Liquid Lithium Loop System to Solve Challenging Technology Issues for Fusion Power Plant*

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Abstract. 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. 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. 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 and its variant, the active liquid lithium divertor concept (ARLLD), taking advantage of the enhanced Li radiation in relatively poorly confined divertor plasmas. 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. 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, cold trap and/or centrifugal separation systems. With ~ 1 l/sec LL flow, even a small 0.1% dust content by weight (or 0.5 g per sec) suggests that the LL-loop could carry away nearly 16 tons of dust per year. 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. It appears feasible to recover tritium (T) in real time from LL while maintaining an acceptable T inventory level. Laboratory tests are being conducted to investigate T recover feasibility with the surface cold trap (SCT) concept.

1. Introduction

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 serious 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 [4], H-mode power threshold reduction [5], and reduction in the divertor peak heat flux [6] while maintaining essentially Li-free core plasma operation even during H-modes [7]. A very thin sub-millimeter thick Li coating on the sub-millimeter molybdenum sprayed surfaces of the LLD (Liquid Lithium Divertor) in

NSTX protected the delicate LLD surfaces and no LLD surface metallic material (moly) was observed in the plasma during plasma operations [8]. 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 tritium (T) and deuterium (D) and impurities generally migrate toward lower temperature LL divertor surfaces and combine efficiently with lithium to form lithium-hydride and other lithium compounds. In this paper, we explore the possibility of using a modest LL-loop system to remove dust and T from the fusion reactor chamber in real time. Perhaps the most technically challenging task for the LL-loop is how to remove tritium from the LL in real time. We propose two concepts for tritium removal, a a surface cold trap system taking advantage of the large solubility change with LL temperature and a centrifuge system taking advantage of the factor of two difference in the specific gravity of LL and LiT. In Sec. 2, we give a brief summary of the RLLD and ARLLD concepts. In Sec. 3, a candidate LL-loop system is described. In Sec. 4, a dust removal concept is discussed. In Sec. 5, two concepts, surface cold trap and centrifugal based systems for T extraction, are proposed. An estimate of overall T inventory in the LL system is discussed in Sec. 6. In Sec. 7, the conclusions and discussions are presented.

2. Review of Radiative Liquid Lithium Divertor Concepts

The Radiative Liquid Lithium Divertor / Active Radiative Liquid Lithium Divertor (RLLD/ARLLD) concepts were proposed to reduce the divertor heat flux via radiation of injected lithium in the divertor plasma as shown in Fig. 1. The RLLD/ARLLD are placed at the bottom of the reactor chamber for obvious reasons from the LL handling point of view, and also to capture any impurity particles including dust generated within the reactor chamber as illustrated in Fig. 1. The LL is introduced at the upper part of the RLLD at multiple toroidal locations, and the LL flows down the RLLD side wall as a thin film via gravity and capillary action. The thin LL film thus formed should provide very effective pumping (or entrapment) of the working gases, impurities, and dust generated within the reactor chamber. The RLLD chamber being at the lowest temperature in the reactor chamber, together with the usual divertor action, should facilitate the pumping of the entire reactor chamber. It should be noted that since the envisioned divertor chamber wall surface area in the fusion power plant can be quite large i.e., ~ 100 m^2, the flowing LL film of ~ 0.2 mm for the entire wall would only amount to be 20 l of LL. As shown in Fig. 1, the RLLD chamber wall temperature can be in the 250 – 400 °C range, which is significantly lower than that envisioned for the fusion reactor first wall at ~ 600 – 700°C. The hot reactor first wall should be able to keep the wall surfaces clean from the working gas and also the impurities including Li and Li-related compounds [9]. The LL flowing down the divertor side wall accumulates at the bottom of RLLD where the divertor strike point is placed. By placing the LL surface in the path of the divertor strike point, the LL is evaporated from the surface through sputtering, evaporation, and chemical processes [10]. The evaporated Li is quickly ionized by the plasma and the ionized Li ions can radiate strongly, reducing the heat flux to the divertor strike point surfaces and protecting the substrate material. Perhaps the last line of defense is the LL evaporation from the LLD tray. Through evaporation, Li can carry some heat away from the material surfaces analogous to the way the latent heat of vaporization clamps the temperature rise. The evaporated Li could also form a Li vapor cloud in front of the divertor surface and provide some additional protection [11]. The ARLLD concept [3] which is based on active injection of lithium closer to the divertor entrance has the advantage of inducing radiative loss well away from the divertor plate, thus improving the chance of spreading the heat more evenly throughout the divertor chamber wall. Active Li injection from the divertor side wall also has the advantage of a relatively narrow divertor plasma channel (short radial travel distance) for Li delivery. The Li therefore can be delivered to the plasma quite rapidly, i.e., ~ 1 msec. Since the particle confinement time of injected Li is estimated to be ≤ 1 msec even for DEMO parameters, the ARLLD overall response time maybe only ≤ a few msec, which should be fast enough to protect the divertor PFCs from transient events.

3. Liquid Lithium Loop System

To remove dust / impurities and working gases (i.e., deuterium and tritium), a modest LL circulating loop of ~ 1 l /sec was proposed as illustrated in Fig. 1 [2]. This relatively modest level of LL circulation ensures timely removal of generated dust and impurities, including tritium, while keeping the LL purity to be sufficient for smooth LL flow. It should be noted that the LL flow rate of 1 l/sec is much larger than that required to reduce the heat flux via the RLLD and ARLLD. Therefore most of the circulating LL can be utilized to coat the divertor side wall to provide sufficient pumping for the reactor system. The circulating LL system can also remove dust generated in the reactor chamber which if unchecked can lead to serious T inventory and reactor safety issues. We should note that a similar LL purification loop has been developed for facilities related to the International Fusion Materials Irradiation Facility (IFMIF) [12]. We envision the ARLLD / RLLD LL purification loop for a power plant to have a ~1000 liter capacity, which is an order of magnitude smaller than the LL-loop envisioned for IFMIF. The relatively low operating temperature range of the RLLD and its associated LL loop system of ≤ 400 °C is advantageous from the material corrosion and safety point of view. The low operating temperature also makes available broader choices of steel-based alloy materials which might be more practical to employ as a divertor-LL substrate and loop material that is compatible with a reactor environment.

4. Dust Removal System

A modest LL loop can collect and carry away considerable amount of the dust generated within the fusion chamber [1]. 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 which maybe similar to the expected PFC erosion rates. 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. Alternatively, a j x B force driven mechanism can move LL within the divertor toward the LL exits [13]. There is also thermoelectric based mechanism for moving LL [14]. One could remove heavier and larger size dust particles by letting the dust to settle to the 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 conventional 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, at ~ 400°C, 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 the tritium removal system described in the next section operating around 200°C for tritium removal before sent back to the divertor as shown in Fig. 1.

5. Tritium Removal by Cold Trap and Centrifuge

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 [16]. 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, 3], 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. 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). Once Li-T is isolated, T can be regenerated by heating Li-T to release temperature which is estimated to be 400 – 500 °C. As noted in Fig. 1, after regeneration, the gaseous tritium is separated from D-T mixture and other impurities in a conventional tritium separator.

The surface cold trap (SCT) system for tritium removal – For removing T from LL, a cold trap (CT) system appears to be quite energy efficient (particularly with a heat exchanger) since LL is only required to cool down to about 200°C. The CT uses the property of large changes in the solubility of T in LL with temperature [15]. At 400 °C, the solubility is ~ 0.5 % by weight, while the solubility goes down by more than an order of magnitude to ~ 0.02% at 200 °C. This solubility difference gives the opportunity to extract tritium from LL in the CT. A new type of CT termed Surface Cold-Trap (SCT) is proposed here to improve the tritium recovery time, reduce the tritium inventory issue, and increase the LL-loop system safety and reliability as illustrated in Fig. 2. While the conventional CT fills the CT volumetrically with LL, in the present SCT, LL covers only the surfaces and thus greatly reducing the LL volume. The SCT consists of a number of concentric steel-based metal cylinders or a series of rectangular flat plates within the container. The LL enters from the top of the SCT. There are number of holes or slits which would allow the LL to flow on to the top of the shells or flat plates. The LL then flows down the vertical surfaces of the cylindrical or flat plates and exits at the bottom of the SCT. For example, a SCT with 1 m wide and 1 m long cylindrical shells placed in 1 cm radial increment would provide about 100 m^2 Li-T collection surface area. Alternatively one can envision a series of rectangular plates with 1cm separation in ~ 1 cubic meter container, again providing about 100 m^2 collection area. The LL would then flow down the vertical surfaces and can be modelled by well known incompressible viscous fluid flow equation. With LL flow rate of 1 l / sec in a typical ~ 1 m cube size SCT, one can show that the LL flow layer thickness is only ~ 0.14 mm with ~ 10 cm/sec flow speed. If more SCT units are used in parallel, the LL flow layer thickness is further reduced and the flow speed slowed to further facilitate LiT collection. With an estimated LiT diffusivity (from the Stokes and Einstein equation) of ~ 0.64×10^{-4} cm^2/sec at 200 °C, the characteristic diffusion time is ~ 3 sec through a LL layer of ~ 0.14 mm thickness, which is fast enough compared to the LL transit time of ~ 10 sec. Perhaps more importantly, due to the very thin LL layer thickness, the downward LL flowing motion itself is likely to cause dynamic mixing of LiT as it flows down the SCT plate surfaces, facilitating LiT collection. To facilitate crystallization on its collection surfaces, the SCT plate surface maybe roughed, for example, with sand blasting. It should be noted that it is only necessary for the SCT to capture a fraction of the passing LiT. With 1 l / sec flow with 0.5% T or 2.5 g /s of T flow, only a 20% capture fraction is necessary to recover 0.5 g of T in real time. Moreover, because of the very thin LL thickness ~ of 0.2 mm, the total LL volume within the SCT for this size is only ~ 20 l. As the SCT provides a large surface area of ~ 100 m^2 in a compact volume of ~ 1 m^3 yet only contains relatively modest amount of LL of ~ 15 l, one can for example envision running as many as 10 SCT units in parallel and switch off some of the units for tritium regeneration as described below. If 10 SCT unites were used in parallel, the total LL volume in the SCT units would be about 73 l.

Once a SCT is ready for tritium regeneration, the LL flow is switched off and the remaining LL drained, leaving thin LiT layers covering the surfaces of SCT plates. The LL draining from the SCT is rapid, since the LL has to only flow down the vertical smooth steel wall of 1 m length. Even if the time required to drain most of the LL is assumed to be 20 times the LL transit time of 15 sec, it is still only about 300 sec. The SCT can be then heated to 400 - 500°C to release tritium and deuterium. Since the LiT (or LiD) layer is very thin, the release of T (or D) should occur at a relatively modest elevated temperature of 400 – 500 °C, as shown in lab tests assuming that adequate pumping is provided to remove the released T and D [9. 16]. The present simple SCT design with no moving parts, therefore, should be well suited for repeated regeneration cycle, while minimizing the tritium inventory issue.

The centrifugal system for tritium removal – The centrifugal Li-T separation approach takes advantage of the significantly higher mass density of Li-T (~ 1 g/cc) compared to LL (~0.5g/cc). Enriched Li-T can be then channeled out of the separator as shown in Fig. 3. An advantage of the centrifugal method is that the Li-T can be extracted continuously. The unit however does have moving parts and needs to be located well away from the magnetic field. Commercial centrifuge units handle fluid flow rates of 20 l /sec, which is an order of magnitude larger than the present system requirement. As the supersaturated Li-T is removed from the centrifuge, it is important to prevent LiT crystallization on surfaces of the centrifuge and exit pipes. In this case, one may choose to make the outer region of the centrifuge (where heavier Li-T accumulates) deliberately turbulent by applying magnetic perturbations to prevent deposition of Li-T on the outer wall. One should also choose a wall material which is smooth and slippery to LiT, so the LiT would not stick and crystalize on the wall surfaces.

6. Tritium (T) Inventory of the LL system

With a LL-loop system, it is important to consider the T inventory issue for such a system. It is clearly desirable to minimize the T site inventory. It is assumed that the LL volume inside the fusion reactor chamber is about 100 l. This is relatively small since the thickness of the LL film for particle pumping purposes can be very thin, i.e., ~ 0.2 mm so that the divertor area of ~ 100 m^2 can be covered with only about 20 l of LL. For the divertor strike point area for handling heat flux, the LL thickness can be ~ 1 mm. With the strike point area ~ 10 m^2, a 1mm thick film is only about 10 l of LL per SCT. For the strike point area, it is important to keep the surface covered with LL to protect the solid divertor substrate. The actual equivalent volume of LL within the plasma is negligible, bring at most ~ 1 l for the RLLD and ARLLD. We therefore estimate 100 l to be more than sufficient for the in-vessel LL volume. We then assume that the LL-loop system up to the entrance of the cold trap volume is 500 l, including the dust filters and connecting pipes. The surface cold trap has only 10 l per unit and the pipes back to the vacuum vessel, including the circulating pumps and reserve tanks, is assumed to be another 500 l. The total lithium volume is therefore estimated to be 1100 l. If the LL volume turns out to be larger or smaller than 1100 l, the tritium inventory level will scale accordingly. Assuming a LL flow rate of 1 l / sec, then to recover 0.5 g/sec for tritium in real time for a steady state 1 GW-electric fusion power plant operation, the T concentration must be reduced by 0.1% by the cold trap. In Table 1, for the 1 l/sec LL-loop system, the T inventory as a function of the reactor chamber T concentration in % is shown to be from 0.3% to 0.7%. The T consumption rate of 400g per day is assumed with a 1 % fusion burn rate. Naturally, the T inventory increases with the in-chamber T concentration, but even the T inventory of 9 days is generally considered to be acceptable. The T inventory depends only weakly on the LL flow rate. It should be noted that the T collection is taking place not only in the cold traps but also in the dust filters. As will be discussed in Sec. 7, the T collection should also be taking place in the entire LL-loop system through the double wall configuration. It is important to make sure that T inventory and transport is well understood and controlled throughout the LL system.

T % in VV	T(kg)	Days
0.7	3.6	9
0.6	3.05	7.6
0.5	2.5	6.3
0.4	1.95	4.9
0.3	1.4	3.5

Table 1. Tritium concentration and tritium inventory. The Days are assuming 400g tritium consumption per day.

7. Conclusions and Discussions

In previous publications, we described the radiative lithium based divertor concepts (RLLD and ARLLD) to solve divertor heat flux issues while improving the plasma performance of fusion reactors. In order to support the RLLD and ARLLD, we proposed a relatively modest LL loop system operating at ~ 1 l/sec. In this paper, we examined the utility of the LL-loop in a fusion power plant including dust, impurity, and T recovery. The T recovery from LL is particularly of concern, since timely recovery of T is crucial to support the T fuel cycle and maintain the T inventory to an acceptable level. The previous CT used in the IFMIF-related LL-loop facility was mainly used to extract oxygen, since the cold trap does not work for very low T concentration, i.e., < 0.02% of LL which is the case of IFMIF. By operating the T concentration well above the CT limit, the CT can be used to recover T for the RLLD/ARLLD LL-loop. In this paper, we proposed a new type of CT, termed SCT (surface cold trap), to extract tritium from the surface of the collection plate, rather than volumetrically for the case of the conventional CT. The SCT has the advantage of reducing the LL volume and facilitate T recovery. Because of its simple and durable mechanical design, the SCT can be recycled multiple thermal cycles as needed for tritium recovery. We also proposed another innovative tritium removal concept, using centrifuge technology, as LiT is twice as dense as LL. In this case, we need to ensure that the precipitated LiT would not stick and accumulate on the centrifuge wall and associated pipes. We may also note that dust removal using LL could be an important channel for T recovery, as the dust particles could contain significant amounts of T. If the amount of generated dust turns out to be large, the T released from dust regeneration may be sufficient to recover most of the T needed for the T fuel cycle, reducing the reliance on SCT. For this reason, the periodic regeneration of the dust filter for T release is an essential element of the dust filter design. We also examined the T inventory issue and concluded that the inventory is likely to be acceptable for a LL-loop system with about a 1100 l LL capacity, even with the upper limit of ~ 0.7% for T concentration and yielding 3.6 kg or 9 days of T inventory. In terms of LL safety, it is important to operate the LL system at or below ~ 400 °C to reduce long term corrosion issues, especially since corrosion could accelerate at higher temperatures. Another important consideration is a double walled configuration with an evacuated outer layer. The vacuum layer provides good thermal insulation to keep the LL hot. It also helps detect LL leaks relatively quickly and provides a LL safety barrier. Quite importantly, the vacuum layer could also provide a means of recovering T which could be diffused through the hot LL pipe wall, contributing to T inventory control.

As the NSTX-U device is starting its operation with various lithium tools and related diagnostic systems, it is now possible to investigate pertinent physics issues related to radiative liquid lithium divertor concepts [17, 18]. The NSTX-U Li evaporator system, which provides Li coatings over the lower divertor plate, can offer important information on the RLLD concept, and the Li granule injector [19] will test some of the key physics issue for the ARLLD concept. In particular, the actual lithium radiation level achievable in edge and divertor plasmas per injected lithium particles [20, 21] is critical to better quantify the amount of lithium needed to reduce the divertor heat flux to an acceptable level. A LL-loop is also being prepared off line for prototyping future use on NSTX-U. A manageable aspect of the LL-loop development is that the required R&Ds can be performed with a relatively modest laboratory setting, where various aspects of the LL-loop such as the dust filter, surface cold trap, tritium recovery (using hydrogen), etc., can be performed separately.

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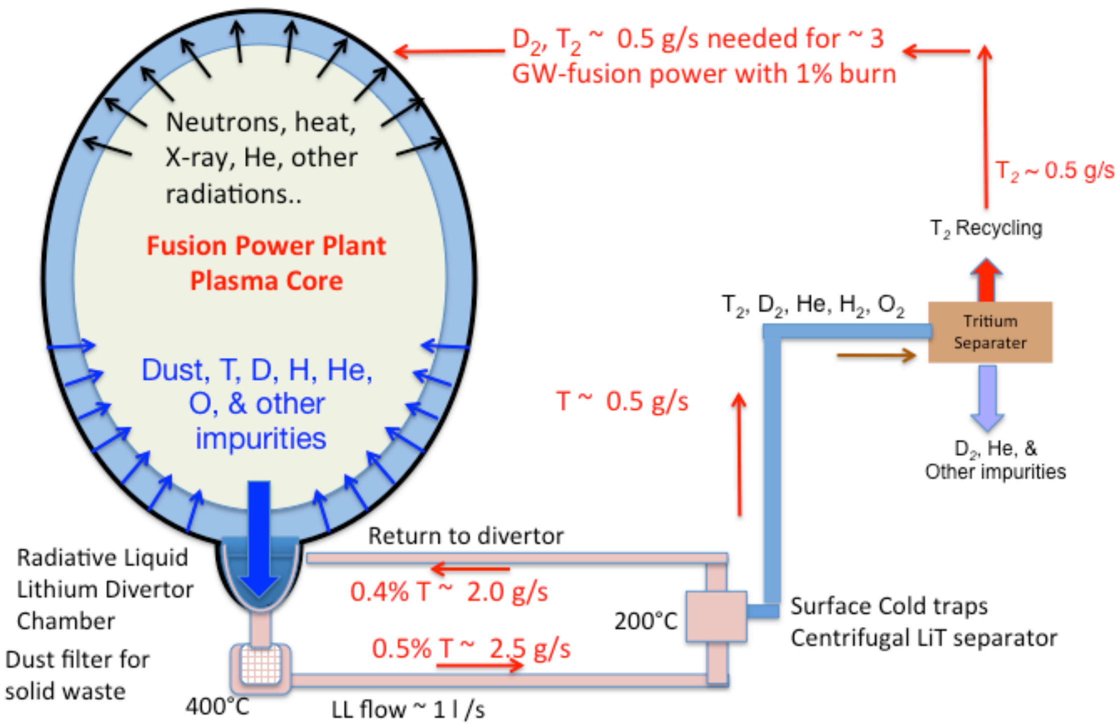

Fig. 1. A simplified schematic for the LL purification loop for RLLD/ARLLD in a fusion power plant.

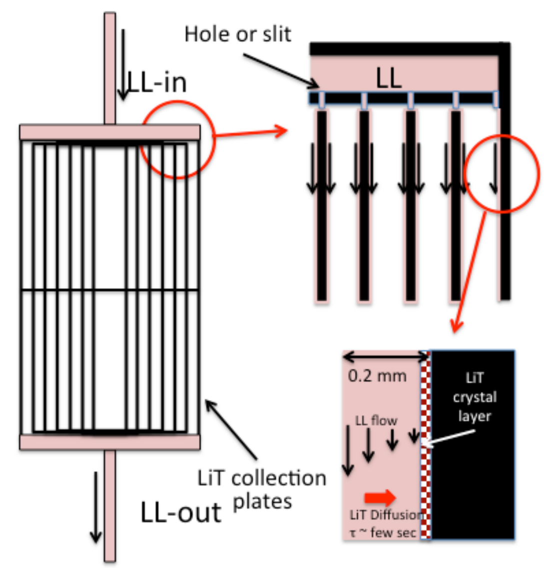

Fig. 2. A schematic of the proposed surface cold trap (SCT). Liquid lithium flow down the surfaces of the plates as depicted in the upper right insert..

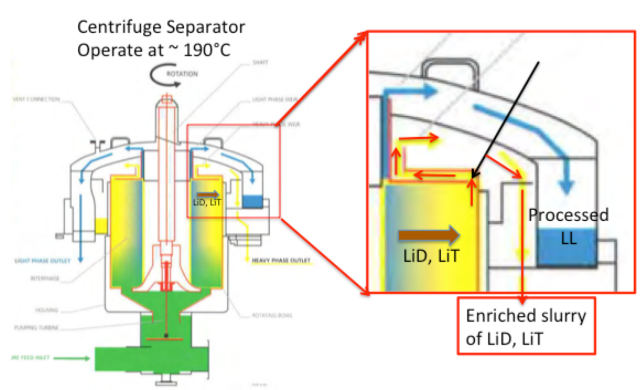

Fig. 3. A schematic of the proposed centrifuge LiT separator.