Fusion Engineering and Design 89 (2014) 1989–1994

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

# Normal operation and maintenance safety lessons from the ITER US PbLi test blanket module program for a US FNSF and DEMO

B.J. Merrill[a,*], C.P.C. Wong[b], L.C. Cadwallader[a], M. Abdou[c], N.B. Morley[c]

[a] Fusion Safety Program, Idaho National Laboratory, Idaho Falls, ID, USA
[b] General Atomics, San Diego, CA 92186-5608, USA
[c] University of California Los Angeles, Los Angeles, CA 90095-1597, USA

ARTICLE INFO

*Article history:*
Received 2 September 2013
Received in revised form 6 April 2014
Accepted 23 April 2014
Available online 12 June 2014

*Keywords:*
Fusion
Safety
DCLL
DEMO
PbLi
Blanket

ABSTRACT

A leading power reactor breeding blanket candidate for a fusion demonstration power plant (DEMO) being pursued by the US Fusion Community is the Dual Coolant Lead Lithium (DCLL) concept. The safety hazards associated with the DCLL concept as a reactor blanket have been examined in several US design studies. These studies identify the largest radiological hazards as those associated with the dust generation by plasma erosion of plasma blanket module first walls, oxidation of blanket structures at high temperature in air or steam, inventories of tritium bred in or permeating through the ferritic steel structures of the blanket module and blanket support systems, and the $^{210}$Po and $^{203}$Hg produced in the PbLi breeder/coolant. What these studies lack is the scrutiny associated with a licensing review of the DCLL concept. An insight into this process was gained during the US participation in the ITER Test Blanket Module (TBM) Program. In this paper we discuss the lessons learned during this activity and make safety proposals for the design of a Fusion Nuclear Science Facility (FNSF) or a DEMO that employs a lead lithium breeding blanket.

© 2014 Elsevier B.V. All rights reserved.

## 1. Introduction

A leading power reactor breeding blanket design candidate for a fusion demonstration power plant (DEMO) being pursued by the US Fusion Community is the Dual Coolant Lead Lithium (DCLL) concept. The safety hazards associated with the DCLL blanket concept during bounding accident scenarios have been examined in several ARIES reactor design studies [1,2]. These studies identify the largest radiological hazards as those associated with the dust generation by plasma erosion of the blanket module first wall, oxidation of module structures at high temperature in air or steam, inventories of tritium bred in or permeating through the ferritic steel structures of the blanket module and blanket support systems, and the $^{210}$Po and $^{203}$Hg produced in the lead lithium (PbLi) breeder/coolant.

While these reactor design studies produced a reasonable preliminary safety assessment of this blanket concept, what they lacked is the scrutiny associated with a licensing review process. An insight into this process was gained during the US participation in the ITER Test Blanket Module (TBM) Program [3], where, in addition to identifying the accident hazards to public and environment associated with the US DCLL TBM, issues of occupational radiation exposure, Test Blanket System (TBS) component classification and commensurate design requirements, operational and maintenance radioactive releases within the facility, and operation and decommissioning waste stream classifications had to be addressed. The results of this detailed safety analysis of a single prototypical test blanket module can be found in the US DCLL TBM Preliminary Safety Report (PrSR) [4].

In this paper we discuss the primary operational and maintenance safety hazards identified for PbLi during this process and, based on the lessons learned from this experience, make proposals for safety features in the design of a reactor concept such as a FNSF or DEMO that employs a PbLi breeding blanket.

The following sections of this article give a brief overview of the DCLL blanket concept, describe the hazards associated with this blanket concept, review the operation and occupational exposure issues of the DCLL, and propose potential design features that address these safety concerns.

## 2. Physical layout of DCLL TBM systems

Fig. 1 contains a schematic of the US DCLL TBM blanket concept [4]. This blanket concept has two separate cooling systems. Fast flowing helium in toroidal channels behind the first wall

* Corresponding author. Tel.: +1 208 526 0395.
*E-mail address:* Brad.Merrill@inl.gov (B.J. Merrill).

http://dx.doi.org/10.1016/j.fusengdes.2014.04.076
0920-3796/© 2014 Elsevier B.V. All rights reserved.

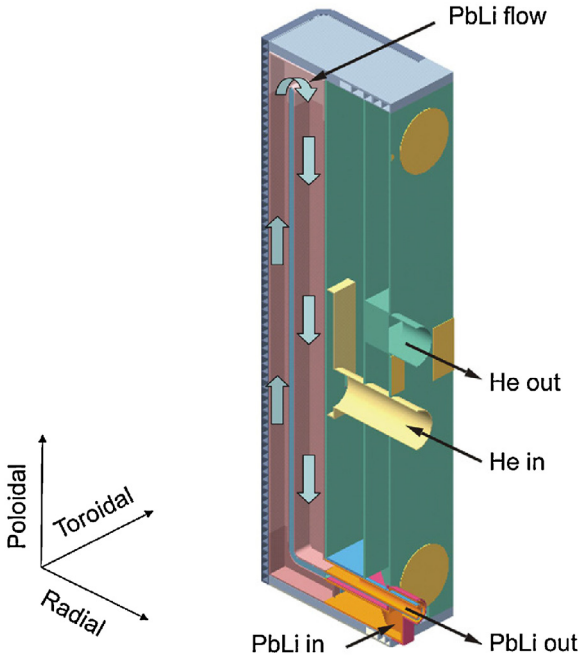


**Fig. 1.** DCLL design schematics for ITER TBM.

(FW) and poloidal and radial channels inside the internal flow dividers provide cooling for these low activation F82H ferritic steel structures of this concept. Two PbLi breeding zones are radially situated behind the FW. PbLi is both the tritium breeding material and coolant for these breeding zones. SiC-composite inserts in these breeding zones provide thermal insulation for the internal blanket walls from the poloidally flowing PbLi. In addition, these inserts serve as an electrical insulator to reduce the magneto-hydrodynamic (MHD) forces within the flowing PbLi.

For the DCLL TBM Test Blanket System (TBS), there are two separate units that supply coolant to the TBM while it is inserted in an ITER test port. The helium system is located in ITER's Tokamak Cooling Water System (TCWS) vault. The PbLi system is located in an ITER equatorial port cell behind the ITER biological shield and in line with this test port. Fig. 2 contains a schematic of the PbLi system.

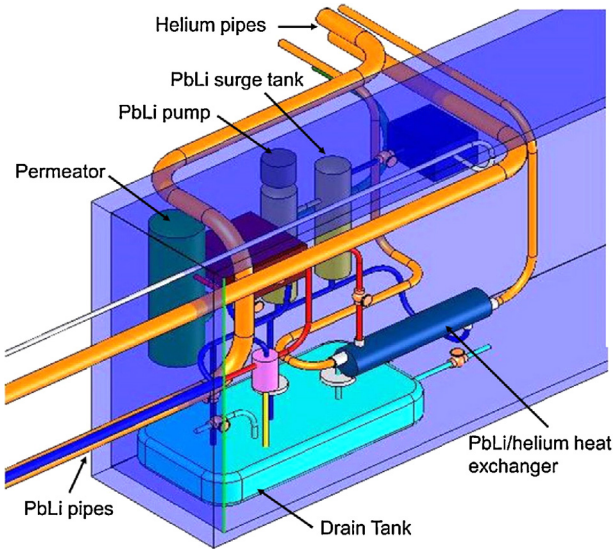


**Fig. 2.** View of the DCLL TBM PbLi system.

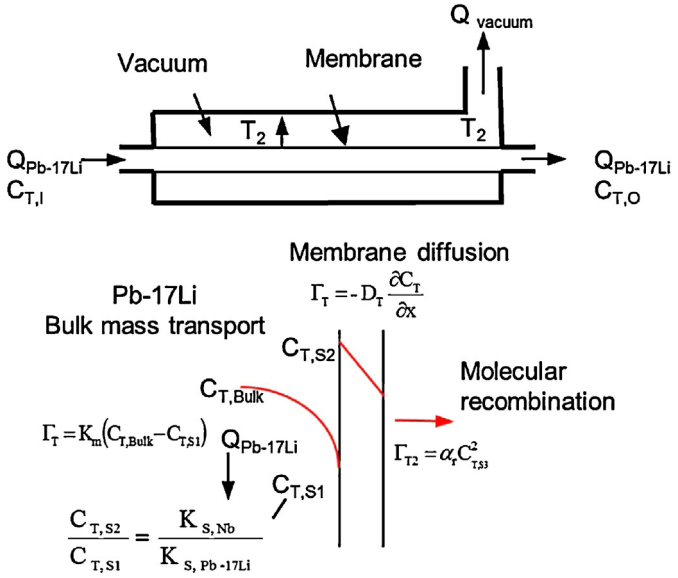


**Fig. 3.** Schematic of a vacuum permeator tritium extraction system, where $C_{T,i}$ are tritium concentrations ($m^{-3}$), $\Gamma_{T,i}$ are tritium fluxes ($m^{-2}\ s^{-1}$), and $Q$ is volumetric flow rate ($m^3/s$).

The DCLL TBM is intended as a prototypical blanket module. In contrast, the TBM TBS is not intended to be a scaled prototype of the systems required for a DEMO reactor. But this TBS does contain all of the components required for DEMO reactor operation, such as components for tritium extraction, fluid flow, pressure regulation, heat extraction, coolant supply, purification, and storage during maintenance operations.

## 3. Operational radiological hazards

In this section we discuss some of the operational safety radiological hazards of PbLi, in particular inventories of tritium bred in or permeating through the ferritic steel structures of the DEMO blanket module and blanket support systems, and the $^{210}$Po and $^{203}$Hg produced in the PbLi breeder/coolant.

### 3.1. Tritium operational concerns

A DEMO or power reactor must breed ~$5.6 \times 10^4$ g of tritium, per 1000 MW fusion power, per year. It has been well documented that tritium permeation for presently proposed blanket concepts is an operational safety concerns for the public and plant staff. Besides Refs. [1,2], see also Refs. [4–6]. To ensure that the annual operational tritium release to the environment is less than 1 g/yr, these concepts rely on a high efficiency coolant tritium extraction system (TES), permeation barriers, and air detritiation systems to achieve the required overall facility tritium retention factor of 1 part in $5.6 \times 10^4$ per 1000 MW fusion power per year. This is a challenge because of the mobility of tritium through most structural materials, and the low solubility of tritium in PbLi.

For the DCLL blanket concept, the TES must be >90% efficient. A TES concept that is predicted to obtain this efficiency is called a "vacuum permeator" [2]. A rough schematic of this concept is shown in Fig. 3. Its basic operation creates turbulent PbLi flow (velocity at ~5 m/s) through 1 cm diameter, 5 m long tubes (permeation membrane) that reside within a vacuum environment. Radial transport of the tritium in the flowing PbLi is by turbulence enhanced diffusion, and is the primary factor estimated to govern extraction efficiency. Once the tritium arrives at the PbLi side of the permeation membrane surface, the tritium is assumed to come

out of solution and enter the permeation membrane under equilibrium conditions, which is surface transport as defined by Sievert's law. After diffusing through the membrane material, the tritium is released into the vacuum by molecular recombination and swept away by the vacuum pump to the reactor's tritium fueling and storage system. In theory, permeator units that have a combined permeator tube capacity of 2060 tubes (325 $m^2$ of membrane surface area) could process the entire PbLi flow stream of a DCLL reactor blanket at outlet temperatures and keep tritium pressure above the PbLi to <1 Pa. At this condition, the annual operational release to the environment meets the required 1 g/yr limit.

However, this "vacuum permeator" has not been tested even on a "bench top" scale. To date, the highest PbLi TES component efficiency which has been demonstrated is that of a compact mass extractor. This device was demonstrated to achieve extraction efficiency between 20 and 30% for tritium pressures ranging from 220 to 1350 Pa as tested in the Melodie facility [7]. This facility tested an 800 mm high packed bubble column consisting of four vertically stacked Sulzer Company (Mellapak® 750Y series) cylindrical packed beds, each cylinder bed having a diameter of 60 mm. The operation of these units is to introduce the liquid (PbLi) at the top of the column, allowing the liquid to drain to the bottom, and to introduce a carrier gas (argon) at the bottom of the column, allowing the gas to flow in opposite direction to the top of the column. By flooding the unit (immersing the beds in liquid), the column acts as a bubble column. In this configuration, the efficiency was only ~10%. By operating the column in a non-immersed state, the efficiencies increased to the higher reported values. Presumably in this operation state, the liquid adheres to the bed's metal surfaces forming a thin flowing layer. Since the major resistance of hydrogen release from PbLi is diffusion in the liquid, then the thin films allowed a higher mass transfer to occur to the sweep gas.

The requirement for high efficiency tritium extraction for PbLi is not only a requirement for a DCLL DEMO blanket concept but also for a Helium Cooled Lead Lithium (HCLL) DEMO blanket concept. Gastaldi [5] is recommending a TES efficiency of 80% to achieve the annual tritium release limit of 1 g/yr. To obtain 80% extraction efficiency, it would take five columns, as tested in Melodie (PbLi flow of $1.4 \times 10^{-5}$ $m^3/s$), arranged vertically in series (height would be 4 m). For an inventory (585 $m^3$) recirculation rate of once per day, the volumetric flow is $6.8 \times 10^{-3}$ $m^3/s$, requiring a packed bed surface area and volume of 3385 $m^2$ and ~5.5 $m^3$, respectively. However, Gastaldi is proposing a required rate of ~14 times per day resulting in a tritium pressure for the PbLi of ~120 Pa (within a factor of 2 lower than Melodie conditions). For a DCLL, the rate is ~95 times per day, with the added requirement of tritium pressures 200 times lower than examined in the Melodie facility.

The tritium production rate in the US DCLL TBM is estimated at $4.2 \times 10^{-7}$ g/s during an ITER pulse [4]. Results in Ref. [4] from modeling the entire Test Blanket System (TBS which includes the TBM, and PbLi and Helium cooling systems) show that after a year of consecutive ITER pulses that of the ~0.7 g of tritium produced by the DCLL TBM, 69.6% permeates through the permeator and is stored on the tritium getter bed, 12.5% permeates into the helium piping where the majority of it could be removed by the helium cleanup system before it permeates into the reactor building, 7.8% permeates through the PbLi piping into the port cell, and 6.1% permeates through the TBM walls into the ITER VV. The remaining 4% resides in structures and coolants of the TBM systems. This prediction was very conservative because: (1) the vacuum permeator used ferritic steel membranes with a calculated extraction efficiency of only 70% (no effort to prove this efficiency by adding additional membrane surface area was undertaken for this analysis), (2) tritium extraction from the helium cooling systems was not included, (3) no permeation counter measures for the TBS were employed, such as the $Er_2O_3$ permeation barriers being developed by Refs. [8], (4) tritium permeation through the FW from plasma implantation was not considered because the FW was assumed to be clad with beryllium, and (5) ITER will execute 3285 consecutive full-power pulses over three consecutive months of continuous operation (25% availability for the year). Under these assumptions, 0.14 g/a were predicted to be released into ITER building's port cell and Tokamak Cooling Water System vault during this three month time period. This release should not be a problem for the ITER air detritiation systems.

A problem associated with any operational release in to a reactor building is occupational radiation exposure, often described in terms of a derived air concentration (DAC). One DAC is the allowed air concentration for entry into a given room by plant personnel. The air concentration of tritium that produces a dose rate of 1 DAC in the ITER port cell is $3.4 \times 10^5$ Bq/$m^3$ [9]. For the port cell, the tritium air concentration for the estimated release from the DCLL PbLi system during normal operating conditions would be well above 1 DAC. If the PbLi system could be redesigned to fit within a stainless steel enclosure, whose helium atmosphere was continually swept through U-beds to capture any permeating tritium, the DAC could be reduced.

While this analysis may be somewhat conservative for an ITER TBM TBS, it may not be for an FNSF or a DEMO, where steady state operation is a requirement. But this same tritium confinement strategy could also work for an FNFS or DEMO reactor with a DCLL PbLi system if needed.

### 3.2. $^{210}$Po and $^{203}$Hg operational concerns

Based on a preliminary conservative assessment, two additional radiological concerns for DEMO PbLi blanket are the production of the radionuclides $^{210}$Po and $^{203}$Hg. The reasons for concern over these isotopes are: (1) according to dose coefficients developed by [9], $^{210}$Po is $10^5$ times more hazardous than HTO, and $^{203}$Hg is $10^2$ times more hazardous than HTO, (2) these isotopes are mobile in PbLi [10], and (3) $^{210}$Po usually constitutes the largest dose contributor during PbLi spills and maintenance operations [1,2,4].

Both of these isotopes have well defined primary production pathways in PbLi beginning with the Pb isotopes $^{208}$Pb and $^{206}$Pb, respectively as follows:

$$^{208}\mathrm{Pb} \xrightarrow{n,\gamma} {}^{209}\mathrm{Pb} \xrightarrow{\beta-} {}^{209}\mathrm{Bi} \xrightarrow{n,\gamma} {}^{210}\mathrm{Bi} \xrightarrow{\beta-} {}^{210}\mathrm{Po} \quad (1)$$

and,

$$^{206}\mathrm{Pb} \xrightarrow{n,\alpha} {}^{203}\mathrm{Hg} \quad (2)$$

Given the radio-toxicity of $^{210}$Po, the proposed approach for controlling the generated inventory of $^{210}$Po in the PbLi of a fusion power reactor is to have an online bismuth removal system that limits its concentration to 1 ppm. The result is a PbLi equilibrium concentration of $^{210}$Po of below 0.1 ppb [11]. As far as the authors are aware, such a removal system has not been developed or experimentally verified. For a fusion reactor, a 0.1 ppb concentration of $^{210}$Po represents an equilibrium inventory for the reactor of 95.3 TBq (~0.16 TBq/$m^3$). Confining this inventory during operation is not an issue for a cooling system that employs a vacuum permeator TES. However for a bubble column, $^{210}$Po will be released into the extraction system gas stream. Using the release rate equation determined by Schipakin [10] and taking into account radioactive decay ($^{210}$Po half-life is 138.4 d), after ~3 yr of bubble column operation at 400 °C (Melodie test temperature) and a recirculation rate of 14 times per day, an equilibrium inventory of ~76.2 TBq will be removed by the bubble column and will reside somewhere in the TES. This TES inventory is comparable to that in the entire blanket PbLi volume. At the predicted removal rate, the $^{210}$Po concentration in the PbLi would drop to 0.02 ppb, given the $^{210}$Po production rate

calculated from the decay of $^{210}$Po at 0.1 ppb. Even at a recirculation rate of once per day, the TES inventory would be ~5.6 TBq. While this does not seem like much $^{210}$Po, remember that it is equivalent in radiological toxicity to the accidental release of ~1.5 kg of HTO.

According to Petti [1], after 40 full power years (FPY) of power plant operation, the inventory of $^{203}$Hg will reach $4.3 \times 10^4$ TBq if no removal of the Hg is attempted, which is still less hazardous than $^{210}$Po at 0.1 ppb (95.3 TBq) since $^{210}$Po is $10^3$ times more radio-toxic than $^{203}$Hg. However, Hg is ~$10^4$ times more volatile than PbPo (the volatility of PbPo is a rate limiting mechanism in Po release from PbLi [10]). Using a production rate for $^{203}$Hg based on Fig. 5 of Petti [1], and the same bubble column modeling assumptions used for $^{203}$Hg, except with a half-life of 46.57 d and an evaporation flux based on Hg vapor pressure instead of that for PbPo, then at the end of 40 FPY most of the $4.3 \times 10^4$ TBq of that inventory resides in the TES and the PbLi $^{203}$Hg concentration remains at ~0.08 ppb. The applied model suggests that the bubble column will be ~4.5% efficient at removing Hg from the PbLi.

The end-of-life TBM production of $^{210}$Po and $^{203}$Hg are 66.6 and 1332 GBq, respectively [4]. If this entire inventory of $^{210}$Po were to be accidentally released to the environment as a stacked release during average weather conditions (P-G stability conditions D with a wind speed of 4 m/s), the dose at the site boundary would be 0.08 mSv [12]. A similar calculation for $^{203}$Hg results in a dose of 0.002 mSv. During operation, the only mode for release of PbLi would be though small leaks in pumps, valves, etc. However, a helium-swept enclosure that could house the PbLi system, if feasible, as proposed earlier to prevent tritium releases to the reactor building, will also contain the releases from these leaks. This enclosure will also contain PbLi spills that occur during accidents, and allow ease in replacing the TBS (replaced by a second enclosure and TBS) without spreading the contamination through the building.

## 4. Maintenance radiological hazards

Because of the radioactivity associated with PbLi, it has been assumed that the most likely DCLL DEMO maintenance mode would be robotic. For example, in the US, a transition point from hands-on, contact handling to robotic handling is 2 mSv/h [13]. Individual facilities may adopt lower dose values above which remote handling should begin, but clearly the component activation of a DCLL PbLi system will warrant use of remote maintenance. While remote maintenance is most likely the only possibility, there may be instances where personnel access to assist robotic maintenance may be required or to perform routine equipment inspections that cannot be performed by robotic equipment. The only mode of maintenance for the US ITER DCLL TBS is hands-on maintenance due to the experimental nature of the DCLL TBS. The approach to accomplish this mode of maintenance for the PbLi system is to drain the PbLi into a shielded holding tank. As was the case for operational radiological hazards, tritium, $^{210}$Po and $^{203}$Hg are also maintenance hazards. For instance, Schipakin's data [10] clearly demonstrates that even at room temperature PbLi will release $^{210}$Po. Some of the maintenance concerns for the plant staff regarding PbLi systems are discussed in this section.

### 4.1. Ionizing radiation maintenance concerns

There will be two sources of radioactive material in the PbLi systems that will produce gamma radiation: activated PbLi and activated F82H corrosion products from the internal surfaces of the blanket and PbLi system. During maintenance, the PbLi will be emptied to a drain tank. However, based on TRITEX facility experience [14], it was discovered that after draining the TRITEX loop, PbLi films were found on the pipe walls that were on average ~45 mg/cm$^2$. Similar films are expected to adhere to the inner surfaces of the components in the DCLL PbLi system. According to activation calculations [15], these PbLi films will be radioactive, with the activity after one week dominated by $^{203}$Pb (2.7 MeV gamma emitter).

In addition to the activated PbLi films, the hot surfaces within the TBM and the PbLi outlet pipe wall will undergo corrosion at the PbLi/Ferritic Steel (FS) interface. The corrosion products from blanket FS surfaces will be activated. These FS corrosion products will mix into the PbLi bulk flow and be deposited, along with the non-radioactive FS corrosion products from the PbLi outlet piping, on to the inside surfaces of the lower temperature components of the TBS, such as pipe walls, heat exchanger tubes, valves, permeator tubes, etc. The anticipated corrosion rate of these hot surfaces is 20 μm/yr at 450 °C based on experimental data. Over the lifetime of the TBS (0.528 FPY) ~11 μm of erosion from the hot surfaces will result in a re-deposited layer thickness in the TBS of ~1.8 μm. Fe-55 and Mn-54 (0.8 MeV gamma emitter) will dominate the radioactivity of this film one week after reactor shutdown. The dose rate from positions around the PbLi system from these gamma emitters was calculated with the QADMOD computer code for use in an Occupational Radiation Exposure (ORE) analysis [16]. The predicted peak hands-on-maintenance dose was near the PbLi/helium heat exchanger at ~0.1 mSv/h, which is at the ITER target for port cell maintenance with the port cell bioshield removed.

An ORE analysis is based on the product of dose rate incurred by a technician performing a maintenance or inspection activity on a TBS component multiplied by the product of time required to complete the maintenance activity and the frequency per year that the activity was be performed for the entire DCLL TBS [16]. The repair time and frequency of component replacement was based on component failure rate data for the three cooling systems. The annual dose for maintenance, testing, and calibration of the PbLi TBS is estimated to be 6.3 p-mSv. The dose for same activities for the two helium loops is 1.9 p-mSv. The major maintenance activity is TBM replacement scheduled to occur every three years which is estimated to be 22.9 p-mSv, or 7.6 p-mSv/a. Using the dose values from this very preliminary and conservative assessment, the total annual dose commitment for the DCLL TBS is 15.8 p-mSv, which is much higher than the ITER International Organization's goal for all six TBMs of 5 p-mSv/a. Additional consideration should be given to more rapid repair, film removal, or remote procedures in future ORE analysis of the DCLL TBS.

While this ORE analysis results does not directly translate to the possibility of hands-on-maintenance for a DEMO, draining the PbLi heat transport system into a shielded holding tank and removing the adhering PbLi film, would significantly reduce the radiation field in which robotic equipment must operate.

### 4.2. Ingestion radiation maintenance concerns

The possibility of inhalation doses exist from tritium and volatile activation products (in particular $^{210}$Po and $^{203}$Hg) being released during maintenance activities that open the PbLi TBS to a room atmosphere. As mentioned above, administrative controls limit these occupational doses by establishing radiation and ventilation zoning requirements that are based on a derived air concentration (DAC). For $^{210}$Po and $^{203}$Hg the air concentrations that constitute 1 DAC for the ITER Port Cell are 4.1 Bq/m$^3$ and $9.1 \times 10^3$ Bq/m$^3$, respectively. The inventories of $^{210}$Po and $^{203}$Hg that can be readily mobilized during maintenance activities that open the PbLi system up to air are those in the 50 μm PbLi film adhering to the inside surfaces of the PbLi system's components and the free surface within PbLi drain tank. A 50 μm film adhering to the ~25 m$^2$ internal surface area of the PbLi system represents a PbLi volume

that is ~2.5% of the total PbLi inventory. This translates to a $^{210}$Po and $^{203}$Hg inventory of 1680 and 33,520 MBq, respectively. Because these film inventories produce $^{210}$Po and $^{203}$Hg release rates that exceed 1 DAC in the ITER Port Cell even as solidified PbLi films, one proposed shutdown procedure for a DEMO system, that should be investigated further, is to include an active sweep gas within the systems to reduce the level of activity of $^{210}$Po, $^{203}$Hg, and tritium in the PbLi film. The $^{210}$Po and $^{203}$Hg can be removed from the gas with a cold trap, and the tritium removed by U beds. As a consequence, by the time that port cell entry is allowed ten days after shutdown, the $^{210}$Po and $^{203}$Hg film inventories will drop to 12 MBq and 230 MBq, respectively, based on the $^{210}$Po surface release rate information experimentally determined by Schipakin [10].

At these lower inventory levels, $^{210}$Po diffusivity of $3 \times 10^{-16}$ m$^2$/s, a release velocity of $2.9 \times 10^{-9}$ m/h at 20 °C [10], and an air exchange rate of 24 volume exchanges per day, the port cell concentration would reach 3.4 Bq/m$^3$ and 68.4 Bq/m$^3$ for $^{210}$Po and $^{203}$Hg, respectively. The resulting inhalation dose rate would be $8.3 \times 10^{-1}$ and $7.5 \times 10^{-3}$ DAC based on ICRP worker dose conversion factors [9] and a breathing rate of 1.1 m$^3$/h. As can be seen, the DCLL PbLi system with this proposed procedure now complies with the radiation and ventilation zoning for this room. This approach could also be used in FNSF and DEMO if needed.

## 5. Component classification

Each component of the DCLL TBS had to be listed and categorized in accordance with ITER Environmental, Safety and Health requirements and in accordance with French Regulatory licensing requirements [4]. The DCLL TBS is composed of four sub-systems, which are:

1. The DCLL TBM, that makes use of 2 coolants: PbLi, which also serves as the tritium breeder and a neutron multiplier for this concept, and helium.
2. The helium cooling system, composed of two cooling loops: a primary helium heat transport loop that cools the reduced activation ferritic–martensitic steel structures of the TBM, and a secondary helium loop that cools the liquid breeder by way of a PbLi/He heat exchanger.
3. The PbLi system, which provides PbLi circulation, heat removal and tritium extraction.
4. The tritium extraction system, which uses the concept of vacuum permeator for the removal of tritium from the PbLi. Tritium extracted by this system is to be stored in a gettering unit until transferred to the tritium building for analysis.

One category of component classification, besides safety, seismic, remote handling, vacuum and tritium component classifications, is the component's classification pressure classification under the French Regulatory Pressure Equipment Directive (PED) and Equipment Under Pressure Nuclear (ESPN) Order. To fall within the PED, the fluid pressure confined within the component has to exceed 1.5 Bar. In addition to 1.5 Bar pressure, to fall under the ESPN Order the radioactivity contained within the fluid has to exceed 370 MBq, with an exception for several gases (including tritium) of 370 GBq (~1 mg of tritium).

Based on these criteria, at the time the assessment was made the TBM and PbLi cooling loop components were classified as ESPN components. Components that fall under this classification are similar to ASME "N" stamped components. This implies that similar systems for a US FNSF and DEMO could also fall under this classification. The quality assurance requirements from design through manufacturing make these components more expensive to produce, and the pre-service and in-service testing and inspection requirements for these components makes the operation and maintenance of these components labor intensive.

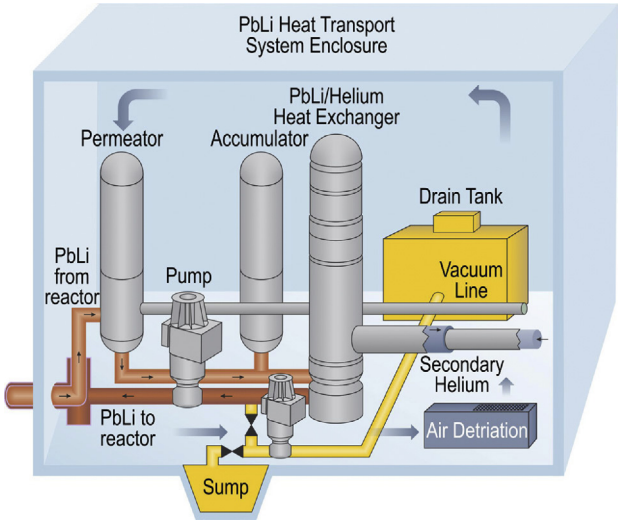


**Fig. 4.** Schematic of PbLi HTS for a DCLL blanket inside of a proposed safety enclosure.

## 6. Summary of lessons learned

Based on this DCLL TBM experience, design options are available to make this concept safer for the public, environment and plant employees.

The primary radioactive confinement boundaries of a DEMO, the vacuum vessel and primary coolant heat transport system (HTS), will be comprised largely of ASME N stamped components. These components will require periodic in-service inspection and testing. However, in the US, like LWRs, a regulatory framework for licensing a fusion DEMO should be developed. This new regulatory framework should allow an exemption for in-vacuum-vessel components.

Unlike some blanket concepts, one HTS, the PbLi HTS for a DCLL blanket, will probably require remote maintenance and inspection equipment that can withstand an intense ionizing radiation environment. However, the intensity of this radiation can be greatly reduced by removing the PbLi from the HTS and storing it in a shielded holding tank. In addition, enclosures, as illustrated in Fig. 4, can be used to control tritium permeating from this HTS, and any PbLi spills from this system.

Because of the radio-toxicity of $^{210}$Po and $^{203}$Hg, the control of these isotopes is a safety concern for DEMO during maintenance and accident remediation activities. Even though they are relatively immobile in solidified PbLi, at room temperature present data [10] indicates that surface releases from reactor components can result in unacceptable air concentrations (>4 Bq/m$^3$) for workers.

Because the half-life of $^{210}$Po is 138.4 d (even shorter for $^{203}$Hg), the best approach to deal with this isotope may well be sweep gases to remove $^{210}$Po from any PbLi films prior to maintenance, and storage of PbLi in solid form to allow for radioactive decay before final component disposition is determined. For example, according to [1], the equilibrium concentration of $^{210}$Po in ARIES-AT PbLi is $1.5 \times 10^{11}$ Bq/m$^3$. After ~14 yr of radioactive decay, the concentration will drop to 4 Bq/m$^3$.

## Acknowledgments

This work was prepared under the auspices of the US Department of Energy, Office of Fusion Energy Science, under the DOE Idaho Field Office contract number DE-AC07-05ID14517.

## References

[1] D.A. Petti, B.J. Merrill, R.L. Moorea, G.R. Longhurst, L. El-Guebaly, E. Mogahed, et al., ARIES-AT Safety Design and Analysis, Fusion Eng. Des. 80 (2006) 111–137.
[2] B.J. Merrill, L.A. El-Guebaly, C. Martin, R.L. Moore, A.R. Raffray, D.A. Petti, et al., Safety Assessment of the ARIES Compact Stellarator Design, Fus. Sci. Technol. 54 (2008) 838–863.
[3] L.M. Giancarli, M. Abdoub, D.J. Campbella, V.A. Chuyanova, M.Y. Ahnc, M. Enoedad, et al., Overview of the TBM Program, Fusion Eng. Des. 87 (2012) 395–402.
[4] B.J. Merrill, L.C. Cadwallader, WongF C.P.C, M. Dagher, A. Ying, M. Youssef, US Dual Coolant Lead Lithium ITER Test Blanket Module Preliminary. Safety Report, Idaho National Laboratory Report, 2010, June, INL/EXT-10-18169, Rev1.
[5] O. Gastaldi, N. Ghirellia, F. Gabrielb, L. Giancarlib, J.F. Salavy, Tritium transfers and main operating parameters impact for demo lithium lead breeding blanket (HCLL), Fusion Eng. Des. 83 (2008) 1340–1347.
[6] F. Franza, Tritium Transport Analysis in HCPB DEMO Blanket with the FUS-TPC Code, Karlsruhe Institute of Technology Report, KIT-SR 7642, 2013.
[7] N. Alpy, A. Terlain, V. Lorentz, Hydrogen extraction from Pb–17Li: results with a 800 mm high packed column, Fusion Eng. Des. 49–50 (2000) 775–780.
[8] T. Chikada, S. Naitoh, A. Suzuki, T. Terai, T. Tanaka, T. Muroga, Deuterium permeation through erbium oxide coatings on RAFM steels by a dip-coating technique, J. Nucl. Mater. 442 (2013) 533–537.
[9] International Commission on Radiological Protection (ICRP), Dose Coefficients for Intakes of Radionuclides by Workers, ICRP Publication 68, Ann. ICRP 24 (4) (1994), Pergamon Press, New York.
[10] O. Schipakin, N. Borisov, S. Churkin, Experimental study of $^{210}$Po release from 17Li–83Pb eutectic, Fusion Eng. Des. 29 (1995) 164–169.
[11] S. Malang, R. Mattas, Comparison of lithium and the eutectic lead–lithium alloy, two candidate liquid metal breeder materials for self-cooled blankets, Fusion Eng. Des. 27 (1995) 399–406.
[12] M.L. Abbott, L.C. Cadwallader, D.A. Petti, Radiological Dose Calculations for Fusion Facilities, Idaho National Laboratory Document, INEEL/EXT-03-00405, 2003, April.
[13] Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant, U.S. Department of Energy, 2013, April, DOE/WIPP-02-3122, revision 7.4.
[14] H. Feuerstein, S. Horn, G. Kieser, "TRITEX–A Ferritic Steel Loop with Pb-15.8Li Facility and Operation," Forschungszentrum Karlsruhe Report, FZHA 6286, 1999, May.
[15] C.P.C. Wong, et al., Design Description Document for the U. S. Dual Coolant PbLi (DCLL) Test Blanket Module, General Atomics Report, GA-C25027, Rev. 3, 2005, November.
[16] B.J. Merrill, L.C. Cadwallader, M. Dagher, A preliminary assessment of the occupational radiation exposure for maintaining the US ITER DCLL TBM, Fusion Sci. Technol. 56 (2009) 252–256.