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聚变实验增殖堆氚系统设计研究

TRITIUM SYSTEM DESIGH STUDIES OF FUSION EXPERIMENTAL BREEDER

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聚变实验增殖堆氚系统设计研究

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摘　要

介绍了核工业西南物理研究院聚变实验增殖堆工程概要设计（FEB-E）中的氚系统设计研究。第一部分介绍包层氚增殖区的划分、几何尺寸、装料特征和用蒙特卡罗程序计算得到的液态锂中的氚浓度分布；第二部分描述根据聚变堆氚物理基础构造的氚循环系统，共分成 10 个子系统及它们之间氚的流程图。运用研制的程序 SWITRIM 计算了各个子系统中的氚投料量随时间的变化，满功率运行一年后各个子系统中的氚投料量。研究结果表明起动 143 MW 聚变功率 FEB-E 堆所需要的初始氚投料量大约为 319 g。第三部分对不同的运行状态下的氚泄漏问题进行了分析。潜在的氚泄漏危险可能来自于偏滤器系统从等离子体中抽出的气体。得到的结论是提高 FEB-E 堆芯等离子体的燃耗份额从而减少氚的通过量对降低氚的泄漏危险是重要的。

关键词： 氚系统设计　聚变实验增殖堆　氚投料量　氚泄漏分析

Tritium System Design Studies of Fusion Experimental Breeder

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ABSTRACT

A summary of the tritium system design studies for the engineering outline design of a fusion experimental breeder (FEB-E) is presented. This paper is divided into three sections. In first section, the geometry, loading features and tritium concentrations in liquid lithium of tritium breeding zones of blanket are described. The tritium flow chart corresponding to the tritium fuel cycle system has been constructed, and the inventories in ten subsystems are calculated using SWITRIM code in section 2. Results show that the necessary initial tritium storage to start up FEB-E with fusion power of 143 MW is about 319 g. In final section, the tritium leakage issues under different operation circumstances have been analyzed. It was found that the potential danger of tritium leakage could be resulted from the exhausted gas of the diverter system. It is important to elevate the tritium burnup fraction and reduce the tritium throughput.

Keywords: Tritium system design, Fusion experimental breeder, Tritium inventory, Tritium leakage

INTRODUCTION

Based on the engineering outline design of a Fusion Experimental Breeder (FEB-E)[1,2], the tritium concentrations in liquid lithium of the tritium breeding zones have been calculated after ten days operation for the outboard blanket and one day operation for inboard blanket. The tritium breeding ratios are calculated by using 3-D Monte Carlo code MORSE-CGT. The tritium inventory in the beryllium multiplier is also estimated after one-year operation. The tritium flow chart of FEB-E reactor system is preliminarily outlined [3]. Tritium recovery from plasma exhaust gas has been considered by combining fuel cleanup unit with cryogenic distillation.

A dynamic subsystem model has been constructed to describe tritium fuel cycle system, and a computer simulation code SWITRIM has been developed to simulate the tritium cycle system [4]. The time-dependent tritium inventories in ten subsystems are calculated during one-year operation period. This code is utilized to predict the necessary initial tritium storage to start up the experimental breeder with a fusion power of 143 MW, which is about 319 g [5]. The calculation results show that the required initial tritium inventory strongly depends on the mean residence times in the subsystems of plasma exhaust gas, fuel cleanup unit (FCU) and the isotope separation system (ISS). Besides, the required initial tritium inventory also depends on the tritium permeability in the first wall, limiter, divertor, blanket and the tritium production in the beryllium neutron multiplier as well.

Finally, the Sieverts's law and SWITRIM code have been used to perform the tritium leakage analysis under the circumstances of normal operation and the accident cases [6]. The results show that the tritium partial pressures in the liquid lithium of FEB-E blanket are not high for both cases. The potential danger of tritium leakage may be resulted from the exhausted gas from the pumping system of the divertor.

1 DESIGN DETAILS OF TRITIUM BREEDING BLANKET

1. 1 Blanket outline features

The blanket has been divided into sixteen sectors along toroidal direction. There are three outboard modules and two inboard modules in one sector. Each outboard

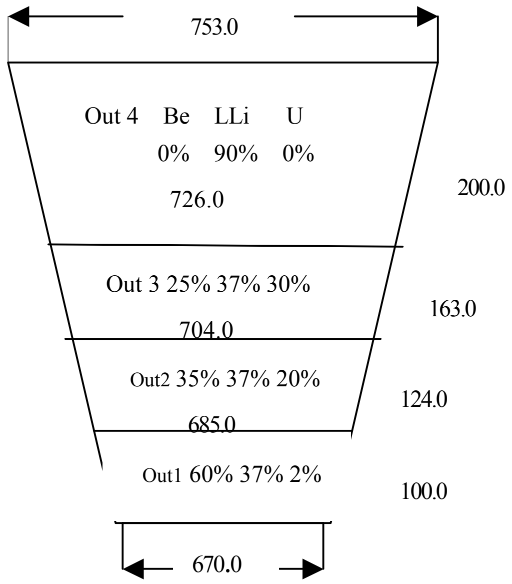

Fig.1 (a) Outboard module

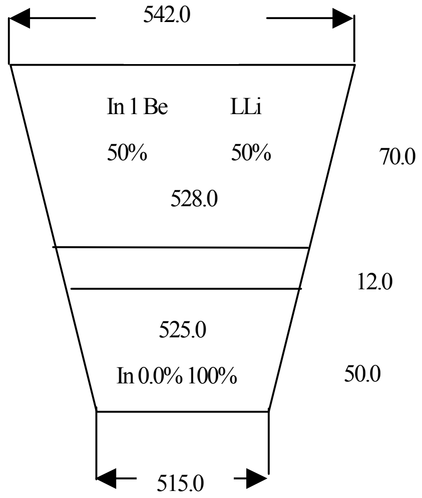

Fig. 1 (b) Inboard module

module consists of four breeding zones denoted by Out 1, Out 2, Out 3 and Out 4; while inboard module has two zones denoted by In 1 and In 2. The details of geometry and loading material composition of each module are shown in Fig. 1, Table 1 and Table 2. The three figures of percentage in four zones of Fig. 1(a) from left-hand side to right side represent the fractions of nuclide number of beryllium (Be), liquid lithium (LLi) and uranium (U) respectively. Their corresponding mean nuclide number densities are given in Table 1. The figures of length and thickness of these zones are in unit of millimeter.

Table 1 Loading material details of each module in inboard and outboard blanket (in 10^{22} nuclide·cm^{-3})

Zones	^{6}Li	^{7}Li	^{9}Be	^{238}U	^{235}U
In 1	1.800	0.20	5.29		
In 2	3.600	0.40			
Out 1	0.040	1.62	7.25	0.093	2.34×10^{-4}
Out 2	0.039	1.59	4.12	0.908	2.28×10^{-3}
Out 3	0.038	1.54	2.87	1.330	3.33×10^{-3}
Out 4	0.090	3.66			

Table 2 Volume and surrounding area of each zone

Zones	Volume 10^5/cm^3	Area 10^4/cm^2
In 1	1.48	4.77
In 2	1.09	4.76
Out 1	2.60	5.97
Out 2	3.64	6.93
Out 3	5.50	8.29
Out 4	7.86	9.99

1. 2 3-D Monte-Carlo calculation

The liquid lithium is used as the tritium breeder and high-pressure helium gas as the coolant in FEB-E blanket. The beryllium grain particles are uniformly immerged in liquid lithium as the neutron multiplier. The plasma burn-up fraction of 2.8% is designed. The three dimensional Monte Carlo code MORE-CGT is used to calculate the tritium-breeding ratios of these zones in FEB-E blanket and the results are given in Table 3. The total TBR=1.10 and TBR_{out}=0.45, TBR_{in}=0.65 are the partial tritium breeding ratios contributed from outboard and inboard blanket respectively.

Table 3 Tritium breeding ratio distribution in the breeding zones

Zone	T_f
Out1	0.2065
Out2	0.1271
Out3	0.0534
Out4	0.0630
In 1	0.4275
In 2	0.2224

1. 3 Tritium recovery

The tritium recovery from liquid lithium is difficult to use tritium window because the permeation rate through niobium is too low[7, 3]. At present, the modified cold trap[8] would be a better choice. Tritium recovery schemes from plasma exhaust gas, helium coolant and liquid lithium have been discussed in our previous work[3], and will be mentioned briefly in the following section.

2 COMPUTER SIMULATION

2. 1 Subsystems and physics bases

The FEB-E tritium cycle system is divided into 10 subsystems. The simulation has been treated as time-dependent problem and 100% availability is assumed. The physics bases of SWITRIM code are based on the following assumptions:

1) Tritium storage and fuelling subsystem have initial tritium storage Y_0 (0)=0.5 kg. The tritium container has been coated with permeation barrier to reduce non-radioactive loss fraction to 0.0001/d and radioactive decay constant λ=1.54×10^{-4}/d is also considered. Pellet fabrication, acceleration and launching processes take total time T=20 minutes. Fuelling rate N=1.073 kg/d and plasma fractional burn-up β=0.0208 have been calculated in previous FEB-E design.

2) Outboard blanket: Tritium breeder liquid lithium is designed for being periodically moved out to recover tritium every 10 days with a tritium concentration of about 10 appm (1 ppm=10^{-6})[3].

3) Inboard blanket: The liquid lithium (LLi) is moved out to recover tritium every 24 h with a tritium concentration of about 10 appm. Total tritium breeding ratio Λ=Λ_1+Λ_2=1.10 is obtained from neutronics calculations, Λ_1 =0.45 is

contributed from outboard blanket and Λ_2=0.65 from inboard [5]. Tritium permeates to the helium coolant from LLi with fractions ε_1 for outboard and ε_2 for inboard. The method of mean residence time description is adopted to describe the tritium transfer from one subsystem to other subsystems.

4) First wall, limiter and divertor are cooled by helium. The tritium is recovered from helium coolant every 100 days if blanket temperature runs higher than 680℃. Tritium permeation from plasma to the structural materials of first wall, limiter and divertor is characterized by a fraction of σ= 0.01%/d [9] with considering PDP (plasma driven permeation) enhancement. However, no trapping effects of defects produced by neutron damage [10] are considered. The tritium held up in structural materials is regarded as unrecoverable. The tritium loss from structural materials to the coolant is taken into account by ε_3 =0.0001/d.

5) Plasma exhaust consists of the unburned fuel particles, helium ash, tritiated water and other impurities. Tritium mean residence time in this subsystem takes 0.5～1 h, and non-radioactive fractional loss rate is taken as ε_4=0.0001/d.

6) Fuel cleanup unit (FCU) is a palladium membrane reactor which consists of permeator and catalytic reactor to shift water and methane into gases. The processing time takes 1～2 h.

7) Isotope separation system (ISS) is combined with cryogenic fractional distillation system, which has been improved by cold trap[8], with necessary steps of precipitation and decomposition to recover tritium from LLi, and total residence time takes 3.5～7 h.

8) Tritium waste treatment (TWT) subsystem is designed to treat the solid tritium waste with low tritium concentration; the residence time takes 10 h. The factor *g* denotes a fraction of unrecoverable tritium in the waste.

9) Beryllium neutron multiplier, in this subsystem the tritium inventory comes from the yielded tritium of the reactions ^{9}Be(n, t) ^{7}Li and ^{9}Be+n→^{4}He+^{6}He, ^{6}He → ^{6}Li+β^-, ^{6}Li+n →^{4}He+T and implanted tritium resulted from energetic triton. The parameters b and γ are evaluated from the empirical scaling law derived from ARIES-1 design[11] to take into account the two effects as mentioned above. For the time being, we have not yet considered recovering tritium from beryllium. Non-radioactive fractional loss rate is 0.0001%/d.

10) Helium coolant subsystem, in which the tritium is recovered from helium coolant every 100 days. All above parameters are referred to published materials of TSTA [12].

2. 2 Governing equations and input parameters

The mean residence time method is used to calculate the tritium inventories in ten subsystems. In our mean residence time model, the governing equations to describe the time variation of inventories in ten subsystems and the input parameters of a reference case are given in our previous work [5].

2. 3 Tritium inventories in subsystems

The tritium flowchart in FEB-E tritium cycle subsystem model for the computer simulation is shown in Fig. 2. The inventories in ten subsystems at the end of one-year-operation are shown in Fig. 3. It is found that the tritium inventory in ISS subsystem $Y_6 \approx 0.203$ kg is dominant. This is because of low plasma burn up fraction in FEB-E design. From Fig. 4 we can find that the required minimum initial tritium inventory is approximately 319 g, which would result in a zero storage at 30th day, if the initial tritium storage were not 0.5 kg, but 319 g.

The tritium storage is Y_0=0.767 kg and total tritium inventory is Y_{11}=1.485 kg, respectively at the end of continuous 365-day operation. The tritium inventory in helium coolant is $Y_{10}=7.086\times10^{-3}$ kg. The tritium retention in structural materials of first wall, limiter and diverter is Y_3=0.0437 kg. The tritium retention in beryllium multiplier is Y_9=0.1725 kg. The tritium inventories of the following six subsystems are approaching steady values for above-given working status. The final inventories at the end of 365-day-operation are listed in Table 4.

These results are fairly consistent with the predictions given by Anderson from TSTA (Tritium System Test Assembly)[12] and the results of our previous simulation using different calculation model[4]. Our conclusions are that a higher fractional burn-up β of core plasma is important to reduce the required initial tritium storage which strongly depends on the processing time in plasma exhaust system and tritium processing time in the subsystems of FCU, ISS and LLi etc[4]. Large amount of tritium fueled into plasma is not burned up, but passed through plasma exhaust system. In reality, the minimum tritium storage should support fueling needs until the first batch of tritium is separated from ISS and sent back to the tritium storage and fueling subsystem.

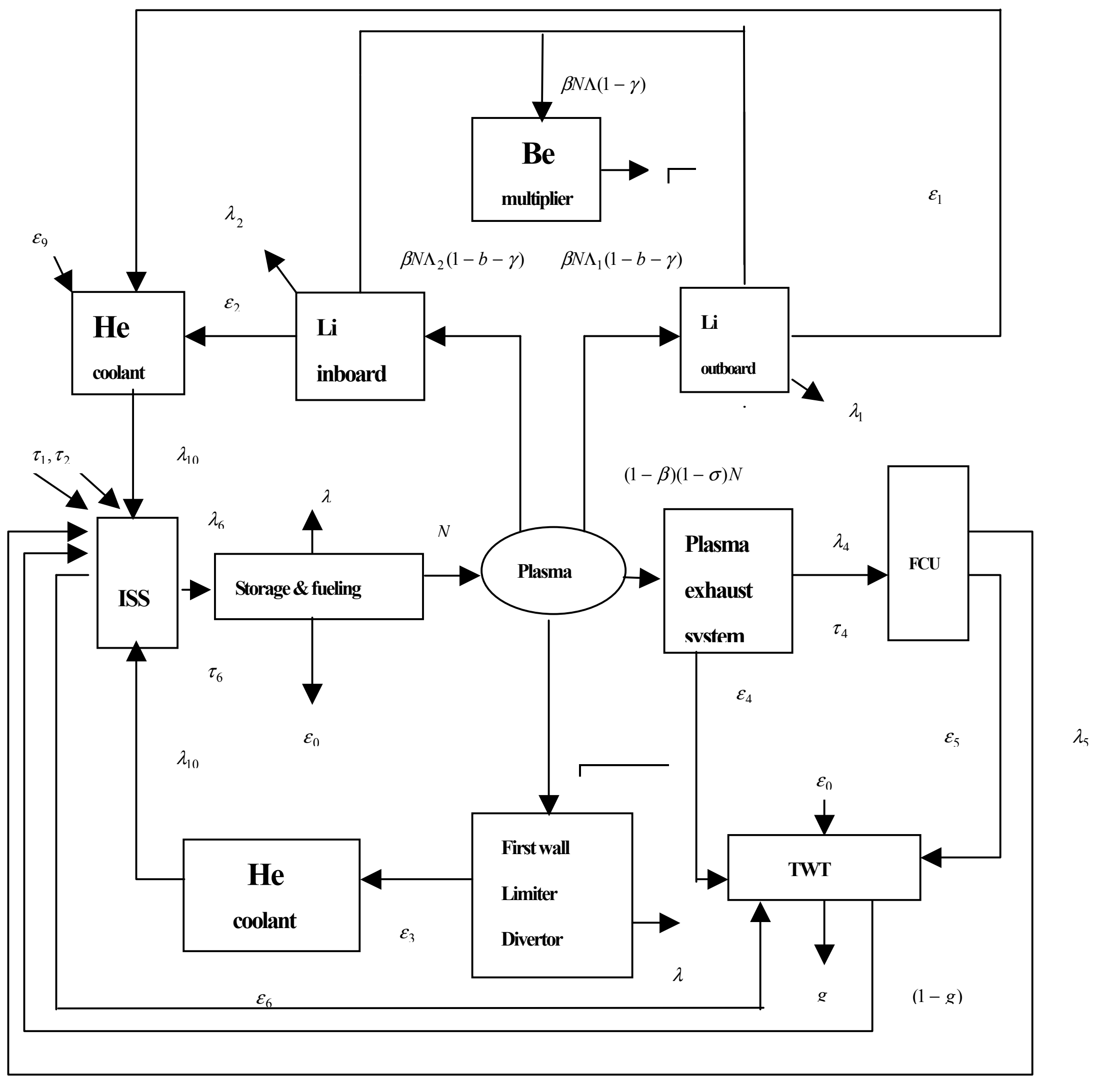

Fig. 2 Computer simulation model of FEB-E tritium cycle subsystems

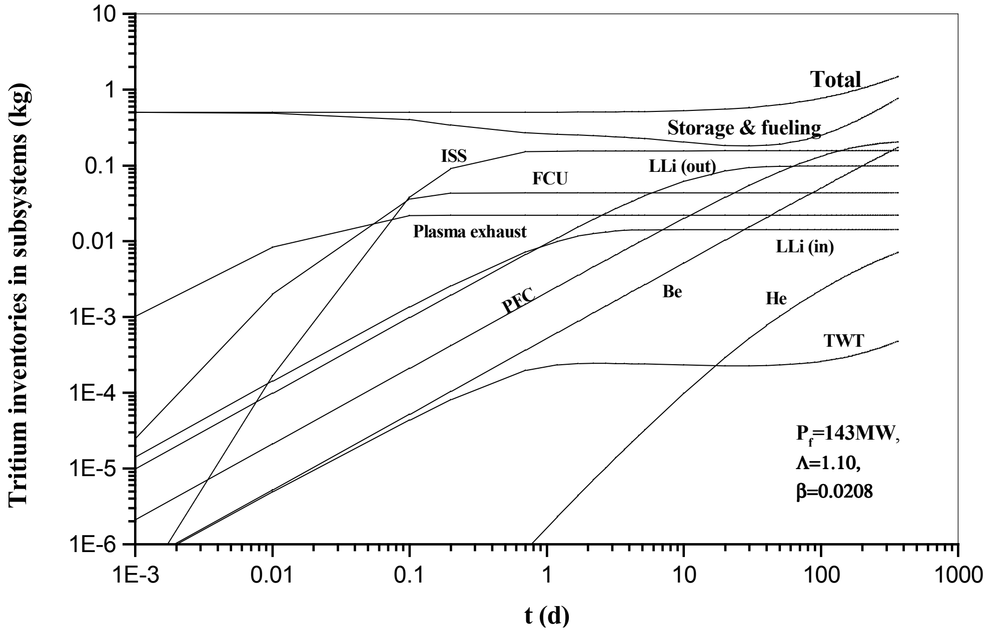

Fig. 3 Tritium inventories vary with time in subsystems

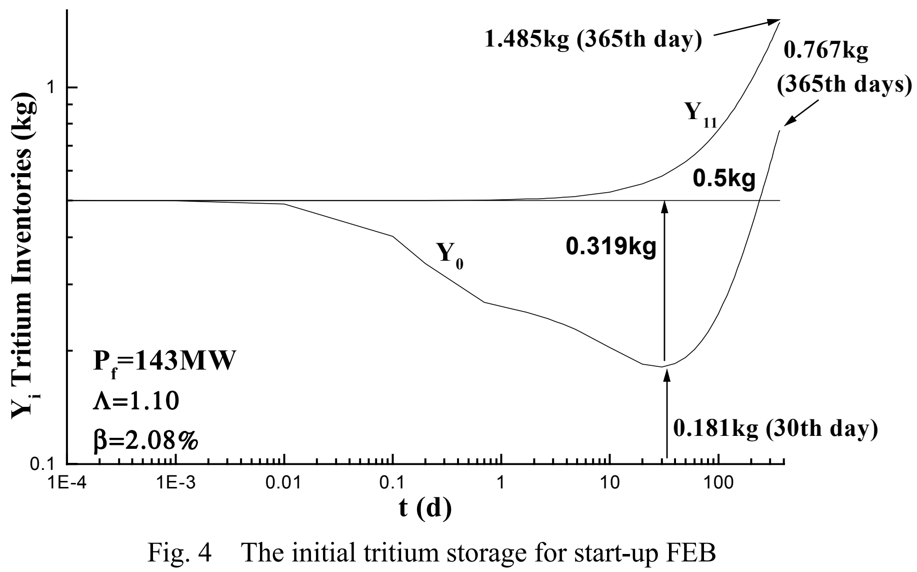

Fig. 4 The initial tritium storage for start-up FEB

Table 4 Balanced tritium inventories in six subsystems

Subsystems	Tritium inventories /kg
ISS	Y_6=0.20314
Plasma exhaust	Y_4=0.02183
FCU	Y_5=0.04372
LLi (inboard)	Y_2=0.09804
LLi (outboard)	Y_1=0.01422
TWT	Y_7=0.00048

3 Analysis of tritium leakage

The tritium control is a very important issue for a D-T fueled fusion reactor, because not only tritium is very expensive but also tritium leakage causes environmental contamination. Usually the allowable tritium leakage rate of a fusion reactor should be less than 10 Ci/d (1Ci=3.7×10^{10} Bq)to the environment. For the FEB-E design, the high-pressure helium gas is used as the coolant and liquid lithium as the tritium breeder; hence the tritium produced in liquid lithium has strong chemical affinity combining into LiT with lithium. Under the normal operational circumstances, the FEB-E blanket temperature≤953 K, the tritium permeation leakage through blanket structural materials is almost negligible due to the low tritium gaseous partial pressure. The tritium leaks out of reactor mainly through plasma exhaust fuel treatment systems (fuel clean up unit and isotope separation system). Because the plasma fractional burn-up in FEB-E is rather low (β=2.08%), therefore, the tritium throughput is high. Tritium diffuses across first wall and diverter target plate to the helium coolant. In this way, the working fluid (helium or liquid lithium) loss leads to the tritium loss. The tritium containing liquid lithium loss during the process of tritium recovery from liquid lithium directly leads to the tritium loss. The SWITRIM code and Sieverts' law are used to estimate the tritium partial pressure in liquid lithium from calculated tritium concentration and the temperature-dependent solubility. The tritium permeation rate across the stainless steel into helium coolant follows a function of the difference of the square root of the partial pressures. The FEB-E tritium leakage analysis is based on the assumption of working fluid loss rates. In the accident circumstances, the tritium leakage becomes a concern under the loss of control of the blanket temperature; LiT is decomposed into tritium, which permeates to the coolant due to increasing tritium partial pressure in liquid lithium.

3. 1 Assumptions

Under the normal conditions, permeation of tritium across the metal barrier results in tritium contamination of coolant, then any working fluids loss can lead to tritium leakage. Under the LOFA or LOCA accident events, the liquid lithium temperature increases rapidly, permeation of tritium across metal barrier into coolant become important. Our calculations are based on the following assumptions:

(1) Helium coolant loss at a rate of 0.2%/d.

(2) Vacuum leakage of plasma exhaust system at a rate of 10^{-9} Pa·m^3/s.

(3) Liquid lithium loss at a rate of 0.0001%/d.

(4) SWITRIM code is used to calculate the inventories of the subsystems.

3. 2 Tritium solubility in blanket liquid lithium

The tritium solubility in liquid lithium follows Sieverts' law[13]:

$$K_s = \frac{C_{\text{LiT}}}{p_t^{1/2}} \tag{1}$$

K_S is the tritium solubility in liquid lithium, in unit of mole fraction.Pa$^{-1/2}$, C_{LiT} is the tritide concentration in lithium in mole fraction, and P_t is the equilibrium tritium partial pressure in Pa. Within the temperature 723 K<T<923 K, the following expression has been reported[14]

$$K_s = 1.2\times10^{-5}\exp(44400/RT) \tag{2}$$

For FEB-E design bases[1,2], the temperatures in each zone have been calculated by thermal and hydraulic analysis, and K_s can be calculated from the average temperatures of each zone in blanket and shown in Table 5.

3. 3 Tritium leakage from blanket lithium during normal operation

Under normal condition, the tritium concentrations of six zones denoted by C_{Li}(Out*, I*) and C_{Li}(In*, j*) (*i=1,2,3,4;j=1,2*) in liquid lithium of FEB-E blanket expressed by tritium mole fractions have been obtained[3]. Based on Sievetrts' law, the tritium pressures P_t in six zones can be obtained and are shown in Table 6.

Where C_{Li} are the tritium mole fractions of each breeding zone in outboard and inboard blanket. Then the tritium partial pressures can be calculated. Results show that for designed FEB-E tritium cycle, during the time duration between two successive tritium extraction phases, the buildup tritium partial pressure could not

Table 5　Average temperature and solubility in each zone of blanket

Zone	T /K	K_s(mole fraction $Pa^{-1/2}$)
Out 1	620	6.6×10^{-2}
Out 2	792	1.0×10^{-2}
Out 3	683	2.9×10^{-2}
Out 4	600	8.8×10^{-2}
In 1	642	4.9×10^{-2}
In 2	700	2.5×10^{-2}

Table 6　Tritium concentration, solubility and pressure in each zone

Zone	$C_{Li}(\times10^{-6})$	$K_s(\times10^{-2})$	P_t /pa
Out 1	4.55	6.6	4.7×10^{-9}
Out 2	6.60	1.0	2.7×10^{-6}
Out 3	4.27	2.9	2.2×10^{-8}
Out 4	1.81	8.8	4.2×10^{-10}
In 1	16.71	4.9	1.2×10^{-7}
In 2	6.78	2.5	7.3×10^{-8}

be high. The tritium in liquid lithium of blanket diffuses through metal barrier and enters the helium coolant, and then the coolant leakage leads to the tritium loss. The flow rate of tritium permeation through a metal is a function of tritium partial pressure, and the tritium partial pressure is related to the tritium concentration by Sieverts' constant. If we omit the effects of activation energy and the surface recombination, the permeation rate can be described as a function of the difference of the square root of the partial pressures:

$$V = \frac{cA}{dM_t^{1/2}}(P_u^{1/2} - P_d^{1/2}) \tag{3}$$

However, for the refractory metal, if the pressure difference between upstream and downstream decreases to as low as 1.33×10^{-4} Pa, the permeation rate will be proportional to the first-power difference of partial pressures across the metal barrier:

$$V = \frac{c'A}{dM_t^{1/2}}(p_u - p_d)\exp(-\frac{Q}{RT}\times10^{-5}) \tag{4}$$

Here Q refers to the effect of diffusion activation energy, c' is the permeation constant measured in experiments, which is not available in existing literature. In this paper we try to extrapolate Sieverts' law to be used in FEB-E blanket. V is flow rate of tritium (STP), in cm^3/h; A is permeation area, in cm^2; c is permeation constant, in cm^3 (STP)mm/(h·cm^2·$Pa^{1/2}$)(1atm=101325 Pa); d is thickness of metal barrier, in mm; p_u and p_d are the tritium pressures of upstream and downstream, respectively in Pa; M_t refers to the atomic weight of tritium, in amu; For FEB-E blanket design, high-pressure helium coolant flows in stainless steel channel. The tritium permeation constants c in stainless steel taken from reference [13] (the effects of surface oxide membrane are not considered) in unit of cm^3 (STP) mm/(h·cm^2·$Pa^{1/2}$) are listed in Table 7.

Table 7 Tritium permeation constant in each zone

Zone	T /K	c (cm^3(STP)·mm/(h·cm^2·$Pa^{1/2}$))
Out 1	620	0.63×10^{-5}
Out 2	792	1.27×10^{-4}
Out 3	683	1.90×10^{-5}
Out 4	600	0.32×10^{-5}
In 1	642	0.89×10^{-5}
In 2	700	2.53×10^{-5}

The wall thickness of coolant stainless steel tube d=2 mm. The surrounding coolant interface areas A of each zone in FEB-E blanket have been obtained in Table 2. By considering the activation energy of tritium diffusion Q=42.3 kJ/mol=10 k cal/mol (1cal=4.1868 J), the tritium leakage rates V can be calculated (in STP) and given in Table 8.

Under the normal operational status, the total tritium leakage into helium coolant from liquid lithium of blanket can be given:

$$V=\sum_{i=1}^{4} V_{\mathrm{Out}i}+\sum_{j=1}^{2} V_{\mathrm{In}j}=5.96\times10^{-2}\ \mathrm{cm^3/d\ (STP)} \tag{5}$$

That is equivalent to 0.15 Ci/d. We can see tritium leakage from helium coolant is small; therefore, the tritium contamination caused by tritium leakage is negligible.

Table 8 Coolant interface area and tritium leakage rate of each zone

Zone	A 10^4/cm^2	V 10^{-3}/cm^3/d, (STP)
Out 1	5.97	0.098
Out 2	6.93	56.69
Out 3	8.29	0.912
Out 4	9.99	0.025
In 1	4.77	0.57
In 2	4.76	1.29

4. 4 Tritium leakage from plasma exhaust under normal case

Because the temperatures of first wall, limiter and diverter materials are higher than 650 K, not only the tritium diffusivities in these materials increase but also the enhancement of plasma driven permeation are put into effect, and tritium will be hold up in helium coolant. SWITRIM code has been used to calculate the tritium inventory of helium coolant[4]. Assuming that helium leakage at a rate of 0.2%/d, the average tritium leakage rate from helium to the environment will be 6×10^{-2} Ci/d.

According to the usual leakage rate of diverter pumping vacuum system about 10^{-9} Pa·m^3/s, the leakage rate of plasma exhaust system will be 6.4×10^{-5}/d based on FEB·E designed plasma volume and plasma parameters. The equilibrium tritium inventory of plasma exhaust system has been obtained 0.022 kg[4]. The tritium that is permeated to the environment through plasma exhaust system will be 14 Ci/d. If the vacuum chamber is sealed with a double layer, and some measures are taken to reduce the vacuum leakage rate to 0.7×10^{-10} Pa·m^3/s, then the tritium that is leaked to the environment through plasma exhaust system will be 0.9 Ci/d. A small amount of liquid lithium possibly loses during the process of tritium recovery from liquid lithium. Hansborough[13] has suggested that liquid lithium loss rate be 0.0001%/d. Total tritium inventory of 0.0126 kg in liquid lithium of inboard and outboard blankets has been obtained by SWITRIM code calculation. The tritium leakage rate resulted from blanket liquid lithium loss will be 0.13 Ci/d. The isotope separation system and fuel clean up unit are enclosed in a highly sealed workshop surrounded by the cover helium gas. The tritium leakage from cover helium gas is not considered in this paper.

3. 5 Tritium leakage analysis under accident case

Assuming that LOFA or LOCA has happened in fuel zones of FEB-E blanket where the temperature runs up to 1273 K, the tritium solubility in liquid lithium can be expressed as follows:

$$K_s = 8.5\times10^{-6}\exp(42300/RT) \tag{6}$$

The tritium flow rates entering helium coolant across the wall between coolant liquid lithium are shown in Table 9. The tritium solubility in beryllium pellets depends on the temperature and has been measured by using permeation method by Kizu [15]:

$$S = 7.1\exp(-14/R\overline{T}) \tag{7}$$

Equation (7) has been used to estimate the tritium inventory in beryllium multiplier under the accident events.

Table 9 Tritium flow rate from lithium into coolant of each zone

Zone	p_t 10^{-2}/$Pa^{1/2}$	V /(C*i*/d)
Out 1	0.99	40
Out 2	3.6	168
Out 3	0.93	52
Out 4	0.39	26
In 1	3.63	116
In 2	1.47	47

Therefore, under the accident events if the lithium temperature rises as high as 1273 K, the tritium leakage rate will be 449 Ci/d from liquid lithium into coolant. According to the regulations, the allowable helium coolant leakage rate is 0.2%/d for the environment safety. Once this kind of accident occurred, some effective measures should be taken to deal with the accident within 24 h. Then tritium leakage to the environment would not be more than 0.9 Ci/d. Under LOVA event, assuming FEB-E vacuum system is getting worse, vacuum leakage rate increases to 10^{-8}Pa·m^3/s, then tritium leakage rate comes to 140 Ci/d from plasma exhaust system to environment. This is the most dangerous situation and not allowable. Because the plasma fractional burn-up of FEB-E design is rather low (β=2.08%), therefore, the tritium throughput is too high. We can conclude that under both normal and accidental circumstances, tritium leakage from blanket liquid lithium is not dangerous. However, designs of vacuum system and diverter pumping system should be careful.

4 Summary

From our tritium system design studies, we can draw some conclusions that the liquid lithium has good tritium breeding property. The tritium produced in lithium is in good confinement if the temperature in blanket keeps below 953 K. The required initial tritium storage to start up the FEB-E (143 MW) is at least 319g. During normal operation, there is no tritium leakage danger. The plasma burn-up fraction should be increased in order to reduce tritium throughput, hence to bring the tritium leakage from plasma exhaust and vacuum system down during normal operation. Under the accident events, the tritium leakage rate from liquid lithium into helium coolant will be getting higher. Some effective measures should be taken to deal with the accident within 24 h.

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