PAPER

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To cite this article: M. Ono *et al* 2017 *Nucl. Fusion* **57** 116056

IOP Publishing | International Atomic Energy Agency Nuclear Fusion

Nucl. Fusion **57** (2017) 116056 (9pp) https://doi.org/10.1088/1741-4326/aa7f41

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

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Received 5 January 2017, revised 10 May 2017
Accepted for publication 12 July 2017
Published 30 August 2017

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 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 concept and its variant, the active liquid lithium divertor concept, taking advantage of the enhanced or non-coronal 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 l s^{-1} 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 s^{-1} LL flow, even a small 0.1% dust content by weight (or 0.5 g s^{-1}) 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 s^{-1} 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 recovery feasibility with the surface cold trap concept.

Keywords: lithium, divertor, fusion power plant, dust removal, tritium recovery

(Some figures may appear in colour only in the online journal)

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 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 Li coating on the 0.17 mm molybdenum sprayed surfaces of the LLD (liquid lithium divertor) in NSTX was used to protect the LLD surfaces. The LLD surface material was molybdenum plasma sprayed with 45% porosity onto a protective barrier of 0.25 mm stainless steel liner that is bonded to a copper substrate of 2.2 cm thick. With Li coating, no LLD surface metallic material (moly) was observed in the plasma during plasma operations even with the divertor strike point moved onto LLD [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 non-coronal Li radiation in relatively poorly confined divertor plasmas. The non-coronal Li radiation values can be significantly enhanced (2–3 orders of magnitude) over the coronal equilibrium values if the Li ions are poorly confined and highly recycling as expected in the plasma edge and in the divertor region [9, 10]. This is because the lithium radiative loss cooling occurs mostly in the initial phase of lithium ionization process where an injected lithium atom under goes multiple ionization and excited spectroscopic states, and the radiation decreases as the lithium ion becomes fully ionized approaching the coronal equilibrium limit. It was estimated that only a few moles s^{-1} of lithium injection would be needed to significantly reduce the divertor heat flux in a tokamak fusion power plant. By operating at lower temperatures (200–400 °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 surface cold trap (SCT) system based on the large solubility change with LL temperature, and a centrifuge system taking advantage of a factor of two difference in the specific gravity of LL and LiT. In section 2, we give a brief summary of the RLLD and ARLLD concepts. In section 3, a candidate LL-loop system is described. In section 4, a dust removal concept is discussed. In section 5, two concepts, SCT and centrifugal based systems for T extraction, are proposed. An estimate of overall T inventory in the LL system and a LL clean-up procedure is discussed in section 6. In section 7, the conclusions and discussions are presented.

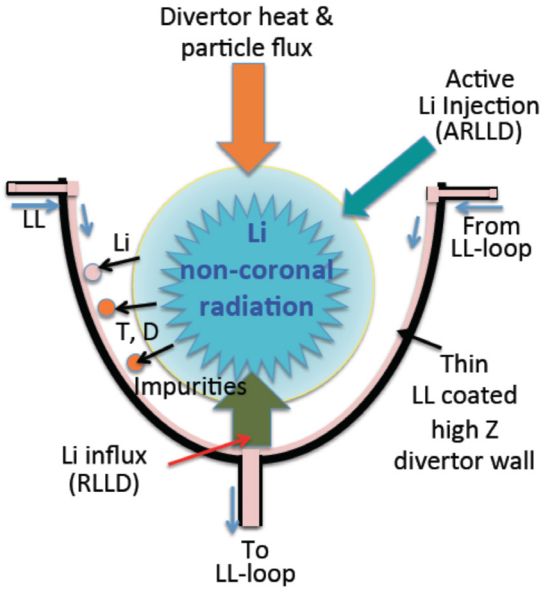

Figure 1. A schematic of RLLD/ARLLD low collisionality radiative divetor concept.

2. Review of radiative liquid lithium divertor concepts

The present RLLD concept and its variant, the ARLLD, build upon the existing quite successful radiative divertor physics and the development of high power flux high-*Z* divertor PFCs. The conventional radiative divertor concept has proven to be effective in significantly reducing the high heat flux on the divertor strike points which can cause destruction/erosion of solid high-*Z* PFCs. However, while the divertor heat reduction is achieved, plasma confinement degradation is also observed due to increased divertor recycling. The confinement degradation is a serious obstacle, since the achievable fusion gain depends very strongly on the plasma confinement. The utilization of lithium as a radiative element could reduce the divertor heat flux without increasing the recycling. The RLLD/ARLLD concepts were proposed to reduce the divertor heat flux via non-coronal radiation of lithium in the divertor plasma as shown in figure 1 [2, 3]. The schematic is simplified to illustrate the basic concept, but the actual RLLD divertor chamber shape is of course more complex and can also have a closed divertor configuration. The closed RLLD divertor may be more advantageous from the point of view of thermal and particle separation between the main fusion chamber and the divertor chamber. The thermal isolation is beneficial from overall electrical conversion efficiency considerations, and the particle separation may help reduce potential lithium flow into the main chamber. This minimizes plasma dilution and reduces overall particle recycling in the main chamber,

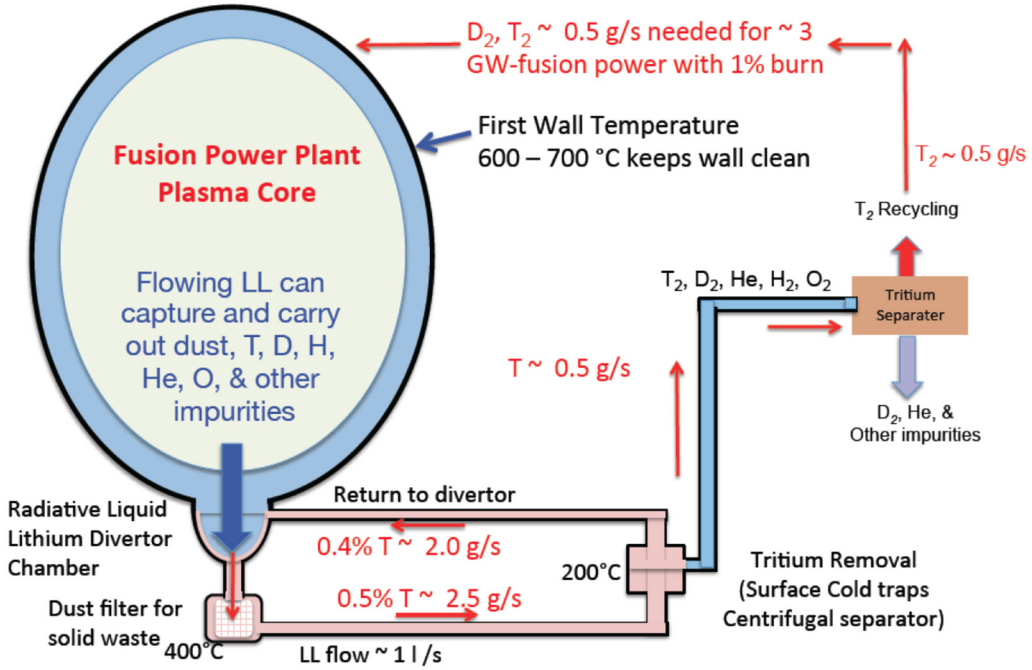

Figure 2. A schematic of a LL-loop for removing dust and tritium/impurities from the power plant vacuum and divertor chamber.

which would help improve confinement. As pointed out in [3], the closed divertor configuration may also give a convenient upstream LL injection location for depositing LL into the main divertor plasma flux impinging on the strike point for the ARLLD. The RLLD/ARLLD are placed at the bottom of the reactor chamber to be compatible with the LL handling and recycling requirement, and also to capture any impurity particles including dust generated within the reactor chamber as illustrated in figure 1. The LL flow is introduced at the upper part of the RLLD near the divertor throat area at multiple toroidal locations, and the LL flows down the RLLD side wall as a thin film via gravity 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. 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 figure 2, the RLLD chamber wall temperature can be in the 200–400 °C range. This is significantly lower than that envisioned for a fusion reactor first wall at ~600–700 °C, which should facilitate the pumping of the entire reactor chamber. The hot reactor first wall should be able to keep the wall surfaces clean from the working gas, and also impurities including Li and Li-related compounds [11]. The LL flowing down the divertor side wall accumulates at the bottom of RLLD, at the location of the divertor strike point. 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 [12]. 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 for the high-*Z* divertor PFC substrate is the LL evaporation from the LLD surfaces. 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 subsequent ionization and radiation provide powerful additional protection [13]. 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 radiative 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 as noted earlier for a closed divertor chamber. The Li, therefore, can be delivered to the plasma quite rapidly, i.e. within a few ms. Since the particle confinement time of injected Li is estimated to be $\leqslant$1 ms even for DEMO parameters, the ARLLD overall response time maybe only $\leqslant$ a few ms, which should be fast enough to protect the divertor PFCs from transient events such as ELMs. The solid high-*Z* divertor PFC designs have been developed to handle 5–10 MW m^{-2} of steady-state heat flux. Yet in a future fusion power plant, the heat flux is predicted to be much higher. The RLLD/ARLLD would reduce the divertor heat flux to an acceptable level of $<$10 MW m^{-2}, and the LL coverage would provide a renewable protective layer for the high-*Z* solid PFC surfaces from any unexpected transient heat flux which may occur even with ARLLD heat flux control. The LL temperature as it enters at the top of the divertor wall should be ~200 °C so that LL can effectively pump T and D. As it flows down the divertor wall, it may be heated by the plasma radiation, but should retain the temperature of its high-*Z* solid substrate which is actively cooled. Near the divertor-strike point, the LL temperature can be a significantly higher

due to the strong heat flux. That is acceptable, however, since some of the LL would be evaporated into the plasma at or near the strike point, and act as a renewable heat shield and divertor radiator to protect the divertor high-*Z* substrate.

3. Liquid lithium loop system

Once LL is introduced in a fusion power plant, it is necessary to recirculate LL to remove the working gas such as tritium and deuterium, but also any impurities produced within the fusion reactor. The need for the LL recirculation is true for any LL concepts being considered because of the long term accumulation issue in a steady-state system. However, with a recirculating LL system, we now have an important tool to remove the dust being generated within the vacuum vessel, which is identified as one of the most serious safety issues for steady-state fusion reactors. We envision a modest LL circulating loop of ~1 l s^{-1} as illustrated in figure 2 [2]. In a 1 GW-electric-class fusion power plant, it is typically estimated that about 0.5 g of T is injected into the vacuum vessel per second or about 40 kg of tritium per day. This rate of injection assumes about 1% fusion burn efficiency, so that about 400 g of T per day or only 0.005 g s^{-1} of T is consumed by fusion reactions to generate the required fusion power of ~3 GW. The small amount of injected T consumed by fusion reactions means that it is highly important to recover the remaining 99% of unused T in real time to maintain the fusion fuel cycle and keep the plant T inventory as low as practical. The relatively modest level of LL circulation of ~1 l s^{-1} 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 s^{-1} (or ~500 g s^{-1}) is still an order of magnitude larger than that required to reduce the heat flux via the RLLD and ARLLD [2, 3]. Therefore, most of the circulating LL can be utilized to coat the divertor side wall to provide protection for the high-*Z* substrate material, sufficient pumping for the reactor system, and a means to carry away dust and particles from the reactor chamber. We envision the ARLLD/RLLD LL purification loop for a power plant to have about ~1100 l capacity, which is an order of magnitude smaller than the LL-loop envisioned for IFMIF [14]. The relatively low operating temperature range of the RLLD and its associated LL loop system of 200–400 °C is advantageous from the material corrosion and safety point of view. The low operating temperature also makes broader choices of available steel-based alloy materials that might be more practical to employ as a divertor-LL substrate and loop material compatible with a reactor environment. Finally, the relatively low circulating LL flow rate reduces the power required to maintain the LL-circulation even with the high magnetic field environment of a fusion power plant. One can also use the non-conducting coating of the internal liner for the LL-loop to further reduce the power required for circulation. We should also note that much of the LL-loop can be located outside the main magnetic field region to minimize the power needed and also facilitate LL component maintenance.

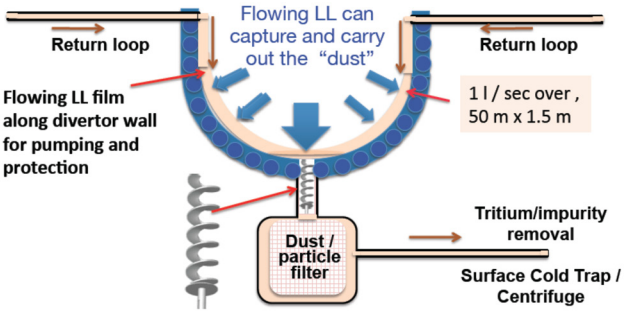

Figure 3. A schematic of dust/solid particle filter.

4. Dust removal system

With 3 GW fusion power production, it is not unreasonable to estimate that ~0.5 g s^{-1} of dust or solid particles may be generated within the vacuum vessel and divertors [1]. While 0.5 g s^{-1} does not sound alarming, the amount of dust accumulated if unchecked would be ~16 tons in a year during steady-state fusion power plant operations. With a 1 l s^{-1} LL flow, 0.5 g s^{-1} of dust represents only ~0.1% by weight of the LL flow, which is quite low. Therefore, such a LL-loop should be capable of carrying away perhaps an order of magnitude more dust/solid particles or up to 160 tons per year if needed. A modest LL-loop, therefore, can carry away a tremendous amount of dust generated integrated over time within the power plant vacuum vessel. While the precise amount of dust being generated within a fusion power plant is not known, it is wise to not let it accumulate, as a significant amount can build up in a steady-state system. The dust, because of the small size, can create enormous activated surface areas per given weight (e.g. 1 cc of dust is known to have ~1 m^2 of total surface area) and could pose serious safety risks and tritium inventory issues. It should be noted that dust generation is predicted to occur predominantly in a high heat flux environment such as the divertor strike point where material erosion, surface cracking, etc can occur. The LL surface protection of such high-*Z* PFCs proposed here should greatly reduce dust generation, but it is still wise to retain a means to remove the dust whenever and wherever generated. The first wall, for example, with neutron and radiative bombardment is also a known source of significant dust generation. As shown in figures 2 and 3, with the dust filter located 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 $j \times B$ force-driven mechanism can move LL within the divertor toward the LL exits [15]. There are also thermoelectric-based schemes for moving LL [16]. One may also choose to operate the LL flow in a 'pulsed' mode to flow much larger amounts of LL at a lower duty cycle, e.g. 1 s LL injection of 10 l s^{-1} every 10 s, to wash down the dust as needed. One could remove heavier and larger size dust particles in the LL by letting dust particles settle to the bottom of the dust filter, reducing the burden on the dust filter and enabling it to filter finer dust particles. One could envision several LL 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 regenerated or removed without stopping the LL flow. Importantly, the dust filter could also filter any solidified Li compounds, including LiT formed within the vacuum vessel, in addition to any divertor or first wall PFC-based metallic dust. Since the dust surfaces and solid lithium compounds could be trapping tritium, the dust filter should be periodically drained of LL and heated to ~600 °C to release any trapped tritium. The released gases 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, it should flow easily and only require small diameter pipes to be sent to the tritium separator system. 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 figure 2.

5. Tritium removal by cold trap and centrifuge

It is generally expected that the tritium and lithium would form Li–T (tritiated lithium hydride) in the divertor chamber and carried away by the LL-loop. One of the challenges for the LL-loop system is whether one can separate T from LL to support a ~0.5 g s^{-1} tritium fuel cycle. As noted above, if the Li–T is in a solid (non-dissolved) form, one could capture it in the dust filter and the T can be regenerated. 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. We propose here two T removal methods; one based on the T solubility dependence on temperature and the other based on a centrifuge to separate the heavier Li–T compared to LL. A cold trap approach takes advantage of the order of magnitude reduction in Li–T solubility in LL at 200 °C compared to 400 °C [17]. While the conventional cold-trap (CT) may provide an acceptable tritium extraction method for the LL-loop for a fusion power plant [2, 3, 18], a new type of CT termed the SCT is proposed here to improve the tritium recovery time, mitigate the tritium inventory issue, and increase CT reliability and maintainability. A centrifugal-based separator takes advantage of the significantly higher mass density of Li–T (~1 g cc^{-1}) compared to LL (~0.5 g cc^{-1}). Once Li–T is isolated, T can be regenerated by heating Li–T to its estimated release temperature of ~600 °C. As noted in figure 2, after regeneration, the gaseous tritium is separated from the D–T mixture and other impurities in a conventional tritium separator.

5.1. 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 which

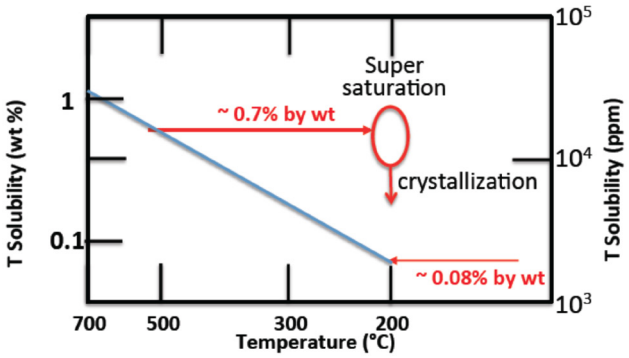

Figure 4. Tritium solubility versus LL temperature.

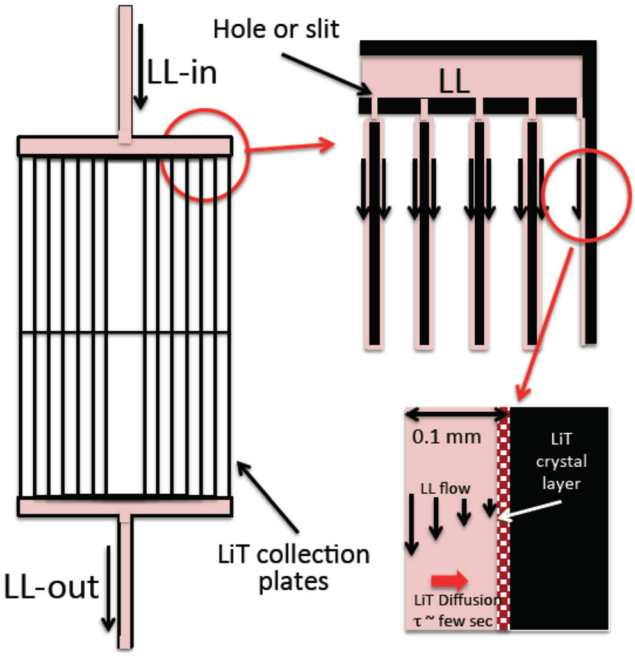

Figure 5. A schematic of the surface cold trap (SCT).

is the injection temperature into RLLD. The CT uses the property of large changes in the solubility of T in LL with temperature [17]. At 400 °C, the solubility is ~0.5% by weight, while the solubility goes down by more than an order of magnitude to ~0.08% at 200 °C as shown in figure 4. This very low solubility at 200 °C allows the extraction of T from LL in the CT. A new type of CT, termed the SCT, is proposed here to improve tritium recovery time, reduce the tritium inventory issue, and increase LL-loop system safety and reliability. While the conventional CT fills the CT volumetrically with LL [19], LL covers only the surfaces of the SCT and thus greatly reduces 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 as shown in figure 5. The LL enters from the top of the SCT. There are a number of holes or slits which would allow the LL to flow onto 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 increments would provide about a 100 m^2 Li–T collection surface area. Alternatively, one can envision a series of rectangular plates

with 1 cm separation in a ~1 cubic meter container, again providing about a 100 m^2 collection area. The LL would then flow down the vertical surfaces and can be modelled by a well known incompressible viscous fluid flow equation. With a LL flow rate of 1 l s^{-1} in a typical ~1 m cube-sized SCT, one can show that the LL flow layer thickness is only ~0.14 mm with a ~10 cm s^{-1} flow speed. As shown in the expanded schematic view of the SCT, the flow velocity goes to zero at the wall surface. Therefore, the LL spends a much greater time at the Li–T collection surfaces, thus further promoting crystallization. If more SCT units are used in parallel, the resulting LL flow layer thickness is reduced accordingly 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 s^{-1} at 200 °C, the characteristic diffusion time is ~3 s through a LL layer of ~0.14 mm thickness, which is fast enough compared to the LL transit time of well over 10 s. 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. Therefore, the diffusion and possible convection could maintain sufficient LiT transport to the collection surfaces to maintain LiT crystallization. To facilitate crystallization on its collection surfaces, the SCT plate surface could be roughened, for example, with sand blasting. Once the initial crystallization starts, further crystallization would be facilitated. It should be noted that it is only necessary for the SCT to capture a small fraction of the passing LiT. With a 1 l s^{-1} flow with 0.5% T or 2.5 g s^{-1} 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 the relatively modest amount of LL of ~20 l, one can for example envision running several SCT units in parallel and switching off some of the units for tritium regeneration as described below. If 5 SCT units were used in parallel, the total LL volume in the SCT units would be less than 100 l.

5.2. Tritium regeneration from SCT

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 the SCT plates. The LL draining from the SCT should be rapid, since the LL has to only flow down the vertical plate 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 s, it is still only about 300 s. The SCT can be then heated to 400–600 °C to release tritium and deuterium as shown in figure 6. Since the LiT (or LiD) layer is very thin, the release of T (or D) should occur at a relatively modest elevated temperature of 400–600 °C, as shown in the lab tests [11, 20]. The present simple SCT design with no moving parts, therefore, should be well suited for repeated regeneration cycles, while minimizing the tritium inventory. With the LL drained, the energy required to heat and regenerate T and other gases should be modest.

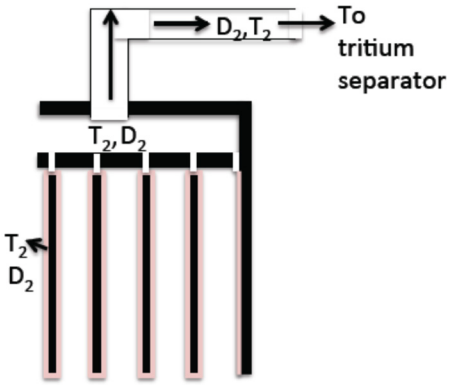

Figure 6. A schematic of the SCT regeneration.

5.3. A 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^{-1}) compared to LL (~0.5 g cc^{-1}). The enriched Li–T can be then channelled out of the separator as shown in figure 7. An advantage of the centrifugal method is that the Li–T can be extracted continuously. The unit does have moving parts, however, and needs to be located well away from the magnetic field. Commercial centrifuge units handle fluid flow rates of 20 l s^{-1}, 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, as opposed to the SCT case, one would 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

6. Tritium (T) inventory of the LL system and lithium clean-up

6.1. Tritium inventory estimate

With a LL-loop system, it is important to consider its T inventory issues. It is clearly desirable to minimize the T site inventory. A simple schematic of the LL volume for the present LL-loop system is shown in figure 8. It is assumed that the LL volume inside the fusion reactor chamber is less than 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 a divertor area of ~100 m^2 can be covered with only about 20 l of LL. For the divertor strike point area needed for heat flux handling, the LL thickness can be as much as ~1 mm. For a strike point area of ~10 m^2, a 1 mm thick film needs only about 10 l of LL per SCT. Here, 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, bringing 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, and if necessary, it can likely to be as low as ~50 l. We then assume

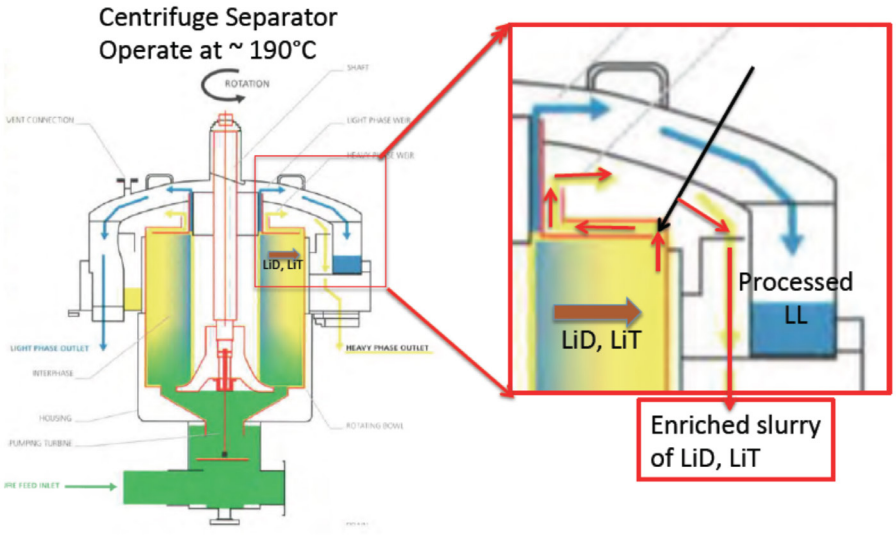

Figure 7. A schematic of the centrifuge Li–T separator surfaces.

that the LL-loop system volume up to the entrance of the cold trap is 500 l, including the dust filters and connecting pipes. Again, the volume can be smaller by careful design. The SCT has only 20 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 s^{-1}, recovery of 0.5 g s^{-1} for tritium in real time for steady state 1 GW-electric fusion power plant operation requires the T concentration to be reduced by 0.1% by the SCT. For a 1 l s^{-1} LL-loop system, the T inventory is a function of the reactor chamber T concentration in %. Naturally, the T inventory increases with the in-chamber T concentration. If we assume the 0.5% concentration which is in the high range of the expected concentration, a T inventory of 2.5 kg 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 section 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. One could also reduce the LL volume by a factor of two by careful LL-loop design, so that the inventory would be reduce to 1.25 kg. One could also run the LL system at a lower T concentration of 0.3%. Together with the LL volume reduction, the total T inventory could then be reduced to less than 1 kg. It is therefore possible to reduce the tritum inventory level as required by plant site requirements. In general, it is advantageous to run the LL system at lower temperatures closer to 200 °C to reduce the T saturation concentration and minimize tritium inventory.

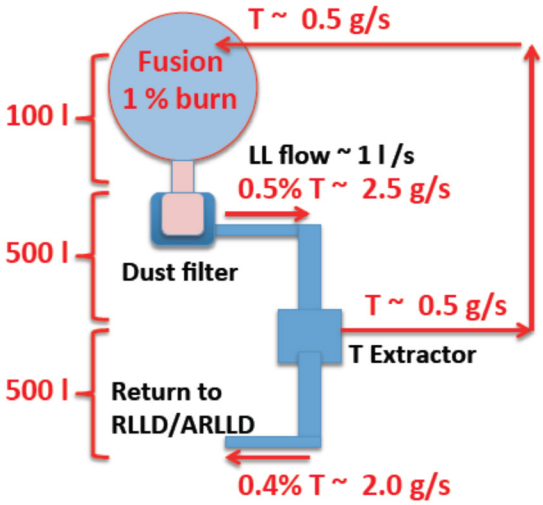

Figure 8. A schematic of the T-inventory.

6.2. *Lithium clean-up*

Once operation of the fusion power plant is complete and the facility is ready for maintenance, it is necessary to clean the LL and T out of vacuum vessel (VV). After the tritium injection is stopped, the LL-loop should be operated to further reduce the T concentration down to a level well below the 0.1% level. Once the T concentration is sufficiently low, the LL is drained into a reservoir kept at 200 °C as shown in figure 9. The VV and the divertor chamber temperatures should be maintained at 600 °C to further reduce the LL and T levels. At 600 °C, the LL vapor pressure is quite large, so that any residual LL and the LL vapor should be condensed into the colder LL reservoir at 200 °C. Similarly, the released residual gases, including tritium, are pumped out through a vacuum pumping system used for the gas/tritium recovery system attached to the LL-loop. After

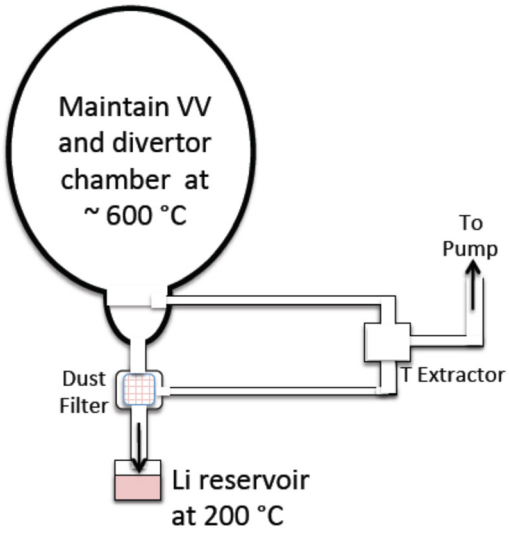

Figure 9. A schematic of the LL-loop shutdown scheme after power plant operation ends.

making sure that the LL vapor pressure and other gas releases are reduced to minimal levels, the VV system can be cooled down more aggressively before introducing air. The air exposure converts any remaining lithium compounds into lithium carbonate, which is the most stable and safe lithium compound to handle.

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 s^{-1}. In this paper, we examined the viability of such a LL-loop in a fusion power plant, including dust, impurity, T recovery, T inventory and Li/T clean up after power plant operations. With a modest LL-loop, we offer a solution to the long standing dust accumulation problem in a steady-state fusion reactor. A major technical issue for such a LL-loop is the T recovery from LL, 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 for extracting oxygen, since the cold trap does not work for very low T concentrations, i.e. less than the saturation concentration of 0.08% of LL at 200 °C as in 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 the SCT (surface cold trap), to extract tritium from the surfaces of the collection plates, rather than volumetrically for the case of the conventional CT. The SCT has the advantage of reducing the LL volume and facilitating T recovery. The T can be recovered by draining the LL from the SCT, and then heating the SCT to ~600 °C to release tritium and other gases from the Li–T accumulated on the SCT plate surfaces. Because of its simple and robust mechanical design, the SCT can allow multiple thermal cycles as needed for tritium recovery. We also proposed another innovative tritium removal concept, using centrifuge technology, as LiT is twice as heavy 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. The LiT collection and T recovery is an important research area for further innovation and optimization in the future. We may also note that dust and solid Li compound removal from the reactor chamber using LL could be an important channel for T recovery, as the dust and particles of solid Li compounds could contain significant amounts of T. If the amount of generated solid particles turns out to be large, the T released from dust/solid compound regeneration may be sufficient to recover most of the T needed for the T fuel cycle, reducing the reliance on the LiT collection and T recovery system. 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 ~0.5% for T concentration, yielding a total T inventory of 2.5 kg. The T-inventory could be reduced significantly below 1 kg by reducing the T concentration and LL volume. Operating the LL-loop system at lower temperature is generally favourable for tritium inventory by reducing T saturation. In terms of LL safety, it is also 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. We may consider maintaining a ~200 °C temperature for much of the divertor surfaces to facilitate particle pumping, while allowing the temperature to rise near the divertor strike point to protect the high-Z PFC surfaces by promoting evaporation and stimulating divertor radiation. Another important consideration for the LL-loop is a double-walled configuration with an evacuated outer layer, which should greatly increase LL safety. The vacuum layer provides good thermal insulation to keep the LL hot. It also helps for the detection of any 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. Finally, we provided a methodology to prepare the machine for periodic maintenance in a way that should essentially eliminate all of the LL and T from the reactor chamber.

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 RLLD concepts [20, 21]. 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 [22] 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 particle [9, 10] 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 facilitating such as various laboratories at the Princeton Plasma Physics Laboratory and the Unicersity of Illinois at Urbana-Champaign, where various aspects of the LL system including the dust filter, SCT, tritium recovery (using hydrogen), etc, are being tested separately and on a smaller scale.

Acknowledgments

This work was supported by DoE Contract No. DE-AC02-09CH11466.

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