Experimental and Design Activity on Liquid Lithium Divertor

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Abstract. Results of lithium divertor investigation designed for 400 MW power removal under steady-state operation condition (DEMO-S project) are presented and a possibility to solve the problems of efficient condensation of evaporated lithium, of heat removal, of lithium flow in a strong magnetic field, of heat output, of tritium recovery from lithium and the others is shown. Feasibility of the lithium divertor concept has been confirmed by the results of successful tests of lithium limiter in various modifications in the T-11M tokamak experiments with hydrogen and helium plasmas and by experimental studies of lithium capillary-pore systems interacting with pulsed plasma and with electron beam in the steady state. Further directions of experimental, calculated and design studies needed for the development and substantiation of the lithium divertor concept with capillary-pore systems are analyzed.

1. Introduction

Power Fusion Reactor with high temperature coolant and extremely high stationary and peak power loads cannot be realized without application of liquid metals, new engineering and design approaches. Development of divertor is one of the most complex problems. The concept of the DEMO-S-scale Liquid Lithium Fusion Reactor [1-4] based on lithium self-cooling blanket and lithium divertor with evaporation-radiation principle of high heat load component protection with lithium capillary-pore system (CPS) as a plasma facing material can provide a successful decision of these problems. The lithium CPS’s basic advantage consists in forming a self-sustaining, self-regulating and splash-resistant lithium film on the surfaces of complex geometry at any spatial orientation by capillary force only. The design, calculations and experimental investigations are directed to the further development of the previous lithium divertor and reactor concept.

2. Divertor Design

Poloidal Lithium Divertor of a fusion reactor with DEMO-S-scale parameters (income power 400 MW) is divided into 16 (by number of divertor vessel ports) independent sectors with systems of heat removal, tritium recovery and energy conversion. Each sector consists of three sections. Each section (*FIG.1*) is a W-shaped structure with two divertor channels formed by four heat removal panels. Heat removal panels are mounted on a support frame. Geometry of channels determines the divertor volume closure (~ 0.5 in our design). Divertor has two independent lithium systems, namely, cooling system and receiving surface protection system. Heat removal from the divertor is ensured by liquid lithium circulation through pipe heat exchangers of the panels. Each panel has two layers of pipe bunches. The external one layer bunch of pipes is the incoming plasma facing and it is covered by capillary structure (CPS) 3-5 mm thick. Lithium CPS is

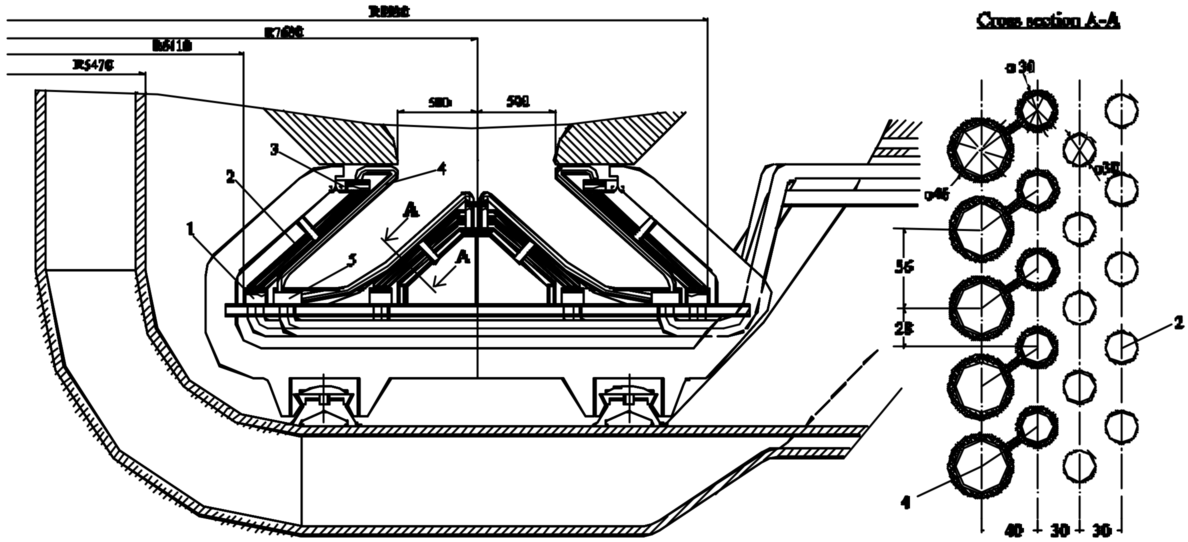

FIG.1. Lithium divertor section design option:
1-input collector, 2-channels of condensation zone, 3-intermediate collector, 4-channels of evaporation zone, 5-output collector.

intended for protection the receiving elements from energy flux and is a lithium vapor source. The internal three layers bunch of pipes is designed for lithium vapor condensation and prevention of vapor flux from the pumping system. An optional design of receiving panel cross-section is presented in *FIG. 1*. Piping configuration and arrangement exclude the direct penetration of particle and radiation flows from the SOL plasma to the lithium condensation zone, support frame etc. Such a design ensures a needed aperture for gas pumping from a divertor volume and for lithium vapor coming to condensation zone. The receiving CPS and the elements of condensation zone are interconnected by CPS inserts for reduction of liquid lithium circulation path in the protection system. Lithium coolant in each panel (*FIG.1*) circulates in identical two-pass separate loops: from inlet collector 1 to condensation zone pipes 2, then to transitional collector 3 and then through pipes of evaporation zone 4 to the outlet collector 5. Inlet and outlet collectors of receiving panels of internal and external divertor channels are connected in pairs. Thus, lithium circulates in each section in four independent loops. Such a circuit makes it possible to control thermal and hydraulic parameters of each panel depending on heat load and to reduce the hydraulic losses to a minimum. Above the collectors 1, 3, 5 there are additional lithium tanks covered with a CPS mat for lithium feeding of a receiving panel external layer. There is a separate lithium supply channel for the additional tanks. All three sections of a sector are incorporated in a common liquid metal system (*FIG.2.*) and connected to external part of service and process system of the sector through divertor port. Service and process system of sector includes lithium loop with auxilliary systems and tritium recovery device (TRD), intermediate heat transfer loop with Na-K eutectic, steam generator and steam turbine generator. The intermediate Na-K loop application allowed for solving the reliability and safety problems owing to a low tritium solubility in Na-K, in-vessel water absence, great experience on Na-K/water heat exchanger operation and liquid metal system technology well developed. In the concept of liquid lithium reactor this system must be common for the appropriate sectors of blanket and divertor. The analysis shows that increasing the output lithium temperature to a really allowable level of the blanket (750-770^{o}C) and divertor (650^{o}C) provides the usage of removed heat for the recovery of tritium from lithium up to ~ 1 appm by nonequilibrium distillation without additional

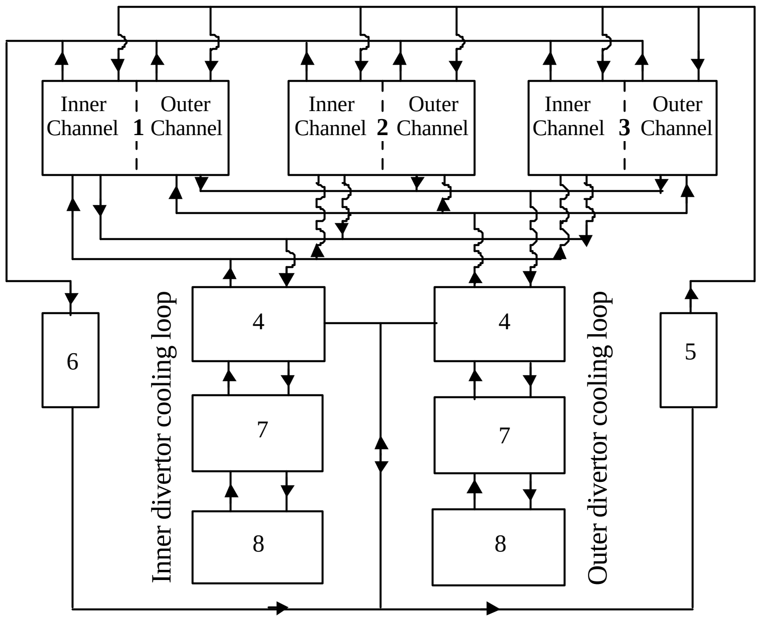

FIG. 2. Heat removing and conversion system scheme of lithium divertor sector 1, 2, 3 - divertor section; 4- lithium cooling channel and tritium recovery system , 5-lithium supply channel of CPS; 6- lithium output channel of CPS; 7-Na-K cooling loop; 8- conversion system.

energy expenses in TRD. In this case it will ensure the tritium recovery for 1% of total lithium flow in the system over the process temperature range of 640-670^{o}C. Furthermore, the high potential heat of lithium from the TRD output and TRD condensers is transferred to energy conversion system by the intermediate Na-K loop. Such approach enables the fusion cycle heat-to-electricity conversion with increased net efficiency up to ~ 36 %. The divertor operating parameters estimations are given in Table. 1.

TAB. I. BASIC PARAMETERS OF DIVERTOR LIQUID METAL SYSTEM

Parameter	Internal channel	External channel
Divertor heat power, MW	140	260
Sector heat power, MW	8.75	16.25
Section heat power, MW	2.92	5.42
Lithium flow rate in sector, kg / s	9.7	10.56
Na-K flow rate in intermediate loop, kg / s	28.44	52.82
Lithium temperature in divertor, in/out, ^{o}C	300 / 650	300 / 650
Na-K temperature in intermediate loop, in/out, ^{o}C	270 / 620	270 / 620
Specific heat load, MW/m^2	1.46	2.25
Lithium temperature in section, ^{o}C In / out of condensation zone In / out of evaporation zone	 300/315 315/650	 300/320 320/650
Maximal temperature of CPS surface, ^{o}C	735	770
Lithium flow velocity in channels, m/s	~ 0.1	~ 0.2

Application of self healing electrical insulating coating on the internal surface of channels is required to reduce MHD flow resistance in the liquid metal systems in a tokamak reactor. The technology developed allows the lithium-resistant self healing AlN-based coating ~ 5 μm thick with 50 $\Omega \cdot m$ resistance to be applied [4]. Calculations have shown that at lithium velocity of ~ 0.1-0.2 m/s the hydraulic resistance in the coated divertor channels does not exceed a technically acceptable value of $2 \cdot 10^5$ Pa. The structural material of divertor in-vessel elements and CPS is low activated and compatible with liquid lithium vanadium alloy of V-Cr-Ti system.

3. Experimental Studies

The lithium CPS application as a plasma facing material for divertor receiving elements and reactor chamber first wall makes it possible to realize the idea of «radiating» divertor based on the transfer of main energy flux from plasma to the chamber wall and the surface of divertor receiving elements by the radiation of lithium neutrals and ions in SOL without essential increasing of Z_{eff}. The lithium is supplied to SOL from CPS surface. The calculation shows [5] that at electron temperatures (T_e 30-300 eV) characteristic for tokamak plasma periphery in H-mode with ELM's it is possible to increase lithium radiation by 2-3 orders in comparison with stationary case from non-stationary ionized states. For example, for ITER reactor with a thickness of radiating layer of 10 cm, $n_e = 2 \cdot 10^{19}$ m^{-3}, $n_{Li} = 1 \cdot 10^{19}$ m^{-3}, life time $\tau = 1 \cdot 10^{-3}$ s it is possible to radiate 100 MW of power. The confinement parameter $n_e \cdot \tau$ (n_e - electronic density) may become low itself as a result of the type III ELM formation or it may be set by stochastization of periphery (ergodic divertor), or by setting up a special limiter. The possibility of the radiation up to 50 % of total energy flux from plasma has been experimentally proved on the tokamak T-11M with lithium CPS based limiter [5,6]. Thermoemission was the main source of lithium influx to plasma. Besides, the experiments with helium plasma showed that lithium thermoemission increased while limiter temperature increased that, in turn, provided reduction of the heat flow to the limiter by one half [7].

The incoming plasma energy removal and its redistribution over the total divertor surface in the divertor area where $T_e << 10$ eV is supposed to be one of two very effective channels, i.e. lithium evaporation from the surfaces being in direct contact with plasma and subsequent its condensation; lithium atoms radiation in the divertor volume. The energy fraction removed via each of them depends on particular divertor design. Thus, an acceptable mode of heat removal (at a level of ~ 0.1-0.5 MW/m^2) by heat conduction to liquid metal cooling system and further to conversion systems may be realized.

Good erosion and damage resistance of lithium CPS has been experimentally confirmed on models of a receiving element under steady-state heat load up to 50 MW/m^2 for short (several seconds) [8] and up to 5 MW/m^2 for long (3 hours when using water cooled mock-up) exposure periods. In the latter case the CPS surface temperature did not exceed 540 ^oC, the lithium losses via evaporation reached 0.1 kg/m^2s, the energy fraction removed by evaporation was ~ 50%. Total lithium amount evaporated from CPS surface during these experiments was 0.8 kg but no decrease of CPS serviceability was observed. Stability of Li CPS under tokamak T-11M plasma impact at heat flux up to 10 MW/m^2 ($n_e = 1 \cdot 10^{19}$ m^{-3}) and in total disruption conditions at initial surface temperature up to 400^oC has been experimentally confirmed. Moreover, the absence of abnormal lithium erosion and of lithium deposition on diagnostic windows has been shown, the possibility to suppress lithium splashing for special CPS design has been proved. Total damage resistance of lithium CPS due to

lithium vapor screening effect in disruption modeling experiments in the plasma accelerator with energy flux up to 5 MJ/m^2 [9] and in «plasma focus » device at 1 MJ/m^2 has been shown.
Furthermore, an effect of sorption uptake both of hydrogen and of helium by lithium surface during tokamak discharge has been registered in T-11M experiments. Helium and hydrogen desorption has been observed in the temperature ranges of 50-100^{o}C and 350-400^{o}C, respectively. It means that hydrogen does not form hydrides (lithium hydride decomposes at 600-700^{o}C). This results confirms the idea of gas separation in lithium fusion reactor [4].

4. Conclusion

Concept and design of lithium «radiating» divertor for DEMO-S -scale reactor with power coming into divertor of 400 MW on the basis of capillary - pore systems which solve energy removal and conversion problems with a net cycle efficiency of 36% have been analyzed. Lithium CPS has been found to provide self-restoring and splash-resistant lithium surface effectively protecting the divertor receiving elements from damage by interaction with high density energy flux under stationary and transition operation modes and under plasma disruptions. Methods to improve energy removal efficiency from plasma by radiation through lithium nonstationary ionized state have been specified. Reradiation of 50% of energy flux from plasma to limiter have been shown experimentally. Helium and hydrogen sorption on the lithium surface have been detected under tokamak condition. The lithium hydride formation in T-11M experiments has not been detected. The gas separation possibility in lithium tokamak reactor has been confirmed. Mechanisms of lithium influx to plasma have been considered. Lithium CPS advantages and compatibility of lithium with plasma should be experimentally confirmed in large tokamak devices.

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