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Summary

$Li_{17}Pb_{83}$ is an attractive breeding material due to its high tritium breeding potential, low tritium solubility and relative inertness toward water. In previous designs, $Li_{17}Pb_{83}$ was used both as the breeding material and the coolant. The high tritium partial pressure, caused by the low tritium solubility, resulted in considerable difficulties in tritium confinement. In addition, there are problems associated with corrosion and corrosion product transport, pumping equipment cost and, for a magnetic confinement system, MHD problems. These problems can be alleviated by using static $Li_{17}Pb_{83}$ only as a breeding material. A gas coolant confined in a coolant tube is used as the cooling medium. The tritium concentration in the blanket is allowed to increase and, finally, tritium will diffuse toward the plasma or the coolant stream. The required tritium partial pressure is 10^{-2} torr, which corresponds to a tritium concentration in $Li_{17}Pb_{83}$ of only 6 WPPB. Since the primary heat exchanger now does not see this high tritium partial pressure, tritium confinement becomes much easier. The gas coolant proposed to be used is high pressure steam. The higher density of steam compared to helium will reduce the pumping power required by a factor of 2. Tritium recovery from steam is straightforward if the concentration is high enough. Safety is not a severe problem due to the inertness of $Li_{17}Pb_{83}$.

Introduction

A blanket design concept for a 5500 MWth modular stellarator power reactor, UWTOR-M, is presented here. The materials chosen for the blanket are $Li_{17}Pb_{83}$ for breeding and HT-9 ferritic steel for structure. To avoid pumping a liquid metal through high magnetic fields and complex geometries, high pressure steam is used as the coolant. Although the blanket concept is specially designed to encounter some special problems associated with a stellarator, it is also applicable to other types of D-T fusion reactors. A side view of UWTOR-M is shown in Figure 1.

The stellarator is one of the earliest proposed magnetic confinement concepts, having been invented in the early fifties. It recently gained recognition in the fusion community. This is mainly due to recent encouraging experimental results which show that confinement in a stellarator does not degrade, but in fact improves, when the net plasma current is reduced to a near zero value.[1] These experimental results also verify that with a rotational transform of > 0.3, the stellarator does not experience plasma disruption. In a separate development, recent innovations have shown that the continuous helices can be replaced by a set of discrete twisted coils making the system entirely modular.[2] These new developments, coupled with the inherent advantages of stellarators, namely steady state magnetic fields and burn cycles, natural divertor and low recirculating power fractions, have rekindled interest in stellarators as a possible candidate for a power reactor.

The University of Wisconsin Fusion Engineering Program in conjunction with the Stellarator/Torsatron Lab has addressed some of the key technical issues of stellarator/torsatron power reactors. As a result of these deliberations, a point reactor design called UWTOR-M has evolved.[3] This paper discusses the blanket design concept and in particular the tritium recovery aspect of this blanket design.

Blanket Description

A cross sectional view of the blanket for UWTOR-M has been shown in Figure 1. Figure 2 shows the effect of utilization of the magnetic divertor on the geometry of the blanket. This figure shows the area of blanket coverage if the toroidal shell in the plane of the first wall is mapped out over one field period. The complicated geometry will cause excess MHD problems if liquid metal is going to be circulated as the primary coolant. On the other hand, the limited breeding space available makes solid breeders, even with neutron multipliers, unattractive.

The proposed blanket design for UWTOR-M is to use static $Li_{17}Pb_{83}$ breeding and gas cooling. To best use the low solubility of tritium in $Li_{17}Pb_{83}$, we investigated the possibility of allowing tritium build-up in $Li_{17}Pb_{83}$ until the partial pressure is high enough so that the tritium will diffuse into the primary coolant from which it can be recovered. The tritium diffusion and recovery will be the major topic in this paper.

Hydrogen Solubility in $Li_{17}Pb_{83}$

There exists some controversy over the subject of hydrogen solubility in the lithium lead mixture. Ihle[4] first reported the solubility of deuterium in LiPb at two different temperatures. It appears that the result shows too large a temperature dependence such that extrapolation to lower temperatures is not possible.[5] Larsen et al.[6] proposed the use of the activity of lithium in LiPb as the correction factor to estimate Sievert's constant of D in LiPb from D in pure lithium, or

$$K_s' = a_{Li(LiPb)}\ K_s \qquad (1)$$

in which K_s' is Sievert's constant for D in LiPb
$a_{Li(LiPb)}$ is activity of lithium in LiPb
K_s is Sievert's constant of D in Li.

Recently, Veleckis measured Sievert's constant in $Li_{17}Pb_{83}$ and concludes that:[7]

1. Within the experimental error, in the range from 400 to 600°C, the Sievert's constant for solutions of hydrogen in $Li_{17}Pb_{83}$ was found to be independent of temperature.

2. For the above temperature range, the Sievert's constant of hydrogen in liquid $Li_{17}Pb_{83}$ is given by

$$K_s = (5.1 \pm 1.3)\ (\text{APPM})/[\text{torr}]^{1/2}\ .$$

Plots of the postulated Sievert's constant calculated from Eq. (1) and the experimental date by Ihle, Veleckis and the data for pure lithium and lead are shown in Figure 3. It appears that the range and temperature dependency of references 6 and 7 are reasonably consistent.

The Sievert's constant for tritium in $Li_{17}Pb_{83}$ is estimated to be ~ 2.9 APPM/[Torr]$^{1/2}$.

$Li_{17}Pb_{83}$ has been used both as the breeding material and the primary coolant.[8] The breeding material process rate for tritium recovery can be estimated by

Breeding material process rate =

$$\frac{\text{tritium breeding rate}}{\text{(T conc. in breeding matl.) x (frac. recovery)}} .$$

To maintain a reasonable breeding material process rate, the tritium concentration in the breeding material cannot be too small. A reasonably small tritium concentration in the breeding material, due to the small Sievert's constant, will result in a very large tritium partial pressure over $Li_{17}Pb_{83}$. This large tritium partial pressure makes the tritium confinement problem in the primary heat exchanger particularly difficult. Therefore, the low tritium solubility, even though it results in low tritium inventory in the blanket, causes excessive problems in tritium confinement.

Tritium Recovery Options

The most frequently suggested blanket breeding/coolant material combinations are of two basic types:

1. Gas cooled solid lithium compound breeding materials, with a helium purge gas for tritium recovery. In such a design, the tritium diffusion mechanism is not clear and the combination of rate limiting steps and radiation effects may lead to unacceptably large blanket tritium inventory. The "STARFIRE"[9] design, for example, estimated a blanket tritium inventory of 7.8 ~ 380 kg.

2. Circulating liquid lithium[10] or $Li_{17}Pb_{83}$.[8] The problems associated with such a design include corrosion, corrosion product transport, MHD and tritium confinement problems.

In the UWTOR-M blanket, we propose using a static volume of $Li_{17}Pb_{83}$ which is cooled either by helium or by high pressure steam. High pressure steam is attractive due to its larger volumetric specific heat in comparison to helium at the same pressure so that pumping power required can be reduced by roughly half. A comparison between helium and steam on heat transfer and pumping power requirements is shown in Table 1. The low tritium solubility, and consequently the high tritium partial pressure, will cause the tritium to diffuse through the HT-9 coolant tube walls

Table 1 Comparison of Steam and Helium Performance

(P = 50 atm, T = 500°C)

	He	Steam
ρ, g/cm^3	3×10^{-3}	1.35×10^{-2}
C_p, J/g-°C	5.19	2.13
K, W/cm-°C	1.56×10^{-3}	2.25×1^{-4}
$V\Delta P$ (relative)	12.8	5.68
h (relative)	1.28	1.43

to the steam coolant. The tritium diffusing into the steam will be oxidized to the form HTO. This coolant steam is circulated to a steam generator, but does not itself drive a turbine. Since T is tied up in HTO, the tritium permeation problem through the steam generator wall is very small. The tritium concentration in the primary steam can be allowed to reach a high level because total water inventory in the primary coolant is only 2000 kg. Conventional techniques for recovering deuterium from water can then be used to recover the tritium. A summary of the tritium permeation and inventory parameters is given in Table 2.

Table 2 . Blanket Tritium Inventory Parameters

Fusion power	5500 MW
Tritium breeding rate	10^{-2} g/sec
Coolant tube area in the blanket	5×10^4 m^2
Tritium permeability through HT-9	$.6 \frac{\text{mole-mm}}{\text{day-m}^2\text{-atm}^{1/2}}$
Sievert's constant	2.9 APPM/[Torr]$^{1/2}$
Tritium partial pressure	1.7×10^{-2} Torr
Tritium concentration in $Li_{17}Pb_{83}$	.38 APPM
$Li_{17}Pb_{83}$ inventory	10^7 kg
Tritium inventory in $Li_{17}Pb_{83}$	66 g
Water inventory in primary steam circuit	2×10^3 kg
Tritium inventory in primary steam circuit	100 g
Tritium dissolved in blanket structure	4 g
Total blanket tritium inventory	164 g

The blanket tritium inventory is only 70 g for a 5500 MW unit. This inventory can be allowed to increase by a factor of 10. Therefore, the tritium permeation requirement can be reduced by a factor of 10. This will account for a safety margin for non-ideal situations, such as the formation of an oxide coating.

Some Problems Associated With Steam Coolant

1. Tritium recovery. One of the main objections against using water as the coolant in a DT fusion reactor is the problems associated with tritium recovery from water. In a steam cooled system, the water inventory in the primary circuit is only 2000 kg. With such a small water inventory, the tritium concentration can then be allowed to build up to a reasonably high level. This should considerably simplify the tritium recovery problem.

One possible tritium recovery process is shown in Fig. 4. A counter current liquid phase catalytic exchange process is used, in which a fraction of tritium in water is transferred to a hydrogen stream. The tritium in the hydrogen can then be recovered from a standard fractional distillation process. The recovery fraction,(1-x), from the catalytic exchange process can be as low as 10%.

2. Safety. The potential chemical reactivity between water and breeding material is of major concern. However, by using $Li_{17}Pb_{83}$ as the breeding material and steam as coolant, the safety problem is not too critical. The blanket stored energy consists

of thermal and chemical energy. Table 3 compares the stored enrgy in UWTOR-M and STARFIRE.[9] The stored energy per GW is very comparable.

Table 3 Energy Stored in Blanket

	$Li_{17}Pb_{83}$/Steam	STARFIRE
Thermal	5.5×10^{10} J/GW	1.4×10^{11} J/GW
Chemical	4.9×10^{9} J/GW	?

3. Effect of Oxide Coating on Tritium Diffusion. For a steam cooled blanket, an oxide coating will develop on the inside surface of the coolant tube. The oxide coating may provide a tritium diffusion barrier as high as 500 times to that of the basic metal tube.[11] This will prevent the tritium from diffusion into the coolant. However, a hydrogen overpressure may be added to the steam circuit so that the steam will be in a reducing environment. If 1 Torr of hydrogen is added to the steam, the oxygen partial pressure will be reduced from 10^{-5} Torr to 10^{-19} Torr. The effect of reduction of oxygen partial pressure on the formation of the oxide coating has to be investigated.

Conclusions

A blanket design using static $Li_{17}Pb_{83}$ as breeding material and a high pressure gas as the coolant is reported. Such a blanket concept reduces the problems associated with a liquid metal cooled blanket such as corrosion, corrosion product transport, MHD and tritium confinement. A high pressure steam is used as the coolant to reduce the pumping power requirement. The tritium bred in the blanket is allowed to diffuse toward the coolant from which it can be recovered. The key to the solution of tritium confinement is to combine the tritium into a compound, e.g. HTO, rather than having it stay in T_2 form. The design is specific for a stellarator reactor, UWTOR-M, in encountering the problems asociated with a complicated blanket geometry and a limited available breeding area. A similar approach is applicable, however, to other DT fusion reactors.

The effect of oxide coating formed in a very low oxygen partial pressure on the diffusion of tritium is a key question associated with the blanket concept. If a positive pressure of 1 torr is maintained in the steam coolant stream, the oxygen partial pressure is only 10^{-19} torr. It is anticipated that such a low oxygen pressure will have a small effect on the permeability of tritium of the HT-9 coolant tubing.

Acknowledgement

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References

1. H. Renner, et al., Max Planck Institute, Plasma Physics and Controlled Nuclear Fusion Research (Brussels, 1980).

2. J.L. Shohet, "Ultimate and Modular Torsatrons", B.3.4., International Workshop on Stellarators, Schloss Ringberg, Tegernsee, Bavaria, FRG, July 1980.

3. I.N. Sviatoslavsky, et al., "UWTOR-M, A Stellarator Power Reactor Utilizing Modular Coils", Third IAEA Committee Meeting and Workshop on Fusion Reactor Design and Technology, October 5-16, 1981, Tokyo, Japan.

4. H.R. Ihle, A. Neubert and C.H. Wu, "The Activity of Lithium and the Solubility of Deuterium in Lithium-Lead Alloys", Proc. Tenth Symp. Fusion, Padova, Italy, September 4-9, 1978, Pergamon Press, New York, 639-644 (1979).

5. N.J. Hoffman, A. Darnell, J.A. Blink, "Properties of Lead-Lithium Solutions", Lawrence Livermore Laboratory, CA, October 1980, UCRL-84273.

6. E.M. Larsen, et al., Comments on the Hydrogen Solubility Data for Liquid Lead, Lithium and Lithium-Lead Alloys and Review of a Tritium Solubility Model for Lithium-Lead Alloys, UWFDM-415, University of Wisconsin, Madison, WI, May 1981.

7. E. Veleckis, Argonne National Laboratory, private communication.

8. D.K. Sze, et al., "Thermal and Mechanical Design of WITAMIR-I Blanket", Proceedings of the Fourth Topical Meeting on the Technology of Controlled Nuclear Fusion, October 14-17, 1980, King of Prussia, Pennsylvania.

9. C.C. Baker, et al., "STARFIRE - A Commercial Tokamak Fusion Power Plant Study", ANL/FPP-80-1, September 1980.

10. D.K. Sze, et al., "Thermal and Mechanical Design Considerations for Lithium-Cooled Tokamak Reactor Blankets", 5th Symposium on Engineering Problems of Fusion Research, March 1973.

11. J.T. Bell, et al., "Tritium Permeability of Structural Materials and Surface Effects on Permeation Rates", Proceedings on the Tritium Technology in Fission, Fusion and Isotopic Applications, April 29 - May 1, 1980, Dayton, Ohio.

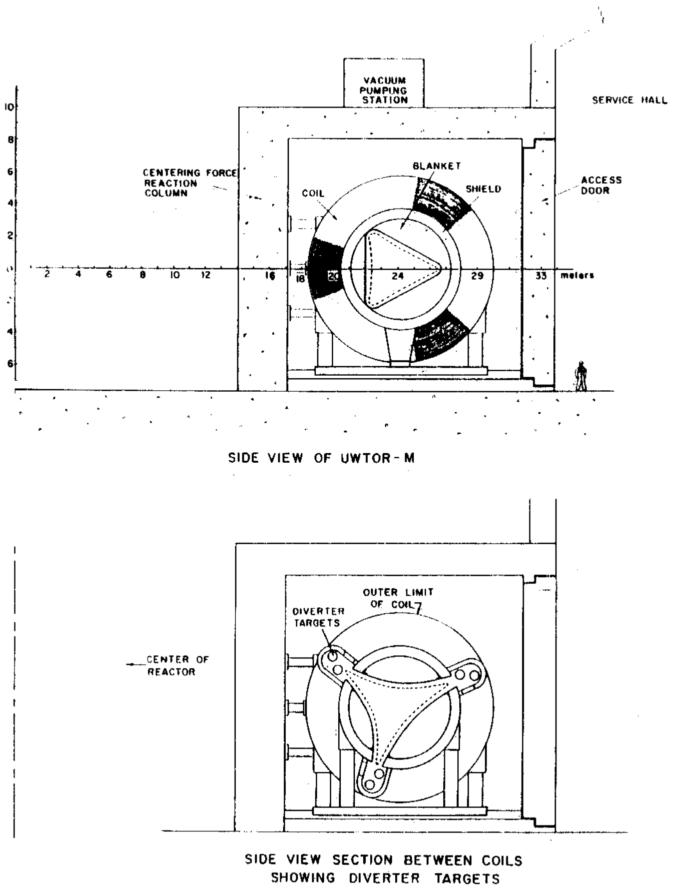

Fig. 1 Side view of UWTOR-M.

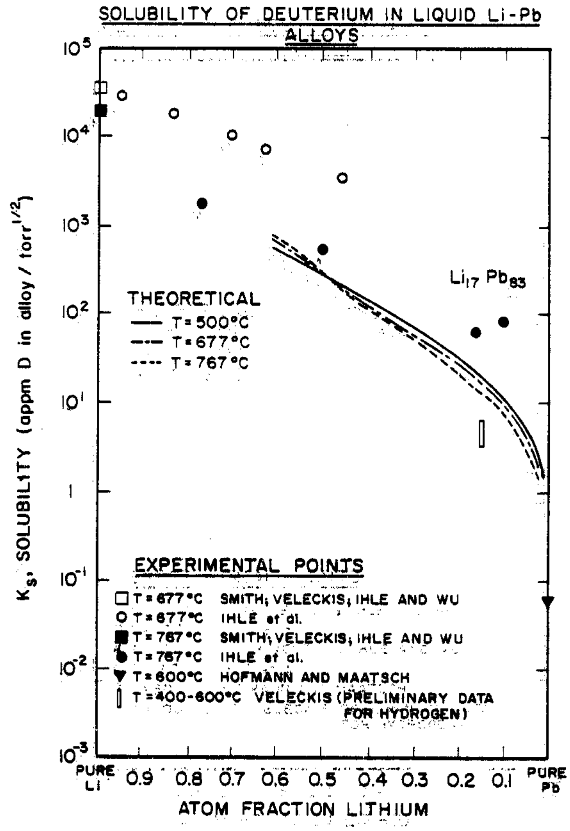

Fig. 3 Solubility of deuterium in liquid Li-Pb alloys.

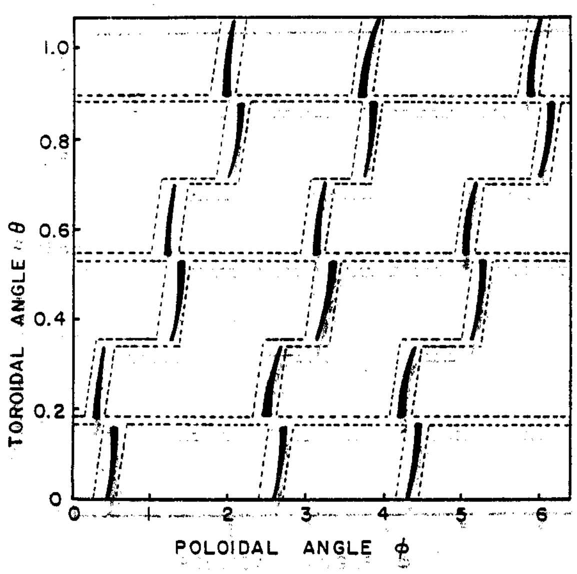

Fig. 2 Available blanket area and divertor zone, mapped into one field period.

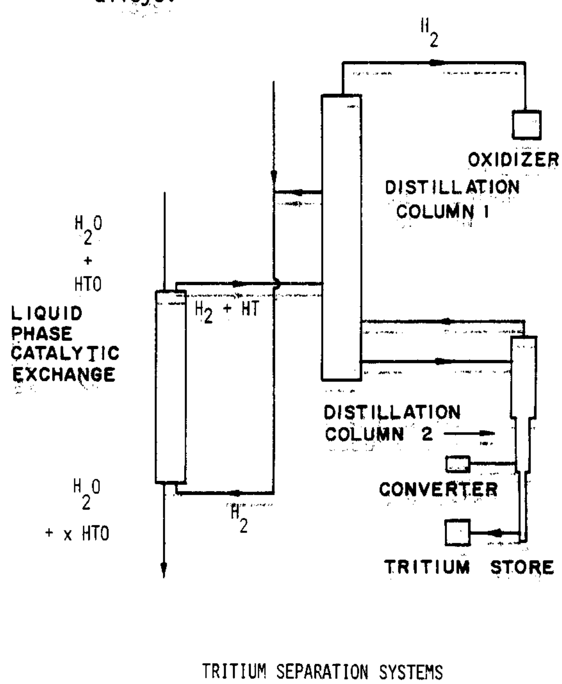

Fig. 4 Tritium recovery system.