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TRITIUM BREEDING AND DIRECT ENERGY CONVERSION

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TRITIUM BREEDING AND DIRECT ENERGY CONVERSION

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ABSTRACT

Tritium breeding and direct energy conversion are key technological problems in the development of practical fusion reactors. Lithium in some form is needed for tritium generation, but certain chemical problems must be overcome. Alternatives to lithium metal should be considered. Better methods must be developed for efficiently recovering the tritium and safely disposing of the unwanted byproducts. Separation of the fusion fuel and waste products will be especially challenging for mirror machines with direct energy conversion. Hopefully, in the long run, these machines will employ the DHe^3 fuel cycle, but this will impose additional demands for efficient helium isotope separation and, possibly, tritium storage in some manner that allows for recovery of the He^3 decay product.

INTRODUCTION

To realize the promise of fusion, we must resolve many questions deriving not only from the basic problems of plasma physics but also from the peripheral technology. The plasma physics problems are, of course, of paramount importance. However, the technological aspects are of growing significance, particularly as they relate to the solution of the basic physics problems.

Two technological areas of immediate concern are tritium breeding and direct energy conversion. Efficient breeding is essential to the development of a practical DT fusion reactor, as we shall discuss below. It requires lithium-containing blankets to surround the plasma. Neutronic and heat transfer studies indicate that flowing lithium is quite attractive for the dual role of tritium breeding and heat transfer. But the chemical problems of hot lithium compatibility with containment material and the need to insulate the flowing lithium from metal tube walls must be solved. Whatever form the blankets take, methods must be developed for recovering the generated tritium efficiently, while tritium holdup is kept to a minimum.

Neutron activation of the blanket's structural material is inevitable. So material selection will be a compromise between neutronic and structural requirements and the chemical requirements for safe and efficient storage of the radioactive wastes. Certain materials, especially niobium, look attractive in this regard.

As presently conceived, fusion reactors will "burn" only a small percentage of the injected fuel. Therefore, efficient collection and separation of the fusion fuel and waste products is necessary. This will be especially challenging for mirror machines with direct conversion because of the large volumes and areas involved. Hopefully, these machines may someday use the DHe³ fuel cycle, but this in turn suggests other problems, which we will discuss.

TRITIUM BREEDING

Why are we especially interested in tritium breeding? To answer this question, we must first consider the principal fusion reactions. These involve the isotopes deuterium (D), tritium (T), and helium-3 (He³):

Fig. 1. Fusion reactor probabilities: values of σv based on Maxwellian distribution for DT, DD (total), and DHe^3 reactions.

Of these isotopes, nature supplies us only one: deuterium. Based on availability, then, the DD reaction would seem the obvious choice. But we find that confining matter at the temperature, density, and times needed to overcome the coulomb barrier and fuse is exceedingly difficult. So if we are to simplify the confinement problem we need the highest fusor reaction probabilities (cross sections) at the lowest temperatures.

When we look at fusion reaction probabilities (Fig. 1), we see that the DT reaction is about 100 times more probable than DD in the temperature range 10-100 keV (1). Because of this difference in reaction probability, DT fusion should be far easier to attain than DD fusion.

For leaky containment - which is what we find - a good measure of performance is the ratio of produced fusion power to power injected for maintaining the reactants at the required temperature. This ratio is called Q .

The power injected per unit volume, P_{in} , depends on the leakage rate and energy:

$$P_{in} = 2 \left(\frac{3}{2} n k T \right) (1/\lambda)$$

where $n \equiv$ ion density in ions/cc, $kT \equiv$ ion kinetic energy in keV, and $\lambda \equiv$ ion mean lifetime.

On the other hand, the fusion power produced, p_f , for the DD reaction is

$$P_f = (n_D^2/2) \overline{\sigma v}_{DD}(T) E_{DD} ,$$

where n_D = deuterium density, $\overline{\sigma v}_{DD}$ = the DD fusion reaction probability, and E_{DD} = DD fusion energy release.

For the DT reaction,

$$P_f = n_D n_T \overline{\sigma v}_{DT}(T) E_{DT} ;$$

and if

$$n_D = n_T = 1/2 n_{DT} ,$$

then

$$P_f = (n_{DT}^2/4) \overline{\sigma v}_{DT} E_{DT} .$$

We can now compare the Q values for DD and DT reactions, assuming the same plasma density, temperature, and lifetime:

$$\frac{Q_{DT}}{Q_{DD}} = \frac{\frac{n_{DT}^2}{4} \overline{\sigma v}_{DT} E_{DT}}{\frac{n_D^2}{2} \overline{\sigma v}_{DD} E_{DD}} ,$$

since

$$\overline{\sigma v}_{DT}/\overline{\sigma v}_{DD} \approx 100$$

and

$$E_{DT}/E_{DD} = 17.6/3.65 = 4.82 ,$$

$$Q_{DT}/Q_{DD} = 1/2 \times 100 \times 4.82 = 241 .$$

While admittedly crude, this comparison points out in a simple way the quantitative advantage of the DT reaction over the DD reaction.

More precisely, we can compare the Q values of DT and DD at their peaks by assuming that the lifetime is proportional to the square root of kT , the kinetic temperature (2). This is with respect to torroidal confinement. The peak values for DT and DD occur at approximately 40 keV and 1000 keV, respectively. Therefore, for equal ion densities,

$$\begin{aligned} \frac{Q_{DT} \text{ (at 40 keV)}}{Q_{DD} \text{ (at 1000 keV)}} &= \frac{\left(\frac{n_{DT}^2}{2}\right) \overline{\sigma v}_{DT} (kT) E_{DT}}{2 \left(\frac{3}{2} nkT_{DT}\right) 1/\sqrt{kT_{DT}}} \\ &= \frac{\left(\frac{n_D^2}{2}\right) \overline{\sigma v}_{DD} (kT) E_{DD}}{2 \left(\frac{3}{2} nkT_{DD}\right) 1/\sqrt{kT_{DD}}} \\ &= 1/2 \left(\frac{\overline{\sigma v}_{DT}(kT)}{\overline{\sigma v}_{DD}(kT)} \right) \left(\frac{E_{DT}}{E_{DD}} \right) \left(\frac{kT_{DD}}{kT_{DT}} \right) \\ &= 1/2 \left(\frac{8 \times 10^{-16}}{2 \times 10^{-16}} \right) 4.82 \left(\frac{\sqrt{1000}}{\sqrt{40}} \right) = 48.2 . \end{aligned}$$

Though this factor-of-five decrease in the Q_{DT}/Q_{DD} ratio is attractive, it is not free. Since the DD plasma is 25 times hotter than the DT plasma, the DD plasma pressure also is 25 times higher than for DT plasma pressure, making its confinement that much more difficult.

So there is a good indication that a satisfactory DT plasma will be some 50-200 times easier to accomplish than a satisfactory DD plasma. [For mirror confinement (3), $Q_{DT}/Q_{DD} \approx 10$.] If this is true, there is great advantage to using the DT fusion reaction. However, for suitable power production, this requires tritium breeding. Thus a satisfactory solution to the technological problem of tritium breeding is crucial for determining the required level of plasma containment.

Tritium Generation

Fortunately, nature provides us the ingredients to produce tritium. Each DT fusion reaction produces one 14-MeV neutron, and the natural isotopes of lithium can be transmuted by neutron reactions into tritium.

The two tritium breeding reactions are:

Breeding takes place in what is commonly called a blanket, which surrounds the fusing plasma to intercept the fusion neutrons. In addition to breeding tritium, the blanket must be able to extract useful heat energy from the kinetic and potential energy of the 14-MeV fusion neutrons, and it must also be able to remove the heat and maintain the vacuum conditions of the plasma region.

Chief components of the blanket are lithium for tritium breeding, a neutron moderator to slow the 14-MeV neutrons, a coolant, and structure. The blanket surrounds the plasma fusion neutron source, as shown in Fig. 2. Outboard of the blanket are thermal and neutron-gamma shields, which protect the cryogenic superconducting magnet coil from overheating.

The simplest blanket one can conceive consists of only lithium, which serves as the fertile fuel, the neutron moderator, and, in the form of liquid metal, as the coolant. Variations add neutron multipliers and/or moderators such as beryllium and carbon. Cooling with helium is another possibility. Another type of blanket employs lithium in the form of a fused salt (lithium-beryllium fluoride) in place of the liquid metal.

Conceptual Blanket Calculations

Can conceptual blankets produce enough tritium nuclei to make up for those that are "burned," plus enough extra to allow for reasonable tritium holdup in the system?

Several people have addressed the tritium breeding question in various conceptual blankets. The general conclusion (4) is yes - that attractive breeding seems possible.

Fig. 2. Cross section of a typical fusion reactor.

Early references to the $\text{Li}^6(n\alpha)\text{T}$ reaction as a means of breeding tritium appear in books by Bishop (5), Glasstone and Lovberg (1), and Rose and Clark (6).

In 1962, Myers *et al.* (7), at Lawrence Livermore Laboratory, performed some one-dimensional diffusion theory calculations on a number of different blankets. The geometry was cylindrical, with an inner radius of 77 cm. The calculations considered various blanket thicknesses and compositions including flibe, which is a mixture of lithium fluoride and beryllium fluoride. The eutectic flibe composition considered by the team was 48 mole % lithium fluoride.

Table 1 summarizes Myers' results. Attractive tritium breeding ratios are indicated. In fact, only the all lithium-6 blanket failed to provide the required tritium

Table 1. Tritium breeding calculations by Myers *et al.*

Material*	Total radial thickness of blanket - cm					
	9	21	39	60	84	96
Eutectic flibe with water layer						
K	0.295	0.649	1.066	1.332	1.433	1.448
K ₇	0.073	0.135	0.175	0.187	0.190	0.190
Eutectic flibe						
K	0.114	0.496	1.133	1.532	1.641	1.662
K ₇	0.092	0.174	0.225	0.241	0.245	0.246
Li ⁶ eutectic flibe						
K	0.129	0.604	1.116	1.361	1.440	1.451
30 mole % LiF flibe						
K	0.066		1.036	1.465	1.645	1.679
K ₇	0.051		0.132	0.144	0.148	
15 mole % LiF flibe						
K	0.033	0.303	0.946	1.418	1.638	1.682
K ₇	0.024	0.046	0.063	0.070	0.072	0.072
Lithium metal						
K	0.264	0.578	0.934	1.232	1.453	1.528
K ₇	0.254	0.503	0.668	0.722	0.736	0.737
Li ⁶ metal						
K	0.065	0.298	0.632	0.853	0.955	0.976
LiNO ₃						
K	0.091	0.222	0.652	1.013	1.238	1.298
K ₇						
LiNO ₂						
K	0.114	0.326	0.742	1.104	1.313	1.365
K ₇	0.109	0.239		0.405	0.424	0.428

*K - Total tritium atoms produced per incident neutron

K₇ - Tritium atoms produced by Li⁷(n,Tn) He⁴ per incident neutron

Table 2. Tritium breeding calculations by Impink.

Material (first wall)	Coolant channel	Primary attenuator	Tritium per neutron
<u>Variation in first wall</u>			
None	Std	Std	1.0717
Ni (1 cm)	Std	Std	1.0042
Mo (1 cm)	Std	Std	1.1414
Mo (2 cm) (std)	Std	Std	1.1679
<u>Variation in Li^6 isotopic action</u>			
Mo (2 cm)	Nat	Nat	1.1499
Mo (2 cm)	0.20	Nat	1.1806
Mo (2 cm)	0.50	0.50	1.2163
<u>Additional Be in primary attenuator</u>			
Mo (2 cm)	Std	None	1.1499
Mo (2 cm)	Std	+5 cm Be	1.5526
Mo (2 cm)	Std	+9 cm BeO	1.4942
Ni (1 cm)	Std	+10 cm Be	1.5328

breeding. Also note that the $\text{Li}^7(n,n'\alpha)\text{T}$ reaction can be an important part of the total tritium breeding. In the case of the natural lithium blanket (lithium metal), the lithium-7 reactions account for nearly half the tritium.

One major drawback of Myers' calculations is that the fluorine and nitrogen cross sections were unavailable, so oxygen cross sections were used in their place.

In 1965, Impink (8) studied the effect on tritium breeding of several blanket variables. His standard blanket was made up of a 2-cm molybdenum first wall, a 6.25-cm coolant channel containing flibe, and a 56-cm primary attenuator containing 79 v/o flibe plus 21 v/o graphite. His flibe was 66 mole % lithium fluoride and 34 mole % beryllium fluoride. Some of Impink's results are listed in Table 2.

In another 1965 paper, Bell (9), of Los Alamos Scientific Laboratory, described tritium breeding calculations for a number of different blankets. Calculations were

Table 3. Tritium breeding calculations by Bell.

Problem	R_1 -cm	M_1	Geometry*		R_3	Tritium nuclei production**				
			R_2 -cm	M_2		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 ₂ C + 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	177.01			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		113.3	0.81	0.39	0.36	1.20	1.56
TU4A	10	Mo	13.3		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	1.701	0.12	0.09	1.82	1.91
TU6A	10	Cu	13.3	0.5 Be ₂ C + 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 Flibe	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

* R = radii in cm; M = material.

** Based on one 14-MeV incident neutron. T_6 and $T_7 = \text{Li}^6(n,\alpha)\text{T}$ and $\text{Li}^7(n,n'\alpha)\text{T}$ reactions; L = neutron leakage; $T^+ = T + L$.

made by a discrete-ordinates neutron transport code with 25 energy groups and the S_4 approximation. The blanket geometry in each case was an infinitely long cylindrical annulus.

Table 3 gives Bell's blanket descriptions and the resulting breeding calculations. The 14-MeV neutron source is inside R_1 and material M_2 is between R_2 and R_3 , where R refers to the radii in centimeters. The quantities T_6 and T_7 are the tritium nuclei produced per incident 14-MeV neutron by the $Li^6(n,\alpha)T$ and $Li^7(n,n'\alpha)T$ reactions, respectively. The total tritium produced (T) is the sum of T_6 and T_7 . The quantity L is the neutron leakage, and T^+ is the sum of T and L.

Bell's infinite medium results indicate that lithium, beryllium plus a little lithium-6, and beryllium carbide plus a little lithium-6 are attractive tritium breeding materials. Flibe is marginal. The major problem with flibe appears to be inelastic scattering by fluorine, which can degrade the fast neutron's energy below the $Li^7(n,n'\alpha)T$ and $Be(n,2n)$ thresholds. The fluorine cross sections are the most uncertain of the lot and should be suspect.

Bell's results also show that pure lithium blankets can offer attractive breeding but need to be thicker than blankets containing beryllium. Problems TU and TU7A, in Table 3, show that flibe plus beryllium can also give attractive breeding.

Up to this point, we have dealt with idealized blanket studies. These are very useful in that they indicate that the necessary tritium breeding looks possible and show what directions one might go to come up with real blanket designs.

In 1969, I examined the effect of varying the amount of structure in a lithium blanket (10). Some results are displayed in Fig. 3. The blanket was a sphere 2 m thick, with the inner and outer radii 1 m and 3 m, respectively. By calculation, more than 95 percent of the tritium breeding occurred in the first 1.3 m. These calculations were run with a 66-group Monte Carlo code using 2500 source neutrons each.

The tritium breeding ratio drops to 1.0 at a niobium fraction of about 20 percent. A structural fraction of

Fig. 3. Tritium and energy generation per 14.1-MeV source neutron versus niobium volume fraction (first wall 0.1-cm niobium).

20 percent is higher than considered necessary. Evidently blankets containing only lithium and required structure can meet tritium breeding requirements.

At the same time, I examined the effect of lithium-6 enrichment in a similar blanket with 5 v/o of niobium. These results are shown in Fig. 4. The calculations employed a one-dimensional discrete-ordinates code.

From these results, it seems that lithium-6 enrichment of natural lithium is not worthwhile. In fact, the lithium can be depleted in lithium-6 while still giving the respectable tritium breeding ratio.

In 1969, Steiner (11) compared the nuclear behavior of two designs differing only in their vacuum wall coolants. One design employed lithium, the other lithium-beryllium fluoride (flibe). Calculations were performed by a one-dimensional discrete-ordinates transport code employing 100 neutron-energy groups and 21 gamma-energy groups.

Fig. 4. Tritium breeding versus Li^6/Li fraction in 95 v/o lithium plus 5 v/o niobium homogeneous blanket. The terms T_6 and T_7 refer to the $\text{Li}^6(n,\alpha)\text{T}$ and $\text{Li}^7(n,\alpha)\text{T}$ reactions, respectively.

Steiner's blanket geometry was a slab with perfect reflection at the inner surface. The blanket configuration included a 0.5-cm niobium vacuum wall, 3 cm of vacuum wall coolant (lithium in design 1, flibe in design 2), a 0.5-cm niobium second wall, 60 cm of lithium coolant diluted by a 6 percent niobium structure, 30 cm of graphite moderator, and 6 cm of lithium coolant diluted by a 6 percent niobium structure. The total blanket thickness was 100 cm.

As shown in Table 4, the tritium breeding in Steiner's blanket was 1.35 and 1.22 for designs 1 and 2.

In 1970, we performed an integrated heat transfer and neutronic blanket study (12) for a high power density DT reactor based on the ASTRON confinement scheme. The blanket makeup and results are listed in Table 5.

Table 4. Tritium breeding calculations by Steiner.

Design summary				
Region	Description	Thickness- (cm)	Material*	
			Design 1	Design 2
1	Vacuum wall	0.5	Nb	Nb
2	Vacuum wall coolant	3.0	Li	Flibe
3	Second wall	0.5	Nb	Nb
4	Coolant plus structure	60.0	94% Li, 6% Nb	94% Li, 6% Nb
5	Moderator	30.0	Graphite	Graphite
6	Coolant plus structure	6.0	94% Li, 6% Nb	94% Li, 6% Nb
Breeding calculations**				
Design	T_6	T_7	L	T
1	0.89	0.46	0.023	1.35
2	0.86	0.36	0.020	1.22

* All percentages are volume percentages.

** Based on one 14-MeV neutron incident on vacuum wall.
 T_6 = tritium breeding ratio in Li⁶; T_7 = tritium breeding ratio in Li⁷; L = neutron leakage; $T = T_6 + T_7$.

Blanket Chemistry

Adequate tritium breeding appears practical; and blankets in which lithium (or lithium plus carbon and/or beryllium) performs the multiple functions of fertile material, coolant, and moderator are neutronically superior to the other conceptual blankets examined.

We have previously observed that blankets, besides breeding tritium and converting the kinetic and potential energy of the remaining neutrons to thermal energy, must be designed also to remove and use both tritium and heat. To drive an attractive heat engine, we assume the blanket loop maximum temperature to be ~1100°K.

Table 5. Tritium breeding and energy generation calculations for ASTRON.

Blanket geometry and materials					
Subzone	Inner radius- cm	Thickness cm	Material	Volume fraction- %	
1	350	0.05	Nb	100	
2	350.05	9.95	Li	75	
			Nb	5	
			Void	20	
3	360	10	Li	40	
			Be	35	
			Nb	5	
			Void	20	
4	370	20	(Same as subzone 3)		
5	390	20	Li	20	
			Be	55	
			Nb	5	
			Void	20	
6	410	20	(Same as subzone 5)		
7	430	20	(Same as subzone 5)		
Breeding and energy generation calculations*					
Case	Li ⁶ /Li ratio- %	Tritium production-T/n			Energy generation- MeV/n
		T ⁶ /n	T ⁷ /n	T/n	
1	7.42**	1.30	0.30	1.60	22.6
2	3.0	1.16	0.32	1.48	23.6
3	0.2	0.80	0.34	1.14	25.3

* Per 14.1-MeV neutron versus Li⁶/Li fraction.

** Occurring naturally.

Some of the more obvious chemical questions that arise in this context are materials compatibility, tritium recovery, and tritium containment. In addition to the fundamental question of lithium's compatibility with the other materials in the blanket-heat transfer loop (such as niobium, beryllium, and graphite), we find ourselves addressing at least the following:

- Tritium-lithium chemistry at low tritium concentrations;
- Tritium removal from the blanket loop;
- Tritium containment;
- Containment of radioactive waste products;
- Lithium isotope separation;
- Lithium-insulator compatibility;
- Alternative coolants.

Each of these deserves special consideration.

Tritium-Lithium Chemistry at Low Tritium Concentrations.

We need to understand this chemistry better so that tritium removal methods can be devised and tritium's effect on lithium compatibility examined. For example, heat pipes have been proposed for removing tritium, but the Li-H³ phase diagram must be better defined at the low H³ partial pressures that preliminary investigation (13) indicates are possible ($\approx 10^{-5}$ torr at the heat pipe wall).

Tritium Removal from Blanket Loop. We must find practical methods to collect the tritium bred in the blanket at low tritium inventories. The tritium inventory must be kept as low as possible, because the lower the inventory, the easier it will be neutronically (lower breeding ratio) to reach an attractively low tritium "doubling time." By "doubling time," we mean the length of time needed to produce enough excess tritium to start up a second, like plant.

For example, for a 2000 MW_{th} (P_{th}) DT plant with a tritium breeding ratio (BR) of 1.10 and an energy release per fusion (E_f) of 25 MeV, a doubling time (t_d) of 1 y or less requires that the total plant tritium inventory (I_t) be kept below

$$I_t = \frac{t_d \times P_{th} \times (BR - 1.0)}{E_f}$$

$$I_t = \frac{3.15 \times 10^7 \text{ sec} \times 2 \times 10^9 \text{ W} \times 6.25 \times 10^{12} \frac{\text{MeV}}{\text{W-sec}} \times (1.1-1.0)}{25 \text{ MeV/Triton}}$$

$$= 1.57 \times 10^{27} \text{ tritium atoms}$$

$$= 7.82 \times 10^3 \text{ g.}$$

Assuming that most of this tritium is in the blanket loop and that the loop contains $\sim 5 \times 10^7$ g of lithium (based on a neutron wall loading of $\sim 10 \text{ MW/m}^2$), the tritium concentration in the lithium must be kept below ~ 100 ppm by weight. Note that this is an *upper limit* based only on a crude doubling-time analysis for a specific case and that the lowest possible tritium inventory is always best. The practical question of how much lower must be answered.

Tritium Containment. Effective methods must be devised for tritium handling and containment. Leakage of tritium to the environment must be kept very low. For example, it has been estimated (14) that to keep the dose rate to individual members of the public below the recommended rate of 0.5 rem/y, the plant's leak rate must be kept below 5×10^5 Ci (about 50 g/y). This is ~ 0.1 percent of the "burn" rate of our 2000 MW_{th} plant or ~ 0.005 percent of the tritium circulation rate through the reactor, assuming a 5 percent "burn" probability per pass.

Waste Containment. Though the fusion fuel cycles themselves produce no radioactive waste products, the fusion neutrons do. The amount and type of activated material produced depends, of course, on the blanket composition. Luckily, the only required blanket material, lithium, produces no unwanted activation products. So radioactive waste-product management problems can be considered in determining what materials go into blankets.

In our conceptual lithium blanket, the only other material required is structure. Niobium is a likely candidate, and I have estimated (15) the radioactive inventory of a lithium-niobium blanket for a 2000-MW DT reactor to be

<u>Material</u>	<u>Half-life</u>	<u>Inventory (Ci)</u>
Li ⁸	0.0855 sec	3.27×10^6
Nb ⁹²	10.1 d	2.0×10^9
Nb ⁹⁴	2×10^4 y	1.13×10^4
Nb ^{94*}	6.6 m	3.43×10^9

Inventory calculations for these materials assume that those with a short half-life are in equilibrium and that the long half-life niobium-94 isomer has been building up for 1 y.

The problems of containment and storage of the blanket activation products should be relatively simple but examined, of course.

Lithium Isotope Separation. Most present conceptual blankets advantageously employ the lithium depleted in lithium-6 (7.56 percent lithium-6 is natural), so lithium-6 enrichment is not necessary.

Lithium-Insulator Compatibility. As suggested, lithium appears to be an ideal multipurpose blanket material for meeting both neutronic and heat transfer requirements. Unfortunately, there is a problem associated with using lithium as a coolant: the magnetohydrodynamic (MHD) effect associated with moving an electrically conducting fluid in the presence of a magnetic field.

Carlson and Hoffman (16) have shown analytically that the lithium pressure drop in a representative DT fusion reactor blanket can range over 6 orders of magnitude (from <0.1 psi to >100,000 psi), depending on the path and pipe wall conductance.

Figure 5 shows how the flow alignment and wall conductance effect the pressure drop. The term C , plotted against the change in total pressure, is the ratio of pipe wall conductance to lithium conductance. A standard conducting wall pipe such as niobium has a $C = 0.1$, and $C = 0$ represents a nonconduction or perfectly insulated pipe.

Curves A1 through A5b represent different flow paths into, through, and out of a simple mirror field with the centerline magnetic field increasing from 5 tesla at the midplane to 10 tesla at the mirrors. The mirrors are 8 m apart and, for purposes of this analysis, flow is started and ended 4 m outside the mirrors to give a total effective flow length of 16 m. The system is sized to produce about 2000 MW thermal.

Curve A0 shows the pressure drop for a straight path with no magnetic effects. Curves A1 - A5b show pressure drops with the magnetic field on. A1 is for a perfectly

Fig. 5. Total pressure drop due to magnetohydrodynamic effects. The term C is the ratio of pipe wall conductance to lithium conductance.

aligned path. It is lower than A0 because the magnetic field strength is sufficient to suppress all turbulence and laminarize the flow.

Paths A2a, A2b, and A2c are for paths with misalignment angles of about 0.06, 0.6, and 6 deg, respectively. Path A4 approximately follows a field line by using several straight line segments. Path A3 is a single straight pipe.

Paths A5a and A5b are circumferential at the midplane and at the coil location, where the transverse magnetic fields are 3.7 and 14.7 tesla, respectively.

The results show that pressure drop is indeed very sensitive to flow orientation and wall conductance and can be unacceptably high if care is not exercised. The allowable pressure drop is on the order of 100 psi, which would require pumping power of about 8 MW, or about 1 percent of the net output.

As can be seen, the effect of going from a conducting wall pipe ($C = 0.1$) to an insulated wall pipe ($C = 0$) depends on the flow path. For the straight pipe case (A3), the effect reduces an unacceptable pressure drop of ≈ 5000 psi to a minimal 20 psi. To reduce pressure drop in this case to an acceptable 100 psi would require a C of ≈ 0.001 , which requires a reduction of the effective pipe wall conductance by a factor of 100.

There are two conceivable ways to achieve this reduction. The pipe itself could be made out of a semiconductor or insulator. Alternatively, the inside of a metal pipe could be coated with an insulator. General considerations for the selection of suitable materials include at least electrical conductance, compatibility with lithium, coating and wall compatibility, neutron and gamma radiation effects, effect on tritium breeding, and cost.

Werner (17) has suggested the following potential materials: Sm_2O_3 , ThO_2 , TiC , ZrC , Y_2O_3 , and $\text{ThO}_2\text{-Y}_2\text{O}_3$. A major problem is finding a suitable material that is compatible with hot lithium. For example, Y_2O_3 looks promising but data are limited and an analytical/experimental program is needed.

Alternative Coolants. If the lithium MHD problem cannot be circumvented by suitable wall insulation and/or pipe orientation, then the fused salt, lithium-beryllium fluoride, is a possible alternative. The chemistry of a flibe system must be studied, bearing in mind that lithium and beryllium are being transmuted to tritium and helium. Also, while the electrical conductivity of flibe is low enough to negate the MHD pumping problem, voltages will be generated, and electrolysis of the flibe might take place. Both effects would result in free fluorine, the effect of which must be considered.

Helium is another possible coolant and, at first glance, poses no chemical problems. But it has been pointed out that oxygen, nitrogen, etc., cannot be kept out of the helium, and this could wreak havoc with refractory materials such as niobium. This problem also must be analyzed and, if real, circumvented.

DIRECT ENERGY CONVERSION

Our second major technological area to be discussed is direct energy conversion. In general terms, we must consider what it is and its possible importance. Then we can discuss a conceptual reactor system and results of a preliminary systems study. Finally, we will summarize some of the problems associated with such a system.

General Considerations

Starting again with the primary fusion reactions, we have

It can be seen that fusion reactions produce energetic charged and neutral particles, specifically helium and hydrogen ions and neutrons. The energy partition between ions and neutrons depends on the reaction, varying from 80 percent neutrons for DT to 100 percent charged particles for DHe³. While all the energy can be converted to heat to run Carnot limited thermal cycles, such as a conventional steam plant, to produce electrical power with efficiencies of 40-50 percent, the possibility of converting the charged particle energy directly at close to 100 percent efficiency is an intriguing alternative.

While direct conversion is fundamentally applicable to any plasma confinement concept, it appears most practical with the mirror confinement concept. I will confine my comments to applying direct conversion to magnetic mirrors.

Magnetic mirrors are characterized by their "open endedness." Figure 6 shows a simple mirror field generated by cylindrical coils. Confinement is achieved by increasing the field at the ends, where plasma particles rotating

(a) Plasma particle travel in magnetic field

(b) Magnetic mirror effect

Fig. 6. Simple mirror field generated by cylindrical coils: (a) plasma particles in the magnetic field tend to travel in helices around lines of magnetic force, with electrons (-) and positive ions (+) rotating in opposite directions; (b) magnetic mirror effect is produced by a "magnetic bottle" in which the magnetic field increases at each end; the stronger field bends the path of the approaching particle into ever tighter circles and exerts a force that reflects it away from the ends of the bottle.

around the field lines tend to be reflected. While more complex field geometries such as of the "Baseball" coil configuration (Fig. 7) are more stable, the basic principle of Fig. 6 is the same: low magnetic field in the center with high fields at the ends or mirrors.

The chief loss from mirrors is plasma leakage out the ends. It is this directed leakage plus the mirror's higher optimum plasma temperature that makes mirrors seem more practical for direct conversion.

Richard Post (18), of Lawrence Livermore Laboratory, formally proposed a method for direct conversion of the mirror leakage in September, 1969. Since then, analytical and experimental efforts (19) have been underway to refine and improve Post's original concept.

In simple terms, the Post schema consists of four steps:

- Expansion of plasma leakage.

Fig. 7. Baseball coil and magnetic field geometry.

- Separation of plasma ions and electrons.
- Deceleration and collection of charged particles.
- Conversion to a common potential.

The first three steps are illustrated in Fig. 8.

The first step, expansion, is necessary to convert the rotational motion of the plasma particles leaking out of the mirror to the translational motion needed for collection.

Fig. 8. Direct energy conversion for a mirror fusion reactor.

This is accomplished by expanding the magnetic field in a fan-shaped geometry from the high mirror values of about 150 kG down to about 500 G.

To accomplish the second step, separation of the ions and electrons, the low strength magnetic field is abruptly folded ≈ 180 deg. The light electrons follow the field lines adiabatically, but the heavy ions go nonadiabatic and continue straight ahead into the ion collector region.

The third step, collection, is accomplished by electrostatically decelerating and directing the ions onto collector elements maintained at appropriate potentials. The ions have a large energy spread (from about 300-1500 keV, with 500 keV about average), so high efficiency requires multiple collector elements.

Figure 9 shows the makeup of a conceptual 20-element collector, with some computer-generated trajectories. The circles are focusing electrodes, the dotted ones being high transmittance grids.

The collection step is the most difficult and uncertain of the four, and much analytical and experimental work is needed to determine the practical limits of this collection scheme. Preliminary results are encouraging; it is speculated that collection efficiencies in the 85-95 percent range will be practical.

The fourth step, conversion to a common potential, employs an inverter-rectifier system for which 99 percent efficiency is considered state-of-the-art.

We speculate that the overall efficiency of the four steps that make up direct conversion can be in the 75-90 percent range.

Conceptual Reactor System

Figure 10 is a block diagram of a conceptual reactor system employing mirror plasma confinement and direct conversion. The circled numbers, which identify powers into and/or out of system elements, are defined in Table 7.

Fig. 9. Collector geometry and typical ion trajectories.

Mirror reactors are classically low-Q devices (Q being defined, again, as the ratio of fusion power to net injected power required to maintain the plasma). The Q values are expected to range from 0.2 to 2.0. The lower end of the range, ≈ 0.2 to 0.4, is for the DHe³ fuel cycle; the upper end, ≈ 1.0 to 2.0, is for the DT fuel cycle (3,20-23).

To maintain the plasma, energetic fuel atoms are injected into the plasma by the injection system. The mirror leakage is sent through the elements of the direct converter as already described. The fusion neutrons produce thermal power in the blanket (as described earlier), which is used to drive a thermal cycle. Thermal power is also produced by the inefficiencies of the ion handling elements and can also be used in a thermal cycle.

Fig. 10. Mirror reactor system block diagram.

An analysis of this reactor system (24) was performed to determine, in part, which combination of plasma data (principally Q), component efficiency, and cost assumptions produces an attractive power system.

Sample results of system efficiency and cost versus the plasma containment parameter Q are shown in Figs. 11 and 12. These results vividly display the importance of direct conversion for both the DT and DHe³ cycles. Direct conversion reduces the required Q values for a viable DT system by about a factor of 3, from about 1.5 down to about 0.5. And since the calculated Q values for stable DT mirror plasmas are at least 1.0, a DT mirror reactor with direct conversion is an attractive prospect. On the other hand, if the Q for the DHe³ cycle could be increased to near 1.0, the overall efficiency of a DHe³ mirror reactor with direct conversion would

Fig. 11. Effect of direct conversion on the overall efficiency of a mirror machine (1000-MWe power output, nominal component performance). Q is the ratio of fusion power to trapped injected power.

Fig. 12. Total system cost as a function of Q for DT and DHe³ mirror reactors (1000-MWe power output, nominal component performance). Q is the ratio of fusion power to trapped injected power.

be spectacular, almost 70 percent compared to 45 percent for a DT plant. In terms of thermal effluent, this would mean a reduction of almost a factor of 3. This is an important incentive to find ways to get the Q values for DHe³ up to ≈ 1.0 .

Component performance assumptions for these analyses are fixed midway between pessimism and optimism, in our analysis. Input and output for two specific DT and DHe³ cases are listed in Tables 6, 7, and 8. No claim is made for absolute accuracy of these results, because the component performance and cost data are at best crude and in some cases only an educated guess. But assuming we have been reasonable, the

Table 6. System code input for a specific DT and DHe³ reactor system.

Data	Fuel cycle	
	DHe ³	DT
<u>Plasma Data</u>		
Q - Fusion power/trapped injected power F(1)	0.4	1.0
Fraction of fusion power in charged particle kinetic energy F(2)	0.9	0.2
Fraction of injected power trapped in plasma F(3)	0.975	0.975
Radiation fraction of fusion ions + trapped injection power F(4)	0.02	0.02
Ion fraction of expanded leakage power F(5)	1.0	1.0
Average ion energy in reactor (keV) F(6)	435	420
Average ion energy entering collector (keV) F(7) (ion energies optimized)	594	575
Beta - fuel ion pressure/magnetic pressure	0.8	0.8
Fuel - 1.0 = DT, 2.0 = DHe ³	2.0	1.0
<u>Component Efficiency Data</u>		
Ion expansion E(1)	0.94	0.94
Electron expansion E(2)	0.94	0.94
Ion separation and collection E(3)	0.90	0.90
Electron collector E(4)	0.90	0.90
Inverter-rectifier (I-R) E(5)	0.99	0.99
Fusion neutron blanket energy multiplication E(6)	2.00	1.30
Neutral beam collector E(7)	0.95	0.95
Injector system E(8)	0.90	0.90
Thermal conversion of blanket heat E(9)	0.45	0.45
Thermal conversion of collector and injector heat E(10)	0.45	0.45
Auxiliary power fraction of fusion power E(11)	0.01	0.01
Desired output power level (MW _{net})	1000	1000
<u>Cost Factor Data</u>		
Blanket and shield thickness (M) CF(1)	1.0	1.5
Cost per unit area of blanket (\$/M ²) CF(2)	1000	25000
Minimum expander radius for 800 keV average ion energy (M) CF(3)	25	25
Mirror leakage area per hole of bare reactor (M ²) CF(4)	3.15	3.15
Expander expansion ratio CF(5)	260	200
Ratio of collector length to ion beam height in collector CF(6)	25	25
Space charge limited collector flux (N=12, E=800 keV) (N/M ²) CF(7)	3	3
Expander/collector structural cost (\$/M ²) CF(8)	600	600
Expander magnet cost (\$/M ²) CF(9)	425	425
Collector element cost (\$/M ²) CF(10)	640	640
Injector system unit cost (\$/MW into system) CF(11)	30	30
Waste heat disposal cost (\$/MW) CF(12)	5	5
Inverter-rectifier cost (\$/MW) CF(13)	3	3
Blanket thermal conversion cost (\$/MW) CF(14)	86	96
Injector-collector thermal conversion cost (\$/MW) CF(15)	96	86

Table 7. System code output for a specific DT reactor system.

RESULTS - SYSTEM ELEMENT POWERS (P(I)) (MW)

MIRROR REACTOR LEAKAGE(1)	1631.581	PLASMA RADIATION (2)	33.298
COLLECTOR LOSS (3)	1533.686	COLLECTOR ELECTRONICS (4)	0.
EXPANDER HEAT (5)	97.895	ION COLL. ELECTRIC (6)	1380.317
ION COLL. HEAT (7)	153.369	ELECTRON COLL. ELECTRIC (8)	0.
ELECTRON COLL. HEAT (9)	6.	I-R ELECTRIC (10)	1366.514
I-R HEAT (11)	13.803	FUSION NEUTRON (12)	1109.919
NEUTRON INDUCED BLANKET (13)	2219.837	UNTRAPPED BEAM ELECTRIC (14)	32.906
UNTRAPPED BEAM THERMAL (15)	2.668	INJECTOR INPUT (16)	1581.081
INJECTOR HEAT (17)	158.108	EXP+COL+ION HEAT (18)	412.040
THERMAL TO ELECTRIC (19)	1299.329	TOTAL ELECTRIC (20)	2598.749
AUXILIARY(ASSUMED) (21)	17.668	NET ELECTRIC OUTPUT(22)	1000.000
FUSION POWER (23)	1387.398	WASTE HEAT (24)	1497.317
NEUTRAL BEAM POWER (25)	1422.973	AUXILIARY(CALCULATED)	16.061
SYSTEM EFFICIENCY	0.400		

COST RESULTS WITH OPTIMIZED EXPANDER COLLECTOR (S/KWE-NET)

REACTOR MAGNET	41.628	BLANKET	2.986
THERMAL CONVERTERS	103.142	EXP-COLL STRUCTURE	13.139
EXPANDER MAGNET	9.796	VACUUM SYSTEM	0.661
COLLECTOR ELEMENTS	16.822	INJECTOR SYSTEM COST	24.730
WASTE HEAT DISPOSAL	7.487	INVERTER-RECTIFIER	4.100
		TOTAL	219.760

SYSTEM GEOMETRY WITH OPTIMIZED EXPANDER-COLLECTOR

PLASMA RADIUS (IN)	3.129	VAC MAG FIELD (KG)	64.889
NUMBER OF LEAKAGE BEAMS	3.029	BEAM HEIGHT (IN)	0.870
EXPANDER RADIUS (IN)	80.649	COLLECTOR DEPTH (IN)	21.756
EXPANDER ANGLE (RADIANS)	4.189		

results are encouraging; they indicate that mirror machines with direct conversion are promising and should be actively investigated.

Problem Areas

The concept and potential of direct conversion are attractive. But what are some of the problems?

The first order of business, of course, is to develop practical configurations, those which will realize high efficiencies with expander-collector structures of reasonable

Table 8. System code output for a specific DHe³ reactor system.

RESULTS - SYSTEM ELEMENT POWERS (P(I)) (MW)

MIRROR REACTOR LEAKAGE(1)	7044.258	PLASMA RADIATION (2)	143.760
COLLECTION IONS (3)	6621.603	COLLECTOR FLECTIONS (4)	0.
EXPANDER HEAT (5)	422.655	ION COLL. ELECTRIC (6)	5959.442
ION COLL. HEAT (7)	662.160	ELECTRON COLL. ELECTRIC (8)	0.
ELECTRON COLL. HEAT (9)	0.	I-M ELECTRIC (10)	5899.848
I-M HEAT (11)	59.594	FUSION NEUTRON (12)	321.852
NEUTRON INDUCED BLANKET (13)	643.703	UNTRAPPED BEAM ELECTRIC (14)	121.228
UNTRAPPED BEAM THERMAL (15)	10.316	INJECTOR INPUT (16)	6113.040
INJECTOR HEAT (17)	611.304	EXP+COL+INJ HEAT (18)	1706.436
THERMAL TO ELECTRIC (19)	1122.255	TOTAL ELECTRIC (20)	7149.330
AUXILIARY(ASSUMED) (21)	36.790	NET ELECTRIC OUTPUT(22)	1888.080
FUSION POWER (23)	2145.677	WASTE HEAT (24)	1467.529
NEUTRAL BEAM POWER (25)	5501.736	AUXILIARY(CALCULATED)	33.274
SYSTEM EFFICIENCY	0.405		

COST RESULTS WITH OPTIMIZED EXPANDER COLLECTOR (S/MWE-NET)

REACTOR MAGNET	84.366	BLANKET	0.674
THERMAL CONVERTERS	96.514	EXP-COLL STRUCTURE	58.655
EXPANDER MAGNET	36.441	VACUUM SYSTEM	2.844
COLLECTOR ELEMENTS	23.062	INJECTOR SYSTEM COST	92.552
WASTE HEAT DISPOSAL	7.538	INVERTER-RECTIFIER	17.700
		TOTAL	420.142

SYSTEM GEOMETRY WITH OPTIMIZED EXPANDER-COLLECTOR

PLASMA RADIUS (M)	3.959	VAC MAG FIELD (KG)	70.080
NUMBER OF LEAKAGE BEAMS	3.880	BEAM HEIGHT (M)	0.548
EXPANDER RADIUS (M)	282.830	COLLECTOR DEPTH (M)	13.787
EXPANDER ANGLE (RADIAN)	4.189		

size at the high circulating power levels dictated by low Q values. Important specific problems include:

- Ion holdup and damage;
- Vacuum pumping;
- Hydrogen and helium isotopic separation;
- Tritium storage;
- Insulator breakdown.

We will address these problems one by one.

Ion Holdup and Damage. In addition to efficient power handling, direct converters must also allow for efficient recirculation of the fusion fuel material. The expander-collector has large surface areas and handles large currents. The collector (and other surfaces) intercept these large currents of energetic ions. Ion holdup in the surface, and consequent damage, thus could prove important.

Table 9 lists the approximate level of ion fluxes to be expected, based on the conceptual 1000 MW_e net DT and DHe³ mirror reactor systems described earlier.

To design the collectors and other surfaces of a direct converter properly, we must first understand the rate-controlling processes that govern the holdup and release of hydrogen and helium. For the two species, the processes are expected to be materially different. It must also be determined whether the processes for the two species can be decoupled and separate analytical expressions derived for each.

The helium holdup and damage question appears sizeable, both in our ability to analyze the problem and in the probable results.

Since helium is insoluble, bubbles could build up until the pressure physically ruptures the material, causing layers to spall. In some limited experiments, spallation of micron-thick material was observed at helium doses (300 keV) of $\sim 10^{18}$ ions/cm² (25,26).

Most of the plasma ions striking any expander structure will be in a wide energy band (~ 400 - 2000 keV, with 600 keV the average), so much of the helium could be trapped in the spalled material. In the important collector area, the same spallation should occur. But since the ion energies will be much lower (~ 0 - 150 keV, with 50 keV the average), the spall line should be closer to the surface, possibly resulting in a lower fraction of helium holdup.

Perhaps the only way to get a reasonable idea of how much helium is trapped is to run experiments.

Unlike helium, hydrogen is soluble in most metals and, at the temperatures of interest, should readily diffuse. If the simple diffusion model is appropriate, the upper limit to hydrogen transport by diffusion can be determined by the hydrogen ion's range and the maximum concentration permitted by the hydrogen solubility limit. If the hydrogen source

Table 9. Ion fluxes expected on direct converter surfaces for DT and DHe³ reactors (1000 MWe net).^a

Element (comp°C)	Area- cm ²	Flux-ions/sec-cm ²						Ion energy range-keV	Average thermal flux-W/cm ²	
		ϕ_H	ϕ_D	ϕ_T	ϕ_{He^3}	ϕ_{He^4}				
D-T Reactor										
Expendor columns (-700°C)	8×10^4	1.5×10^{13}	1×10^{13}	1×10^{13}	1.5×10^{13}	6×10^{13}	400 - 2000 (Avg = 570)	200		
Collector surfaces (-700°C)	7.2×10^8	1.7×10^{11}	1.2×10^{13}	1.2×10^{13}	1.7×10^{11}	6.7×10^{11}	10 - 100 (Avg = 50)	0.2		
Expendor wall (<100°C)	2.7×10^8	4×10^7	3×10^9	3×10^9	4×10^7	2×10^8	400 - 2000 (Avg = 570)	7×10^{-3}		
D-He ³ Reactor										
Expendor columns (-700°C)	1.5×10^3	3.4×10^{13}	1.8×10^{13}	1.7×10^{13}	2×10^{14}	1.7×10^{13}	400 - 2000 (Avg = 600)	200		
Collector elements (-700°C)	1.1×10^9	1.1×10^{12}	6×10^{13}	5.5×10^8	6.7×10^{12}	5.5×10^8	10 - 100 (Avg = 60)	0.9		
Expendor wall (<100°C)	1.7×10^9	7×10^7	4×10^6	4×10^6	4×10^8	4×10^6	400 - 2000 (Avg = 600)	6×10^{-4}		

^a Based on the following assumptions:

1. For DT reactor:
  - a. $I_D = I_T$ (I = mirror leakage current)
  - b. DT reaction rate = DD reaction rate
  - c. $Q = 1.0$

2. For DHe³ reactor:

- a. $I_D = 0.9 I$ (total); $I_{He^3} = 0.1 I$
- b. DHe³ reaction rate = 0.5 DD reaction rate
- c. $Q = 0.4$
3. Expendable wall current = 10^{-4} A

exceeds this transport limit, the hydrogen could form bubbles and act much like helium. Fortunately, the simple diffusion model predicts hydrogen transport rates in stainless steel to be 3 to 5 orders of magnitude above expected ion fluxes.

However, the presence of helium in the metal complicates the hydrogen transport question. Helium in the lattice might reduce the diffusion coefficient, and helium bubbles might reduce the area and increase the path length for diffusion. Also, damage caused by the passage of energetic ions might grossly affect the hydrogen diffusion rate.

It appears that experiments will be needed to determine how much these effects will reduce the rate at which hydrogen can diffuse out of metal surfaces.

Surface damage due to sputtering and spallation could be especially detrimental to the voltage loading capabilities of elements in regions of high electric fields, such as grids. Electrical breakdown is, of course, intolerable.

In an attempt to circumvent the holdup and damage problems, it has been suggested that the surfaces could be coated with a low-vapor-pressure liquid metal such as gallium or tin, which should allow the helium and hydrogen to escape without damaging the surface. This may be an acceptable solution but its potential and problems, such as liquid metal wetting, vaporization, and sputtering, must be evaluated.

Vacuum Pumping. To minimize ionization and charge exchange losses in the expander-collector, pressures on the order of 10^{-5} torr are required. Efficient, practical pumping systems must be devised to handle the high pumping speeds needed. For example, the rate of hydrogen and helium circulation through the conceptual 1000 MW_e system described earlier is on the order of 10^{22} and 10^{23} atoms/sec, respectively.

Hydrogen and Helium Isotopic Separation. Both the DT and DHe³ fuel cycles produce isotopes of hydrogen and helium that must be removed from the circulating fuel. If allowed to build up, they would poison the plasma. Therefore, in the DT case, hydrogen must be removed from the deuterium-tritium fuel. And in the DHe³ case, hydrogen and tritium and helium-4 must be removed from the DHe³ fuel.

Methods need to be developed to perform these isotopic separations in a way that will minimize the holdup and loss of tritium of helium-3, at reasonable capital and operating costs.

Tritium Storage. As suggested earlier, at a Q value near 1.0 or higher, the DHe³ cycle is most attractive. It has the potential of reducing waste heat generation by a factor of about 3. And its neutron-induced radioactive waste problem can be eliminated effectively by proper choice of the blanket materials.

The only remaining possible pollutant is the tritium produced by the DD reactions necessary for helium-3 production. Fortunately, tritium has a relatively short half-life (12.3 y), and it decays to helium-3. This helium-3 could then be used to help fuel the primary DHe³ reaction, thus reducing the DD-to-DHe³ reaction ratio from about 2.2 to about 1.1.

To realize this reduction in the DD reaction ratio will require a safe, economic method to store the tritium - some method that effectively contains all the tritium while allowing the helium-3 to be removed and used.

Insulator Breakdown. As presently conceived, heat generated by inefficiencies in the direct converter will be removed by flowing helium and used in a thermal cycle. The helium will have to flow through insulators needed to isolate the collector voltages, which will range upwards to about 2 million volts. What effect the flowing helium will have on the insulator's voltage-holding capabilities is not known.

SUMMARY

We have considered the prospects and promises of tritium breeding and direct energy conversion. What can we say in conclusion?

First, that to make the attractive DT fusion cycle a viable power source, we need a method of tritium generation. The only conceivable, practical way to generate it is to use the DT fusion neutrons to breed tritium from lithium. Tritium breeding requires lithium in some form to surround

or blanket the DT plasma as closely as possible, so that the DT neutrons can be utilized efficiently.

Second, we observe that neutronic analyses to date on simple conceptual blankets have given attractive tritium breeding ratios. These analyses suggest the ratios greater than 1.0 should be possible in practical blankets.

We have noted, also, that practical blanket design will require the satisfactory solution of many problems imposed by the tritium breeding requirement, among these:

- Compatibility of the lithium (or materials containing lithium) with other blanket materials such as structure, neutron multipliers, moderators, and coolants, at temperatures up to $\sim 1100^{\circ}\text{K}$;
- Tritium's effect on this compatibility;
- Tritium removal from the blanket loop, by methods which limit the tritium holdup to acceptably low levels;
- Tritium containment in the reactor and blanket loops and in storage;
- Containment and storage of radioactive waste products;
- Lithium enrichment technology (if needed for certain conceptual blankets);
- Electrical insulation of metal tube walls from flowing hot lithium.

Regarding the latter, if semiconductor or insulator materials can be found to provide the electrical insulation required, the magnetohydrodynamic pumping problems associated with flowing liquid metals in a strong magnetic field can be substantially reduced.

Turning to direct energy conversion, we can summarize, first, that the development and application of an efficient and practical direct conversion system will substantially lower the plasma confinement requirements of low Q confinement systems.

Second, for the long term, the low thermal and radioactive waste potential of the DHe³ mirror reactor with direct conversion is very appealing. The plasma physics and reactor technology required by such systems should be pursued actively.

Again we are faced by many problems (in addition to efficient power handling) that need solution, including:

- Methods for determining the fuel holdup and, if necessary, reducing it to acceptable levels;

- Ways to determine the extent of ion-caused sputtering or spallation of the collector elements, and reduce these effects where necessary;

- Practical methods for separating unwanted hydrogen and helium isotopes from the hydrogen and helium fuel isotopes circulating in the system, methods which will minimize holdup and losses;

- Effective ways to store tritium (DHe³ cycle) that will allow the tritium-decay He³ to be collected for fueling the primary cycle.

Obviously there are many problems and questions about both tritium breeding and direct conversion that have not been specified. Many people in many disciplines are needed to look at such conceptual systems. They can help answer the known questions and, perhaps, uncover and solve problems as yet unknown.

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