CHAPTER 16: TRITIUM

Contributors

Lead Author: Dai Kai Sze

16. TRITIUM

16.1 Design constraints

Tritium recovery and containment are some of the key issues associated with breeding blanket design. The tritium inventory has to be kept low for safety considerations. Also, the tritium loss rate has to be kept below design limit. The severity of the issues associated with the tritium inventory and tritium containment depends on the blanket design, the breeding material to be used, the blanket temperature, and the structural material to be used for the power conversion system.

ITER defines that the maximum allowable tritium inventory in an individual system to be less than 200 g. Also, the allowable tritium loss rate has been assumed to be less than 10 Ci/FPD. Although these limits have been debated within the fusion community, they are the most commonly accepted limits.

16.2 Possible Tritium Recovery Method

16.2.1 Lithium

The most well developed process to recover tritium from lithium is the molten salt recovery process [1]. This process involves three steps. First, lithium is intimately contacted with a salt mixture of lithium halides. LiT, which is a salt, is extracted into the salt solution. Second, the salt phase is circulated to an electrolyser and the LiT is reduced to tritium form, which is carried away by a He stream. Third, the helium is circulated to a clean up unit which recovers the tritium and removes impurities.

All the steps of this process have been demonstrated in laboratory scale. The ability to recover tritium to 1 appm limit was demonstrated. An experimental design for the entail process was developed, but never constructed. The key issue is the mutual solubility of the salt and lithium, which may cause problems associated with the lithium chemistry and also activation.

A new tritium recovery process was developed for ITER applications[2]. This process is based on cold trap. Cold trap was demonstrated to be effective to recover tritium from lithium to the saturation concentration at ~250 appm[3]. However, this concentration is a factor of more than 200 above the design goal of 1 appm. The concept proposed is to add hydrogen into the lithium. During the cold trap process, the total hydrogen concentration can be at the saturation limit of ~250 appm, while the tritium concentration can be reduced to lower than the design goal of 1 appm. A system design of this process was carried out for ITER, including the effect of hydrogen addition to the H/D/T separation. However, no experimental work has been carried out. The key advantage of this process is that it does not introduce any impurities to the lithium stream.

16.2.2 Flibe:

The tritium solubility in Flibe is very low, and obeys Henry's law. Therefore, the tritium recovery from the Flibe to a very low inventory is not an issue. The issue is to reduce the tritium partial pressure so that tritium permeation down the primary coolant path is not an issue. To reduce the tritium permeation to the accepted level of ~ 10 Ci/d, the tritium partial pressure facing the primary heat exchanger has to be lower than 10^{-9} Pa. Even if there is a permeation barrier, with a permeation reduction factor of 100, the allowable tritium partial pressure will still have to be lower than 10^{-5} Pa.

A vacuum disengager was designed for tritium recovery from Flibe[4]. The design forces hot Flibe into small droplets and let them fall through a chamber evacuated to nearly the Flibe vapor pressure, with the tritium diffuses out of the Flibe droplets and being removed. The size of the droplets is 70 micro-m, and the residence time of the droplets in the disengager is 0.5 s.

It is calculated that the efficiency of each disengager is 99.9%. Two disengagers will be used in series to reach a combined efficiency of 99.999%. There has been no experimental work on the tritium removal, and the removal efficiency from the disengager.

16.2.3 LiPb

Various methods have been proposed to recover tritium from LiPb, including permeation[5], liquid-gas contacter[6], permeation into NaK and cold trap[7], and gettering[8]. The most promising method is by permeation in NaK and cold trap.

The tritium solubility in LiPb is very low. Therefore, the tritium recovery to a low inventory is not an issue. The issue is to reduce tritium partial pressure low enough so that tritium permeation is not an issue. For the design of the water-cooled DEMO blanket [9], the tritium inventory is less than 20 g. However, the tritium permeation rates into the coolant and outside the blanket modules are 44% and 42% of the total tritium production, respectively. Only 14% of the tritium produced is carried out by the circulating liquid metal to the tritium recovery system. To keep tritium in the breeding material, a very efficient tritium diffusing barrier development will be necessary.

This tritium recovery method is by permeation to a secondary NaK stream, and using a cold trap process to recover tritium from NaK. The process involves the following three steps: 1. Tritium permeation into a NaK filled gap of a double walled steam generator, 2. Tritium removal from NaK by precipitation as potassium tritide in a cold trap and 3. Tritium recovery by thermal decomposition of the tritide and pumping off tritium gas to a getter bed. The entire process was demonstrated in Reference 7.

16.2.4 Sn-Li

The tritium solubility in Sn-Li is not available. No tritium recovery process has been developed.

16.2.5 He

The following three processes have been proposed for tritium recovery from helium:

1. The tritium is oxidized into water. A molecular sieve will be used to recover the water from helium. This process has to introduce extra oxygen, which may have some serious effects on some of the structural materials.

2. Molecular sieve can be used to recover hydrogen isotopes directly from helium. However, this process is most effective at a cryogenic temperature. Power requirement on this process depends on the tritium in-leak rate, and the allowable tritium concentration in the helium. If the in-leak rate is high and the allowable concentration is low, which is the most likely situation, the power required will be high.

3. Permeation window: Recent work from TSTA has shown that a permeation window can be used for tritium recovery from helium[10]. The achievable concentration maybe too high for fusion blanket applications.

Due to the uncertainties on the tritium in-leak rate to the He coolant, no detailed design of tritium recovery system from helium coolant is available.

16.3 Tritium inventory and control:

The tritium recovery process has to satisfy both the inventory and control issues. Most design process concentrate on the inventory issues. All the tritium recovery processes proposed have been able to recover tritium with an inventory below 200 g in the breeding materials. However, it is far less certain that tritium control issue can be resolved, especially for a high temperature blanket as we are considering. To assess the problems associated with the tritium control issue, Table 16.3-1 outlines the material properties of the various breeding materials consided.

Table 16.3-1 Breeding material properties

	Density, g/cc	Cp, j/g-C	Molecular Weight	Ks Atom frac/Pa$^{1/2}$
Li	0.5	4.2	7	6.6×10^{-3}
LiPb	9.2	0.19	173	9.6×10^{-9}
Flibe	2.0	2.4	99	10^{-2}*

*Flibe obeys Henry's law. The unit is mole TD/liter-atm.

Table 16.3-2 outlines the parameters of the different breeding material on the tritium concentration and the tritium pressure. The concentration per pass is the concentration increase per coolant pass through the blanket. The pressure increase per pass is the tritium partial pressure increase per coolant pass through the blanket. If the tritium recovery system outside the blanket is 100% efficient, the tritium partial pressure in the breeding material at the exit of the blanket will equal the pressure increase per pass.

Table 16.3-2 Tritium parameters of difference breeding material.

	Flow rate, g/s	Δc/pass, appb	Δc/pass, appb
Li	3.6×10^{6}	3.36	2.6×10^{-13}
LiPb	8.0×10^{7}	3.75	0.15
Flibe	6.3×10^{6}	27.4	5.5
He/Solid breeder	??	??	5.5*

*Tritium pressure in the purge.

Table 16.3-3 calculates the allowable tritium partial pressure facing the steam generator. It can be seen that, with the exception of the liquid lithium, a pressure reduction of many orders of magnitude will be required between the exit of the blanket and the entrance of the steam generator, even if a diffusion barrier with a barrier factor of 100 can be developed. Therefore, efficient tritium recovery system, and a tritium diffusion barrier development will be essential.

Table 16-3-3 Allowable tritium partial pressure

Temperature	500ºC
Area	2.5×10^{4} m^2
Allowable tritium leakage rate	100 Curie/d
Oxide barrier factors	1 and 100
Tritium permeability	0.55 mol T_2 - mm/d-m^2-atm
Allowable tritium partial pressure	
With no barrier	1.5×10^{-9} Pa
With barrier factor of 100	1.5×10^{-5} Pa

Conclusions: Tritium recovery methods for different breeding materials have been developed, which are capable to reduce tritium inventory to below the design limits of 200g. For most breeding materials, more than one tritium recovery methods are available to achieve this goal. However, tritium recovery to an acceptable partial pressure if far less certain. With the possible exception of liquid lithium, all other breeding materials will have low tritium solubility and, therefore, a high tritium partial pressure. This issue has been recognized. The design of the tritium recovery process from Flibe tries to achieve a recovery efficiency of 99.999%, while the EC program is aiming at developing an efficient tritium diffusion barrier. The most likely solution is to combine these two efforts, i.e., an efficient recovery system together with multiple layers of diffusing barriers.

References

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