Liquid lithium loop system to solve challenging technology issues for fusion power plant*

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Steady-state fusion power plant designs present major divertor technology challenges, including high divertor heat flux both in steady-state and during transients. In addition to these concerns, there are the unresolved technology issues of long term dust accumulation and associated tritium inventory and safety issues [1]. It has been suggested that radiation-based liquid lithium (LL) divertor concepts with a modest lithium-loop could provide a possible solution for these outstanding fusion reactor technology issues, while potentially improving the reactor plasma performance [2, 3]. The application of lithium (Li) in NSTX resulted in improved H-mode confinement, H-mode power threshold reduction, and reduction in the divertor peak heat flux while maintaining essentially Li-free core plasma operation even during H-modes. These promising results in NSTX and related modeling calculations motivated the radiative liquid lithium divertor (RLLD) concept [2] and its variant, the active liquid lithium divertor concept (ARLLD) [3], taking advantage of the enhanced Li radiation in relatively poorly confined divertor plasmas. It was estimated that only a few moles/sec of lithium injection would be needed to significantly reduce the divertor heat flux in a tokamak fusion power plant. By operating at lower temperatures (≤ 400°C) than the first wall (~ 600 – 700°C), the LL-covered divertor chamber wall surfaces can serve as an effective particle pump, as impurities generally migrate toward lower temperature LL divertor surfaces.

To maintain the LL purity in a 1 GW-electric class fusion power plant, a closed LL loop system with a modest circulating capacity of ~ 1 liter/second (l/sec) is envisioned as shown in Fig. 1 [2]. We examined two key technology issues: 1) dust or solid particle removal and 2) real time recovery of tritium from LL while keeping the tritium inventory level to an acceptable level. By running the LL-loop continuously, it can carry the dust particles and impurities generated in the vacuum vessel to the outside where the dust / impurities can be removed by relatively simple dust filter and cold/hot trap systems. In a 1 GW-electric (or ~ 3 GW fusion power) fusion power plant, about 0.5 g / sec of tritium is needed to maintain the fusion fuel cycle assuming ~ 1 % fusion burn efficiency. Using a cold trap system, it appears to be feasible to recover tritium (T) in real time fromLL while maintaining an acceptable T inventory level.

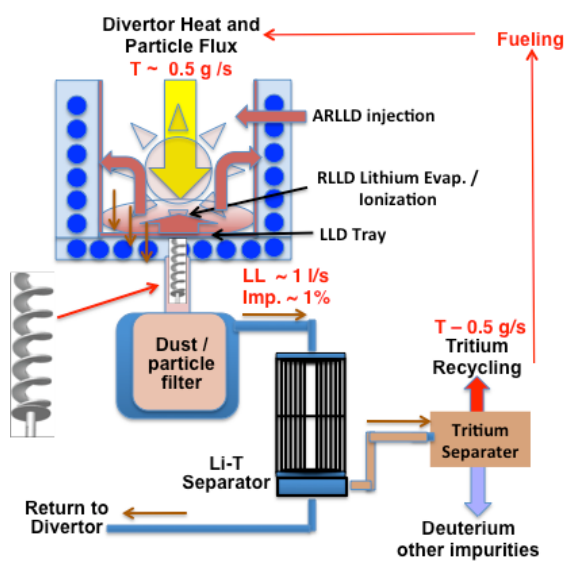

Fig. 1. A simplified schematic for the LL purification loop in a 1GW-e fusion power plant.

A modest LL loop can collect and carry away considerable amount of the dust generated within the fusion chamber. With a 1 l/sec LL flow, even a small 0.1% dust content by weight (or 0.5 g per sec) means that the LL-loop could carry away nearly 16 tons of dust per year. As shown in Fig. 1, with the dust filter located immediately below the divertor chamber, the LL should flow down into the dust filter mostly by gravity, but it may be advisable to devise an additional means of moving LL into the dust filter. This can be done, for example, by a slowly moving screw mechanism to facilitate the movement of LL from the divertor exit into the dust filter as illustrated in Fig. 1. Alternatively, a j x B force driven mechanism can move LL within the divertor toward the LL exits. One could remove heavier and larger size dust particles by letting the dust to settle to he bottom of the trap, reducing the burden on the dust filter. One would envision several LL loop exits and dust filters distributed toroidally around the torus to insure that at least one of the filters operates at any given time, so that the filled dust filters can be removed without stopping the LL flow. The dust filter could also filter any solidified Li compounds formed within the vacuum vessel in addition to the metal-based dust. Since the dust surfaces could be trapping tritium, the dust filter should be periodically drained of LL and heated to release any trapped tritium. The released gas can be sent to a tritium separation system to recover tritium for fuel recycling. After the dust filter is filled, it must be replaced. Since the location of the dust filter is relatively close to the fusion chamber, the dust filter replacement must be done remotely. Once the LL is filtered to be free of dust and other solid materials, it should flow easily and only require small diameter pipes. The small pipe diameter will help minimize the LL-loop volume and reduce the tritium inventory. The filtered LL will be then sent to cold traps operating around 200°C for tritium removal before sent back to the divertor as shown in Fig. 1,

To separate Li-T (tritiated lithium hydride) from LL, one could envision a cold trap approach taking advantage of an order of magnitude reduction in Li-T solubility in LL at 200°C compared to 400°C. We should note that while our present focus is on the Li-T removal, there is an equal amount of Li-D (deuterated lithium hydride) and the same removal process should apply equally to Li-D since their physical and chemical properties are very similar. While the conventional cold-trap (CT) may provide an acceptable tritium extraction method for the LL-loop for a fusion power plant [2, 4], a new type of CT termed the Surface Cold-Trap (SCT) is proposed here to improve the tritium recovery time, mitigate the tritium inventory issue, and increase CT safety and reliability. While the conventional CT is fully filled with LL, only a thin layer of LL in the proposed SCT covers the Li-T collection surfaces, the Li-T capture probability should be sufficient high for the SCT system. Moreover, because of the very thin LL thickness, the total LL volume within the SCT can be an order of magnitude smaller than the conventional CT volume of ~ 100 l, thus, reducing the tritium inventory level to only ~ 0.1 kg compared to ~ 1 kg in the conventional CT system. Once certain amount of Li-T is accumulated in SCT, LL is drained from SCT and the accumulated Li-T can be removed by heating to the melting temperature of Li-T ~ 700 °C. One can also envision vibrating the collection plates to shake the Li-T loose from the plates. During the Li-T removal, the LL flow would need to be diverted to another SCT. An alternate Li-T separation approach proposed here is a centrifugal-based separator taking advantage of the significantly higher mass density of Li-T (~ 1 g/cc) compared to LL (~0.5g/cc). In this case, one may choose to make the outer region of the separator (where Li-T accumulates) deliberately turbulent by applying magnetic perturbation to prevent deposition of Li-T on the outer wall. Enriched Li-T can be then channeled out of the separator. An advantage of the centrifugal method is that the Li-T can be extracted continuously. Once Li-T is isolated, T can be regenerated by various methods including heating to the Li-T dissociation temperature (~ 900 °C), laser-based evaporation, or chemical-based (e.g. CO_2) regeneration. As noted in Fig. 1, after regeneration, the tritium is separated from D-T mixture and other impurities in the tritium separator.

In NSTX-U [4, 5], preparations are now underway to elucidate the physics of Li plasma interactions with a number of Li application tools and Li radiation spectroscopic instruments. The NSTX-U Li evaporators, which provides Li coatings over the lower divertor plate, can offer important information on the RLLD concept, and the Li granule injector will test some of the key physics issues for the ARLLD concept. A LL-loop is also being prepared offline for prototyping future use on NSTX-U.

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