaniya po inzhenernym probleman upravlyaemogo termoyadernogo sinteza" (Proc. All-Union Conference on engineering problems of controlled thermonuclear fusion), Vol. 3, Leningrad (1975) [in Russian].

11. Transmutations, radioactivity, and afterheat in a deuterium-tritium tokamak fusion reactor.- "Nucl. Technol.", 1974, v.22, p. 379-391. Auth.: Vogelsang W.F., Kulcinski G.L., Latt B.W., Sung T.Y.

12. Nass H.W., Potential availability of U.S. and World beryllium mineral resources in the period 2000-2020 A.D.- "Trans. Am. Nucl. Soc.", 1975, v.22, N 16-21, p.57.

Editor: E.D. Markova
Proof-reader: N.N. Cheremnykh
T-01683 5 March 1977 Format 60 x 90/8 Uch.-izd. l. 1,10
No. of copies: 157 Order No. 369 Price 11 kopecks ONTI-IAEh