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Tritium processing system for the ITER Li/V Blanket Test Module

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Abstract

The purpose of the ITER Blanket Testing Module is to test the operating and performance of candidate blanket concepts under a real fusion environment. To assure fuel self-sufficiency, the tritium breeding, recovery and processing have to be demonstrated. The tritium produced in the blanket has to be processed to a purity which can be used for refuelling. All these functions need to be accomplished so that the tritium system can be scaled to a commercial fusion power plant from a safety and reliability point of view. This paper summarizes the tritium processing steps, the size of the equipment, power requirements, space requirements, etc. for a self-cooled lithium blanket. This information is needed for the design and layout of the test blanket ancillary system and to assure that the ITER guidelines for remote handling of ancillary equipment can be met. © 1998 Published by Elsevier Science S.A. All rights reserved.

1. Introduction

The testing foreseen for demo test blanket modules includes the demonstration of a breeding capability that would lead to tritium self-sufficiency in a reactor and the extraction of high-grade heat suitable for electricity generation [1]. To accomplish these goals, the ITER horizontal ports will be used to provide a relevant fusion plasma and the appropriate nuclear environment.

Each of the demo blanket concepts may occupy a portion or all of an ITER horizontal test port. The available test space within a test port may be subdivided for simultaneously testing several test articles relating to a general test blanket concept. In addition, testing of more advanced blanket concepts on scaled submodules may be conducted. The testing of the test articles may begin prior to the first plasmas because valuable data may be obtained concerning liquid metal MHD effects in connection with test coatings. The tests will continue to accumulate fluence through the basic performance phase (BPP) and the extended performance phase (EPP).

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Breeding and recovery of tritium are important goals of the test program. Lithium or lithium–ceramic compounds will be the breeder materials to be investigated. Subsystems to recover the bred tritium will be demonstrated along the test facilities to separate and remove the tritium from the coolant or purge streams. Special designs and tritium handling facilities will be required to meet the ITER safety goals and requirements.

2. Tritium handling and processing subsystem

Many processes have been proposed to recover tritium from liquid lithium [2]. A goal of the design process is to limit the tritium concentration in the lithium to ~ 1 appm. This goal, for a commercial power plant, is to limit tritium inventory in the lithium to < 200 g. Due to the high solubility of tritium in lithium and the required low concentration, the tritium recovery from lithium becomes a difficult technical issue.

The tritium recovery method proposed here is based on the cold trap process [3]. The cold trap process has been demonstrated to recover tritium from lithium [4], sodium [5], and potassium [6] to their solubility limits. For the liquid lithium system, the hydrogen solubility at a cold trap temperature of 200°C is 440 appm, far above the design limit as shown in Fig. 1. Therefore, protium is added to lithium to increase the solubility limit (to 1320 appm). Thus, Li(T + H) is supersaturated, although LiT is far lower than the saturation concentration. The co-precipitation of Li(T + H) has not been demonstrated for the liquid lithium system, but the co-precipitation of Na(T + H) has been demonstrated in the breeder program. The Li(T + H) can be separated from the liquid lithium by gravitational force. A process called a 'meshless cold trap', was developed for the breeder program to separate Na(T + H) from Na by gravitation [7][1]. The Li(T + H) can be decomposed by heating to 600°C, and the (T + H) stream will then be fed to the isotope separation system (ISS). An earlier calculation by ITER-Naka personnel estimated that the addition of this stream from the entire blanket to the tritium plant has very minor effects on both the refrigeration power and tritium inventory (Kveton, ITER JCT Naka, personnel communication).

Fig. 2 shows the ISS and the blanket stream. This ISS arrangement is for a complete breeding blanket. The tritium throughput from the blanket testing module is much smaller than that from a complete breeding blanket, but the composition is very similar; hence the location of the inlet and outlet stream to the ISS are the same, with much smaller size tubes.

Table 1 outlines the parameters of the tritium recovery system. The tritium recovery system from the blanket is a small system; the tritium recovery system from the blanket testing module is even smaller. The lithium flow rate to the

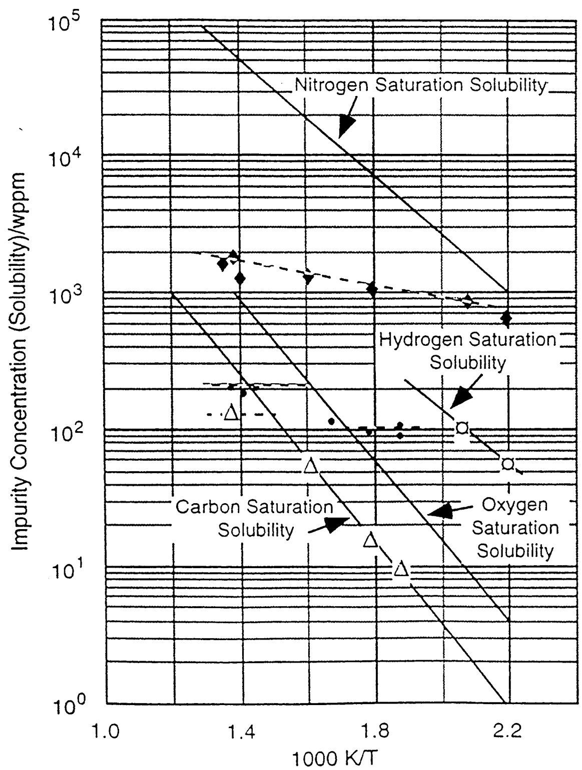

Fig. 1. Cold trap results for different impurities from Li.

[1] The densities of LiH and Li are 0.82 and 0.5 g cm^{-3}, respectively. This density difference is much larger than between NaH and Na.

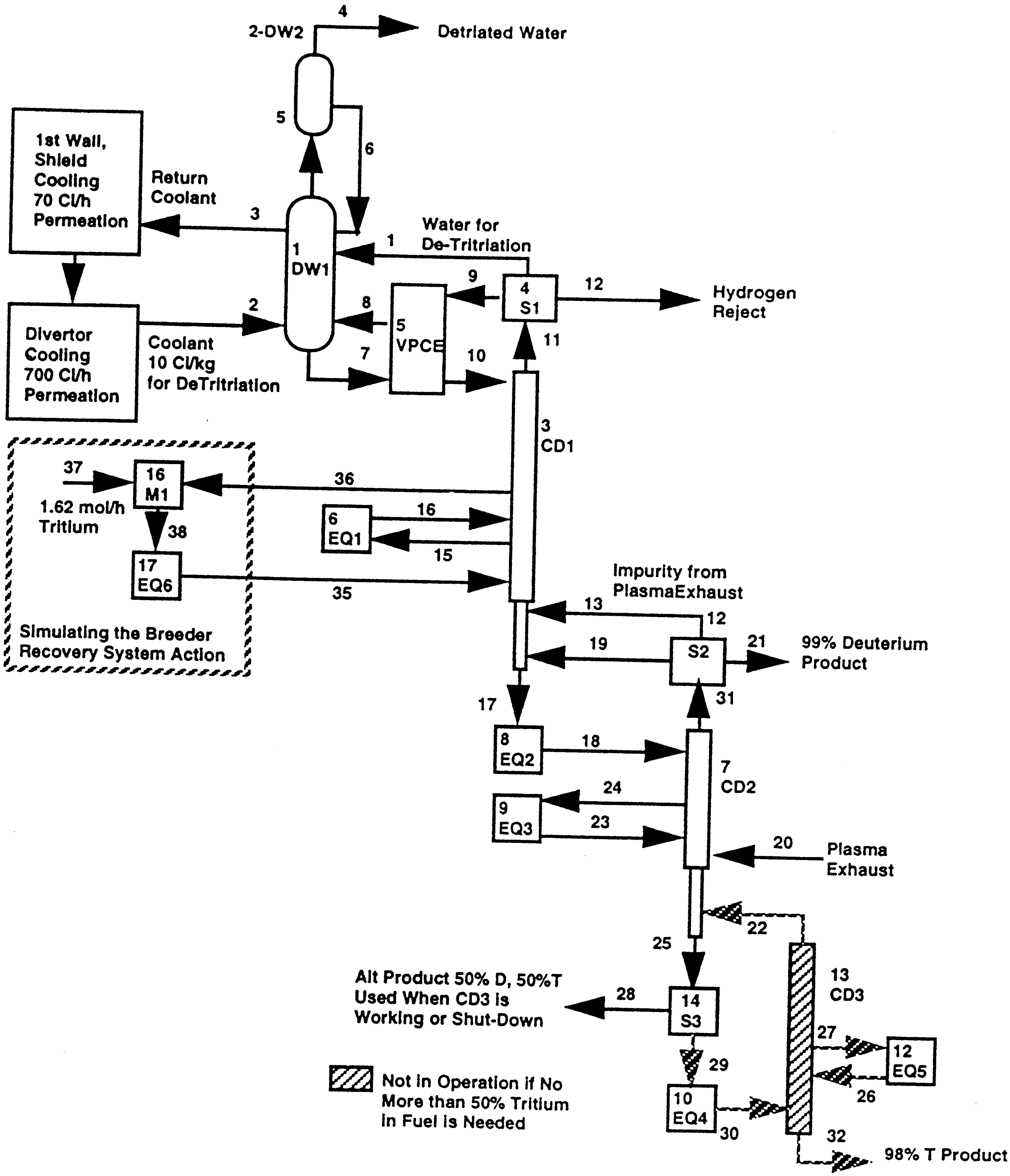

Fig. 2. ITER ISS arrangement.

Table 1
Tritium system parameters

Fusion power (MW)	4
Tritium production rate (g full power day^{-1})	0.6
Tritium concentration at the outlet of the blanket (appm)	1
Tritium concentration after the recovery system (appm)	0.3
Protium concentration in the blanket (appm)	440
Protium concentration at the inlet of the recovery system (appm)	1320
Li flow rate to the recovery system (g s^{-1})	23
Protium addition rate (g s^{-1})	0.003
Li coolant flow rate (g s^{-1})	6000

recovery system is only 23 g s^{-1}. In order to satisfy the requirement for a breeding blanket, the following design goals, shown on Table 2, have to be met. If those design goals cannot be met with the blanket test module, additional experiments will be required to demonstrate those goals that can be achieved in a commercial power plant.

The flow diagram of the tritium processing subsystem is shown in Fig. 3. There is a very limited experimental database for this process applied to liquid lithium systems. But similar steps have been developed for both the liquid sodium and NaK systems

Table 3 summarizes the experimental base which is relevant to this process. Although experimental work will be required to confirm the cold trap process to recover tritium from liquid lithium to ~ 1 appm level, there is reasonable confidence that the subsystem will operate as designed.

Table 2
Tritium system design parameters

Parameters	Commercial power plant	Test module requirement
Coolant temperature (°C)	>500	>500
Tritium inventory	~ 100 g	0.05 g MW^{-1}
Tritium leakage rate	-10 Ci d^{-1}	0.005 Ci d^{-1} MW
Tritium recovery rate	300 g d^{-1}	0.15 g d^{-1} MW

3. Subsystem components

The subsystem components were shown previously in Fig. 3. The experimental base for each step in this process has been listed in Table 3. The following is a summary description of each subsystem components.

1. Protium addition unit. The purpose of this unit is to add protium so that the total hydrogen concentration in the lithium will be higher than the saturation limit at 200°C (440 appm). The protium addition rate is 0.003 g s^{-1}. The hydrogen concentration in the lithium will increase to 1320 appm.
2. Li cooler. Organic coolant is used to cool the lithium from the blanket exit temperature to 250°C.
3. Cold trap. The meshless cold trap is basically a counter-current heat exchanger. Here the lithium is cooled down to 200°C. The cooled lithium will flow against gravitational force, with the Li(H + T) separated from lithium by carefully controlling the lithium velocity.
4. Decomposer. Li(H + T), with some lithium carry over, will be heated up to 600°C. At this temperature Li(H + T) will decompose into (HT) and Li.
5. Cold trap. At 600°C, the lithium vapor pressure is rather high. A cold trap of 200°C will condense the lithium from the hydrogen stream.
6. Palladium effuser. This unit is not shown in the flow diagram but may be desirable. This Pa diffuser will act as a safety barrier for the ISS. Only hydrogen isotopes will pass across the diffuser and be fed into the ISS.
7. Isotope separation subsystem. The tritium stream from the blanket testing module is fed into the isotope separation subsystem of the tritium plant at a location which will match the composition in the ISS.

4. Safety

The goal of tritium recovery from lithium is to reduce the tritium concentration to ~ 1 appm. An out-of-reactor test will be carried out to determine

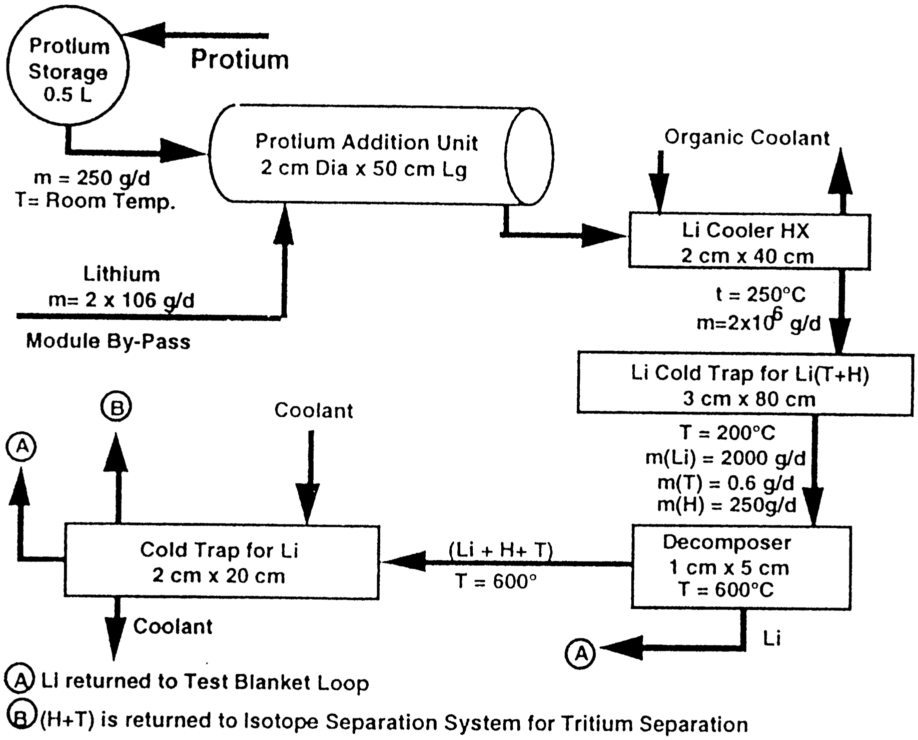

Fig. 3. Cold trap process flow diagram.

that this goal can be achieved. Table 4 summarizes the expected tritium inventory and the tritium partial pressure for the blanket test module system from which tritium safety can be assessed. If these values can be demonstrated, then neither the tritium inventory nor the tritium partial pressure will cause any major tritium safety concerns.

Table 3
Experimental data base for cold trap

Steps	Experimental database
Protium addition	Demonstrated in breeder program
Tritium cold trap	Demonstrated for Li, Na, NaK systems
Co-precipitation of (H+T)	Demonstrated in breeder program
Hydride decomposition	Demonstrated in breeder program
Impurities removal	Demonstrated in TSTA
Isotope separation	Demonstrated in TSTA

5. Summary

The design of the tritium system for the ITER self-cooled lithium test blanket module has been completed. Both tritium inventory and tritium partial pressure in the system are rather low.

Table 4
Tritium subsystem safety-related parameters

Lithium	Volume
Blanket (m^3)	2
Li/organic coolant HX (m^3)	0.94
Main electro-magnetic pump (m^3)	0.30
Li surge tank (m^3)	0.2
Li dump tank (m^3)	1.57*
Total (m^3)	3.44
Tritium concentration in lithium (appm)	1
Total tritium inventory (g)	0.2
Lithium maximum temperature (°C)	600
Tritium partial pressure over lithium (Torr)	7×10^{-10}

* The dump tank is empty during the operation.

Therefore, it is not expected that this tritium system will have any significant safety impact on the ITER system. The tritium processing system can be combined with the ITER ISS, with only moderate impact on both the tritium inventory and the refrigeration power requirement.

The tritium recovery is based on cold trap. This process has not been demonstrated experimentally. However, similar systems have been developed for both sodium and potassium system. Therefore, there is a high degree of confidence that the tritium system will operate as designed. However, experimental verification will still be required.

References

[1] E. Proust, M. Abdou, Y. Gohar, R. Parker, Y. Strebkov, H. Takatsu, DEMO Blanket Testing in ITER and the International Collaboration via ITER Test Blanket Working Group, ISFNT-4, Tokyo, Japan, April 7–11, 1997.

[2] H. Moriyama, E. Tanaka, D.K. Sze, J. Riemann, A Terlain, Tritium recovery from liquid metals, Fusion Eng. Des. 28 (1995) 226–239.

[3] D.K. Sze, R.F. Mattas, J. Anderson, R Haange, H. Yoshida, O. Kveton, Tritium recovery from lithium based on cold trap, Fusion Eng. Des. 28 (1995) 220–225.

[4] J.R. Watson, W.F. Calaway, R.M. Yonco, V.A. Maroni, Recent Advance in Lithium Processing Technology at Argonne National Laboratory, Proc. 2nd Int. Conf. on Liquid Metal Technology in Energy Production, Richland, WA, 20–24, April, 1980.

[5] C.C. McPheeters, D.J. Raue, Cold Trap Modeling and Experiments on NaH Precipitation, Proc. 2nd Int. Conf. on Liquid Metal Technology on Energy Production, Richland, WA, 20–24, April, 1980.

[6] J. Riemann, R. Kirchner, M. Pfeff, D. Rackel, Tritium removal from NaK-cold traps, first results on hydrige precipitation kinetics, Fusion Technol. 21 (1992) 872.

[7] R.L. Eichelberger, Testing of a Meshless Cold Trap from Hydrogen Removal from Sodium, Proc. 2nd Int. Conf. on Liquid Metal Technology in Energy Production, Richland, WA, 20–24, April, 1980.