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by

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ABSTRACT

Calculations have been made of the production of tritium when 14 MeV neutrons are incident on various neutron blankets. Neutron migration and reactions were computed with a 25 group DSN calculation. Blankets were considered containing normal Li, Be, C, F with, sometimes, Cu or Mo between the neutron source and blanket. Blankets of mostly Be, containing enough Li to absorb all slow neutrons, appeared most attractive for tritium regeneration. For example a 60 cm thick blanket of 50 volume % Be and 50% Li having 3.3 cm of Mo between source and blanket, gave a production of 1.79 tritons per incident neutron. Results are summarized in Table III, and we conclude that by use of Be one can achieve tritium regeneration with considerable flexibility of blanket design.

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The author is indebted to Fred Ribe and James Tuck for calling his attention to current interest in problems of blanket design. He is grateful to Roger Lazarus for generating the required multigroup cross sections from libraries of evaluated nuclear data and to Laura Stone for running the DTK problems. The consideration of Be_2C as a blanket material was suggested by David Rose.

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TABLES

I. Introduction

In a thermonuclear reactor which generates energy by reactions of deuterons with tritons, it is necessary for practical operation that more tritons be produced than are consumed in the thermonuclear reactions. To this end one thinks of surrounding the reacting plasma by a thick blanket in which the emerging 14 MeV neutrons should produce as many tritons as possible. Lithium is an obvious choice for a blanket material. A fast neutron on Li^7 will, with fairly high probability, produce a triton and neutron by the $Li^7(n,n'T)\alpha$ reaction, and then the resulting neutron may be moderated and captured by $Li^6(n,T)\alpha$ to produce a further triton. By using normal lithium (7.4% Li^6) one can achieve ratios of tritons produced per neutron incident which approach a factor two, as will be seen.

There are, however, both physical and engineering reasons for thinking that a blanket of mostly lithium may not be the best choice. It seems clear that one will want some Li^6 in the blanket to absorb the slow neutrons. Li^6 and He^3 are the only stable nuclei which have large (n,T) cross sections for slow neutrons, and we may rule out He^3 as too rare or produced by the decay of tritium in the first place. However for producing the slow neutrons and/or tritons from the incident 14 MeV neutrons, Li^7 may not be the best material. In particular Be may be superior by virtue of its (n,2n) cross section, which is somewhat larger than the $Li^7(n,n'T)\alpha$ cross section (see Table I). Also both neutrons from the Be(n,2n) reaction may be capable of further multiplication, whereas there is only one such neutron from the (n,n'T) reaction. In addition Be has a higher atomic density than Li so that thinner blankets may be effective.

For a pulsed thermonuclear reactor one can consider a blanket outside of the plasma-containing magnetic field;[1] and a blanket of Be, cooled by lithium, might not be far from a practical possibility. A few cm of conductor such as Cu would then be required inside the blanket for producing the field.

For a steady state thermonuclear reactor, it is commonly felt (see, for example, reference 2) that the field coils must be superconducting (or very cold anyway) and hence located outside of the blanket, so as to avoid neutron heating. Since the blanket is now exposed to a magnetic field it becomes attractive to think of circulating a cooling fluid through the blanket which has a lower electrical conductivity than Li metal. Molten fluorides of Be and Li (LiF and BeF_2) have been considered for such purposes. Again, the use of Be as a primary blanket material may lead to higher tritium regeneration than use of fluorides alone.

The purpose of this note is to report some calculations of tritium production in various blanket materials.

II. Calculations

All calculations were made using the DTK neutron transport code which is a version of the DSN method[3] in current use at Los Alamos, with 25 energy groups and the S_4 approximation. The geometry was always described by infinitely long coaxial cylinders. For such preliminary calculations we believe that approximations involved in the transport code and geometry idealization are unimportant.

Cross sections were taken from the libraries of evaluated nuclear data of the Lawrence Radiation Laboratory at Livermore, and AWRE at Aldermaston in the UK, as on tape at Los Alamos on July 8, 1965. In particular, cross sections for Cu and Mo were taken from the LRL tape while those for Be, Li^6, Li^7, and F were taken from the AWRE tape. The reduction of the cross sections to multigroup form suitable for the DTK code was made by Roger Lazarus of this Laboratory. Of the cross sections which were used, those for Be, Li^6, and Li^7 have received much careful attention and are presumably quite good. The Cu and Mo cross sections have not been studied as much, but they are believed to be quite reasonable. The F cross sections are the most uncertain. We believe that the AWRE cross sections for neutron absorption by fluorine may be too high, at high energies, and accordingly tried a calculation in which they were reduced, as will be noted. The spectrum of neutrons inelastically scattered by F is quite uncertain.

The energy groups which were used are listed in Table I, together with the group cross sections for some of those reactions which most

TABLE I

Group Cross Sections for Some Important Reactions

Energy Group	Lower Energy	Cross Sections (barns)					
		$Li^6(n,T)\alpha$	Be(n,2n)	$Li^7(n,n'T)\alpha$	F(n,abs)	Cu(n,2n)	Mo(n,2n)
1	12.0 MeV	0.028	0.514	0.356	0.18	0.72	0.98
2	8.3 MeV	0.038	0.555	0.408	0.19	0.08	0.32
3	5.3 MeV	0.065	0.596	0.378	0.25		
4	3.4 MeV	0.104	0.552	0.061	0.09		
5	2.2 MeV	0.167	0.303	0.001			
6	1.4 MeV	0.223	0.002				
7	0.9 MeV	0.279					
8	0.58 MeV	0.352					
9	370.0 KeV	0.727					
10	240.0 KeV	2.02					
11	150.0 KeV	1.58					
12	100.0 KeV	0.866					
13	31.6 KeV	0.93					
14	10.0 KeV	1.41					
15	3.16 KeV	2.18					
16	1.0 KeV	3.6					
17	0.316 KeV	6.5					
18	100.0 eV	11.5					
19	31.6 eV	20.3					
20	10.0 eV	36.4					
21	3.16 eV	64.7					
22	1.00 eV	114.0					
23	0.316 eV	203.0					
24	0.026	426.0					
25	thermal	950.0					

importantly affect the neutron-tritium production in the blanket. The notation F(n,abs) stands for all reactions with fluorine from which no neutrons emerge. It may be noted from Table I that the Be(n,2n) cross section is larger than the $Li^7(n,n'T)$ cross section and also has a lower threshold.

In the DTK calculation, the S_4 option was used and the transport approximation was used to treat anisotropic scattering. This approximation may introduce some uncertainty in the spatial distribution of neutron interactions. However for such preliminary calculations in which the main emphasis is on the neutron-tritium economy, it is believed that the transport approximation should be entirely adequate.

In Table II are given the atomic densities of the various materials used in the calculations. For lithium metal and Flibe (a mixture of LiF and BeF_2), densities were taken from reference 4 corresponding to a temperature of 1300°F.

TABLE II

Atomic Densities $\left(\frac{\text{atoms}}{\text{cm}^3}\right) \times 10^{24}$

Material	Densities			
Li	Li^6(0.0030)	Li^7(0.0378)		
Flibe	Li^6(0.0012)	Li^7(0.0142)	Be(0.0167)	F(0.0486)
Be	Be(0.123)			
Be_2C	Be(0.076)	C(0.038)		
Mo	Mo(0.064)			
Cu	Cu(0.085)			

III. Results

Results of the calculations are given in Table III. The geometry is indicated by the radii R_1, R_2, and R_3 (cm) of coaxial cylinders. Inside R_1 there is always a void (corresponding to the plasma) where the neutron source is uniformly distributed. Material number one, M_1, occupies the region between R_1 and R_2 and material M_2 then extends to R_3.

As is indicated in Table III, T_6 represents the number of tritons formed by $Li^6(n,T)\alpha$ reactions per incident 14 MeV neutron; T_7 is the number of tritons formed by $Li^7(n,n'T)\alpha$ reactions per incident 14 MeV neutron; and L is the number of neutrons leaking out of the system per incident neutron. The total tritons produced per incident neutron is thus $T = T_6 + T_7$; or if one assumes that each escaping neutron can be converted into a triton by a suitable extension of the calculated blanket, then the total triton production is $T_6 + T_7 + L \equiv T^+$.

A number of conclusions can be made from the results of Table III.

(1) From the infinite medium results we would conclude that Flibe is a marginal blanket material. It appears to be bad not only because of absorption by the fluorine but more significantly because of inelastic scattering by fluorine which can degrade the fast neutrons below the $Li^7(n,n'T)\alpha$ and Be(n,2n) thresholds. As mentioned previously, we feel that the fluorine absorption cross sections which were used are larger than justified by the current rather uncertain experimental data.[5] We therefore repeated problems TU01 and TU1B with the fluorine absorption

TABLE III

Triton Production per 14 MeV Neutron Incident on Various Blankets

T_6 is the triton production by $Li^6(n,T)\alpha$ reactions, T_7 is the triton production by $Li^7(n,n'T)\alpha$ reactions, and L is the leakage -- all per incident 14 MeV neutron, $T = T_6 + T_7$ and $T^+ = T_6 + T_7 + L$. The geometry is cylindrical, with the neutron source inside R_1, material M_1 between R_1 and R_2, and material M_2 between R_2 and R_3.

Problem Number	Geometry					Triton Production				
	R_1	M_1	R_2	M_2	R_3	T_6	T_7	L	T	T^+
TU0	0	Li	∞	infinite medium		1.05	0.83	0	1.88	1.88
TU01	0	Flibe	∞	infinite medium		1.02	0.10	0	1.12	1.12
TU3	0	Be + 0.001 Li^6	∞	infinite medium		2.74	0	0	2.74	2.74
TU3A	0	Be_2C + 0.001 Li^6	∞	infinite medium		2.03	0	0	2.03	2.03
TU1A	77	Li	112.0			0.16	0.59	0.88	0.75	1.63
TU1B	77	Li	137.0			0.40	0.73	0.64	1.13	1.77
TU1C	77	Li	177.0			0.72	0.80	0.33	1.52	1.85
TU2A	77	Flibe	112.0			0.73	0.10	0.29	0.83	1.12
TU2B	77	Flibe	137.0			0.97	0.10	0.05	1.07	1.12
TU2C	77	Flibe	177.0			1.02	0.10	0	1.12	1.12
TU4	10	Cu	13.3	Li	113.3	0.81	0.39	0.36	1.20	1.56
TU4A	10	Mo	13.3	Li	113.3	0.86	0.41	0.39	1.27	1.66
TU5	10	Void	13.3	0.5 Be + 0.5 Li	73.3	2.09	0.22	0.15	2.31	2.46
TU5A	10	Void	13.3	0.75 Be + 0.25 Li	73.3	2.47	0.09	0.06	2.56	2.62
TU6	10	Cu	13.3	0.5 Be + 0.5 Li	73.3	1.60	0.11	0.08	1.71	1.79
TU6′	10	Mo	13.3	0.5 Be + 0.5 Li	73.3	1.70	0.12	0.09	1.82	1.91
TU6A	10	Cu	13.3	0.5 Be_2C + 0.5 Li	73.3	1.35	0.11	0.10	1.46	1.57
TU6B	10	Cu	13.3	0.75 Be + 0.25 Li	73.3	1.75	0.05	0.04	1.80	1.84
TU7	76	Mo	77.0	0.5 Be + 0.5 Flibe	117.0	1.46	0.04	0.13	1.50	1.63
TU7A	76	Mo	77.0	0.75 Be + 0.25 "	117.0	1.76	0.02	0.12	1.78	1.90
TU7B	76	Mo	77.0	Flibe	117.0	0.87	0.08	0.19	0.95	1.14

cross section reduced by 0.12 b in the top two groups. This increased T_6 and T by about 10%, indicating that most of the bad effect in Flibe is due to the fluorine inelastic scattering. The spectrum of inelastically scattered neutrons is probably quite uncertain so that the results for Flibe are more uncertain than the other results in Table III. Nevertheless, it appears to us that pure Flibe is a very marginal blanket material. This conclusion is at variance with other reported calculations,[4] but in good agreement with the results of Impink.[2]

(2) From the infinite medium calculations we see that Be is an excellent potential blanket material and the Be_2C is about as good as Li.

(3) In problems TU1A-1C we see that an acceptable blanket of pure Li must be quite thick. With thin blankets, too many neutrons leak out of the system. In principle these leakage neutrons could be captured, for example in a further layer of Li^6H and the values of T^+ which could be thus achieved are not far from infinite medium values.

(4) In problems 4, 5, and 6 we consider some idealized blankets which might surround a pulsed thermonuclear reactor. Those containing Be are seen to be thinner and more productive of tritium than are the pure Li blankets. A conductor of 3.3 cm Cu or Mo inside the blanket is included in problems 4 and 4a, as well as problems 6. From problems 4 we observe that substantial tritium regeneration ($T \geq 1.2$, $T^+ \simeq 1.6$) can be achieved with thick Li blankets outside of the Cu or Mo. From problems 6 we conclude that addition of Be or Be_2C leads to excellent regeneration with thinner blankets.

(5) In problems 7 we consider blankets for a steady state thermonuclear system, where Flibe is used as coolant because of the magnetic fields in the reactor. Once again, with 50% Be in the blanket, or more, good tritium regeneration is found. With Flibe, even the small multiplication of a 1 cm Mo shell has a beneficial effect.

IV. Discussion

We have seen that blankets of Be cooled by Li or Flibe appear attractive for tritium regeneration. For a blanket of about 50 cm thickness, we have found $1.5 \lesssim T \lesssim 2.0$, where T is the number of tritons produced per 14 MeV neutron incident. Even for a practical system where pipes of other materials, end losses, and the like would degrade regeneration, it could be expected that T would be substantially greater than unity.

Of course Be has some disadvantages as a blanket material. It is not cheap and is damaged by irradiation with fast neutrons -- precisely by the $Be^9(n,2n)2\alpha$ and $Be^9(n,\alpha)He^6$ reactions which liberate helium within the Be. If one were content with fairly small values of T, then more modest use of Be in regions of high fast neutron flux would suffice. At any rate it seems clear that the use of Be gives one quite a lot of flexibility in achieving high tritium regeneration.

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