Overview of Kyoto Fusioneering's SCYLLA© ("Self-Cooled Yuryo Lithium-Lead Advanced") Blanket for Commercial Fusion Reactors

R. Pearson, C. Baus, S. Konishi, K. Mukai, A. D'Angiò, and S. Takeda

***Abstract*— This article outlines Kyoto Fusioneering's (KF's) initial engineering and development activities for its self-cooled lithium lead-type blanket: Self-Cooled Yuryo Lithium-Lead Advanced (SCYLLA©). We provide details on overall design, including an initial tritium breeding ratio (TBR) assessment via neutronics analysis, as well as the status of SCYLLA©-relevant R&D. This includes silicon carbide composite (SiC_f/SiC) manufacturing techniques, tritium extraction, materials compatibility, and heat transfer, which are being explored via collaboration with Kyoto University. Results of previous work in relation to this R&D are presented. Permeability coefficients indicate a promising property of SiC_f/SiC tritium hermeticity at high temperatures. Tritium extraction technology via vacuum sieve tray (VST) is shown to be demonstrated at engineering scale. A local TBR of up to 1.4 can be achieved with the SCYLLA© configuration. Fabrication methods for various SiC_f/SiC components including the blanket module, heat exchanger, and flow path components are provided. A tritium compatible high-temperature SiC_f/SiC heat exchanger is discussed. Commercial viability and reactor adaptability are considered as a theme throughout. Finally, KF's plans to build a facility for demonstration reactor relevant testing of a SCYLLA© prototype in the mid-2020s, which will provide a significant step toward commercial fusion energy, are presented.**

***Index Terms*— Composite materials, fusion power generation, fusion reactor design, high-temperature energy applications, industrial engineering, silicon carbide composite, tritium breeding blanket.**

I. Introduction

A KEY challenge to the realization of commercial fusion energy is to establish a blanket system that recovers bred tritium for a self-sufficient fuel cycle while simultaneously having practical heat recovery capability for conversion to usable energy. For commercialization, it is of particular urgency due to its low technology readiness level (TRL) as well as its strong influence on performance, cost, lifetime, waste, operational reliability, and other factors that impact the overall feasibility of a viable fusion reactor. Kyoto Fusioneering (KF) Ltd., a startup company focused on solving fusion engineering challenges, is developing a high-temperature self-cooled lithium–lead (LiPb) blanket that uses a silicon carbide composite (SiC_f/SiC) channel structure. Self-cooled blankets were first conceived in the 1980s as an optimal solution compared with other blanket configurations [1]. The design was most recently considered as a high performance, commercially attractive blanket for the U.S. ARIES tokamak design studies [2], [3]. However, no self-cooled blanket has been developed beyond conceptual design, primarily due to the limited need to develop advanced blanket technology for fusion experiments to date, but also due to the perceived maturity of suitable materials required for the design. Where other blanket designs that have been pursued, such as those for the ITER tritium breeding module (TBM) Program [4], are now further developed than self-cooled blankets, these designs typically have limited performance. Originally developed as conservative design options for the irradiation experiments on ITER, they operate at lower temperatures and thus have reduced overall efficiency, and they are more complex in design. Such factors are likely to impact negatively on commercial viability.

KF's design, Self-Cooled Yuryo Lithium-Lead Advanced (SCYLLA©), is based on advances in SiC_f/SiC technology [5] and liquid LiPb research at Kyoto University [6], [7], [8], [9] over the past two decades. Based on these developments, KF is advancing from the concept phase to the engineering phase, while retaining a particularly sharp focus on the commercial aspects of the design, which is pivotal for successful innovation [10]. KF's development is predicated on the fact that new public fusion programs, including STEP in the U.K., as well as a number of private fusion developers that have recently emerged, will see several fusion demonstrators constructed and operated over the course of the next two decades. As such, the time for the development of an advanced, commercially viable blanket is now, not post-ITER, which will start deuterium-tritium (D-T) operation in the mid-2030s. Most importantly in relation to commercial fusion is the very-high temperature (∼1000 °C) capability of SCYLLA© compared with most other blanket designs currently under development. The maximum operating temperature of SCYLLA© enables process heat applications, such as hydrogen production, as well as electricity production at high efficiency via a Brayton cycle,

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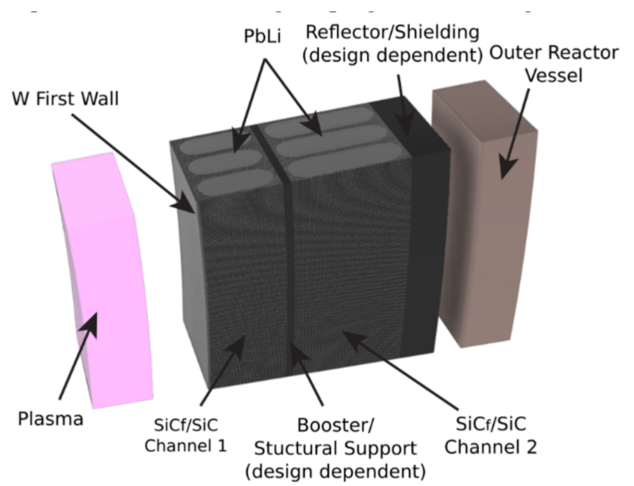

Fig. 1. 3-D concept showing the radial layout of the SCYLLA© blanket.

making it an enabling technology for fusion to be used as a tool for global deep decarbonization.

II. SCYLLA© Blanket Concept

SCYLLA© is KF's commercial concept of the established self-cooled-lithium–lead (SCLL) design. The blanket is constructed from SiC_f/SiC but also assumes the need for a plasma-facing first wall, which is made of tungsten or tungsten alloy, to protect from sputtering and plasma disruptions. The tungsten/alloy first wall is bonded to SiC_f/SiC channels, through which liquid LiPb flows. An optional neutron booster layer (detailed later) as well as an optional reflector and a neutron/photon shield are also variations on the design, which will be included or adjusted based on the required performance, specifically the required tritium breeding ratio (TBR), as the design is progressed (see Fig. 1).

The coolant LiPb also functions as the breeder material with tritium (T or ^{3}H) produced mainly via the ^{6}Li (n, a)T reaction and a much smaller contribution via the ^{7}Li(n, $n'a$)T reaction. Lead (Pb) acts as a neutron multiplier. The eutectic composition of LiPb (83 at%/99.32 wt% and 17 at%/0.68 wt%) has a low melting point of approximately 235 °C. Other LiPb compositions, which contain a greater quantity of lithium but consequently have higher melting points [11], may be favorable to increase overall TBR. Increasing the TBR can also be achieved by way of ^{6}Li enrichment. Importantly, for the SCYLLA© blanket, the lithium content can be continually adjusted to change the amount of tritium breeding during reactor operation, depending on the requirements (online TBR management). Tritium extraction can also be performed without the need of an additional medium such as helium. It is noted that helium will still be generated in LiPb, and thus, separation following tritium extraction is still necessary. Further attractive characteristics include the simple structure and modest pressure with no compressed fluids inside the vacuum vessel (VV).

The structural material SiC_f/SiC is chosen for its capability to withstand high temperatures, as well as its neutron radiation resistance at such high temperatures. In fact, SiC_f/SiC shows strength retention at 800 °C under high-dose neutron irradiation ($\simeq$75 dpa) with a potential margin for performance improvement if crystalline fibers and optimized fiber/matrix interphases are further developed [12], [13], [14], [15]. The material allows for a thin-walled structure without the need for steel, and due to its compatibility with LiPb, it does not require a protective coating [16]. The design thus benefits from a high breeder to structural material ratio. SiC_f/SiC provides electrical insulation, decreasing the magnetohydrodynamic (MHD) effect that the LiPb experiences as a flowing liquid metal in a magnetic field, though careful channel design and experimental work are needed to validate this (detailed later in this article) [17], [18]. This reduces the burden of complex design aspects associated with other separately cooled LiPb-type blankets, such as the helium coolant lithium lead (HCLL), dual-cooled lithium lead (DCLL), and water-cooled lead lithium (WCLL) designs [19], [20]. SiC_f/SiC is also approximately three times lighter than steel, which allows for easier remote handling during maintenance of the blanket structure after the heavy LiPb is drained. Its radio toxicity is lower at end-of-life than other commonly used structural materials [21] even though ^{14}C production is a concern [22], due to its half-life of 5730 years and the possibility of potentially difficult to purify airborne release when formed as a compound, e.g., CO_2.

The SCYLLA© blanket system adopts an innovative approach and will use more than one grade of SiC_f/SiC, each aimed at fulfilling functional requirements that are specific to different subsystems or regions in the blanket system. The material close to the first wall needs to be resistant to high temperature and intense neutron irradiation, while low tritium hermicity is not required (as the component is in-vessel). Behind the bioshield, however, where the neutron fluence is low, a SiC_f/SiC grade optimized for low tritium permeation and with a high heat transfer coefficient will be required to maximize the efficiency of the heat exchanger (and thus energy generation).

Both the breeder (LiPb) and structural material (predominantly SiC_f/SiC) for the SCYLLA© blanket are made from abundant chemical elements compared with those required for other blanket designs (e.g., beryllium (Be), see [23]). Relatedly, the current high cost of SiC_f/SiC could decrease if an efficient manufacturing process and economy of scale in production can be established, owing to the low cost and ample availability of the constituent materials. SiC_f/SiC is now used widely across a variety of applications in the aerospace sector, but currently, there is no production of a fusion-relevant grade of SiC_f/SiC at industrial scale, and the capability to produce more complex shapes, such as manifolds, must be developed. The manufacturing process for SiC_f/SiC is therefore a key challenge and primary cost driver.

The TBR for a radial build of infinite cylinders that are isotropic in toroidal direction with a 14-m high cylindrical neutron source generating 14.1-MeV neutrons, a 3-mm tungsten first wall, and a 70-cm blanket with a SiC_f/SiC structure of 2-cm wall thickness has been calculated using MCNP6 + FENDL3.1c (note that a 2-cm wall thickness for SiC_f/SiC is a very conservative estimate, as SiC_f/SiC with a thickness of a few millimeters is likely to be sufficient) [24], [25].

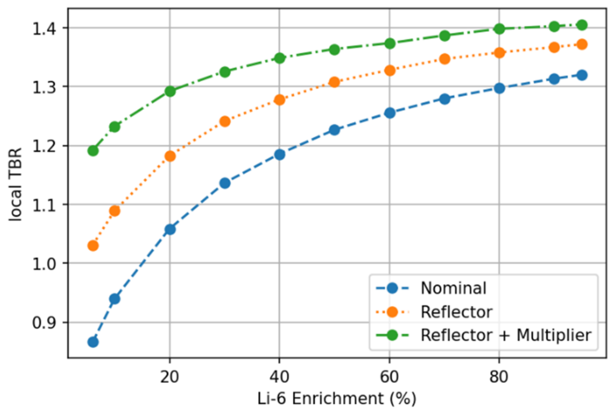

Fig. 2. Local TBR for several SCYLLA© configurations described. The lowest enrichment level corresponds to naturally abundant lithium (^{6}Li at approximately 7.4%) and the highest to 95%. Nominal refers to no reflector and no multiplier.

Two design options have been added with the intention of potentially boosting TBR and perhaps also avoiding lithium-6 enrichment. The first is a thin 2-cm Be multiplier layer at 22 cm, and the second is a 10-cm graphite reflector on the back side at 74 cm.

There are known challenges with the use of graphite in nuclear applications. However, graphite reflectors have been found to have lower activity and produce lower nuclear heat than the commonly used Be multipliers, but due to long-lived ^{14}C and ^{36}Cl, production graphite usage has to be quantified for the individual scenario [26], [27], [28]. Similarly, problems with He are well understood. However, a 2-cm Be layer at 6-m distance from the VV center of a tokamak-type reactor, as proposed as an optional addition on SCYLLA©, would result in the usage of about 9 m^3 volume of Be (approximately 16 tons), which is much less than a solid breeder-type blanket, where He is used as the sole multiplier. It is stressed that using such low quantities of Be in this design may be economically viable compared with other designs, given the low availability and production of fusion-grade Be, as well as the high cost of the material [23]. In all, the inclusion of the three options of a graphite reflector, a Be booster, and the level of lithium-6 enrichment is intended to highlight that there are several "levers" that can be pulled to potentially increase the TBR on SCYLLA© design. It is emphasized, however, that none of these options is integral to the design itself.

Modeling results for the various configurations show that the multiplier plus reflector combination offers the highest local TBR. Using natural lithium (no ^{6}Li enrichment), with a Li17Pb83 eutectic, and the booster and reflector combination, the attainable TBR is almost 1.2. This increases to above 1.4 for 95% ^{6}Li enrichment (see Fig. 2). It was found that a small reflector and multiplier layer can boost the TBR, depending on the enrichment, by 0.05–0.15 each, with their largest effect visible at lower ^{6}Li enrichment.

In a fusion reactor, the blanket cannot cover the entirety of plasma-facing regions that can be used for breeding, and space is needed for ports for diagnostics and detectors, plasma heating/current drive for magnetic confinement reactors, or injectors for magnetoinertial-type reactors, for example. The space required for the divertor(s) in a tokamak-type reactor will typically decrease the TBR value by about 17% [3]. Some high-coverage reactor configurations, such as Z-pinch or field-reversed configuration (FRC), may see smaller reductions in TBR due to fewer penetrations required for ports or systems for fueling and exhaust systems, which may not take up space in the breeding regions. As such, depending on the design, it is possible that SCYLLA© can achieve a TBR above unity without lithium enrichment. Regardless of the design of any magnetic confinement reactor, however, shielding to protect the magnets must be considered for both neutron and gamma radiation to prevent damage, degradation, and nuclear heating when superconducting magnets are used. Some of the elements in magnets could be the source of long-lasting radiation by activation.

III. R&D Status and Challenges

The main focus of SCYLLA© R&D efforts is geared toward the manufacturability of SiC/SiC and its interaction with LiPb. This section details the current status of R&D and outlines future challenges.

A. SiC_f/SiC

Several properties of SiC_f/SiC have been tested under high temperature and under neutron and ion irradiation at institutions worldwide, including Kyoto University and Oak Ridge National Laboratory. Nuclear-grade SiC_f/SiC has stable mechanical properties under neutron irradiation, which has been tested up to approximately 100 dpa [29], [30], [31]. Some studies have provided early indications of this performance. For example, it was found that SiC_f/SiC joints keep their torsional strength under 20-dpa neutron irradiation [32]. However, so far, no irradiation data with neutrons at fusion energies (14.1 MeV) and at blanket operating temperature is available. To develop and verify the material to build a commercial fusion reactor, more studies are needed.

It was found that SiC_f/SiC experiences moderate swelling, which saturates above 1-dpa irradiation at around 2% and below 1% at 300 °C and 800 °C, respectively [30]. A study on the synergic effect of He on microstructural development of irradiated SiC_f/SiC composite showed that the swelling was marginally larger than 1% (approximately 1.5%), but that up to 1000 °C, there was no void swelling [33], [34], [35]. Swelling is thus expected to be less significant at the proposed operating temperature of the SCYLLA© blanket. However, it must be noted that studies in a relevant fusion environment (high neutron flux at an energy of 14.1 MeV) are not available to date.

In general, SiC_f/SiC has a low thermal conductivity in an unirradiated state of about $\kappa = 20$ W/(mK) at 1000 °C, which quickly drops to a very low value of about $\kappa = 5.6$ W/(mK) after irradiation of about 1 dpa [30], [36], [37]. While this does not affect the performance of the heat exchanger, which remains unirradiated due to its location outside the VV, it affects the thermal gradient across the SiC_f/SiC plasma facing wall part of the blanket, thereby adding a constraint to the allowable wall thicknesses of the coolant channels and overall blanket design.

The hydrogen permeation coefficient of monolithic chemical vapor deposition (CVD) SiC is found to be very low [38], [39]. One measurement found monolithic SiC to have permeability four orders of magnitude lower than SS316 at 550 °C, with the SiC at 950 °C having similar permeability to steel (SS316) at 550 °C [40]. Hydrogen isotope permeation of SiC_f/SiC is expected to depend greatly on the production process. The nano infiltration transient eutectic (NITE) fabrication process is found to create a less porous matrix than chemical vapor infiltration (CVI) [41]. The permeation coefficient at 800 °C for NITE SiC_f/SiC was found to be $K_p = 4.2 \times 10^{-16}$ mol/s/m/Pa [7] compared to $K_p = 5.2 \times 10^{-13}$ mol/s/m/$Pa^{0.5}$ for monolithic CVD-SiC [40]. These values are, however, not directly comparable, because at low hydrogen partial pressure, the permeation flux is dependent on ∼Pa limited by surface reaction, compared to close to $\sim Pa^{0.5}$ at high partial pressure, where diffusion is rate controlling [42]. The inflection point is at around 4 kPa. In a recent measurement by KF and Kyoto University, the permeation flux was found to be more than three orders of magnitude higher at 800 °C. K_p is also found to be dependent on the tritium fraction in the hydrogen gas mixture [7]. Permeation at the joints is also a potential issue and is under further investigation. SiC_f/SiC cannot be welded, and current designs at Kyoto University use liquid phase sintering (LPS) or a combination of screw joints with reaction sintering (RS) at around 1450 °C. It is noted that known problems, such as tritium permeation and retention in graphite, which is a possible reflector for the SCYLLA blanket (see previously), may be avoided by the high hermeticity of SiC_f/SiC. Such problems will be resolved in KF's plans to utilize the experimental setup from [42] to further evaluate SiC_f/SiC permeability with LiPb, including with samples that have been irradiated.

While corrosion resistance of CVI SiC_f/SiC with pure liquid lithium was found to be poor, where both Si and C are degraded, it shows good compatibility with LiPb [16]. This was also confirmed for NITE SiC_f/SiC at Kyoto University via experimentation of a rotating disk of SiC_f/SiC in LiPb at 900 °C for up to 3000 h [6], [43].

SiC_f/SiC and tungsten (W) compatibility appears strong due to both having similar thermal expansion rates (W: 4.3×10^{-6} K^{-1}, SiC_f/SiC: 4–4.5 $\times 10^{-6}$ K^{-1}) and thermomechanical stability under neutron irradiation [44]. Tungsten reinforced SiC_f/SiC as a first wall configuration is being explored, for which a tungsten sheet and powder have been sintered with SiC fibers at 20 MPa and up to 1800 °C [32].

Manufacturability of SiC_f/SiC remains an issue. However, recent improvements have been made, including at Kyoto University, to manufacture curved surfaces and other complex shapes using prepreg sheets wrapped around a graphite fixture and then bonded by sintering [45]. Tubes of 0.9 m in length have already been assessed for fission reactor fuel cladding [46]. The long tubes were created through joining shorter tube pieces. Similarly, the SiC_f/SiC elements described in this work that were used in the LiPb loop were joined using LPS.

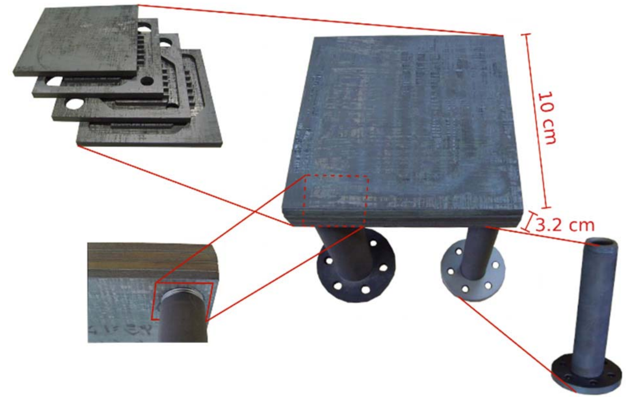

Fig. 3. Mockup of multilayered heat exchanger made of SiC_f/SiC that was used in Kyoto University's LiPb loop.

B. Liquid Lithium–Lead

Liquid LiPb fulfills three essential functions in the SCYLLA© blanket: as the breeder, multiplier, and coolant. It does also provide first-order neutron shielding that is further enhanced by a thick shielding layer attached to the back side (radially) of the breeding zone.

In the previous section, it was shown that the breeding performance (TBR) of an SCLL blanket using LiPb is expected to be sufficient for a commercial reactor. LiPb is well suited for cooling as it has a specific heat at constant pressure, c_p, at 623 K in the range of 145 and 190 J/(kg K) [11], [47], which is very high compared to gaseous blanket coolants, e.g., helium. The thermal conductivity of LiPb, $\kappa = 14$ W/(mK) at 623 K [11], is also excellent in comparison to gaseous and molten salt coolants.

As mentioned, the thermal conductivity of the structural material for the SCYLLA© blanket is expected to be low, and therefore, there is a concern of the heat exchange in the intermediate heat exchanger (IHX), where heat is transferred from the primary coolant LiPb to a secondary coolant, such as water or helium, in order to supply the power generation unit with heat to function (see "power generation system" in Fig. 4). The ex-vessel IHX system is unirradiated, and tests with unirradiated SiC_f/SiC have been conducted to see whether sufficient heat is transferred. These data are unpublished. However, the data were obtained in a proxy experiment on the LiPb loop at Kyoto University as described in [48], which used steel as the heat exchanger material but a module design similar to Fig. 3. The steel IHX module testing demonstrated that power in excess of 1 kW can be transferred between the two coolants (primary and secondary). It was found that the heat flux from LiPb to He (for a diffusion model, simply: $q = -\kappa \cdot dT$) would not be governed by the low κ of SiC_f/SiC but instead limited by the heat exchange between the solid surface of SiC_f/SiC and He and thus driven by fluid dynamics instead of diffusion. The flow, turbulence, and diffusion in the helium gas need to be better understood and supported by further experimental data. Currently, no large-scale experiment has been conducted for a heat exchanger capable of dealing with DEMO-type power output (toward and beyond 1 GWth). With higher flow rates and larger surface area, KF anticipates

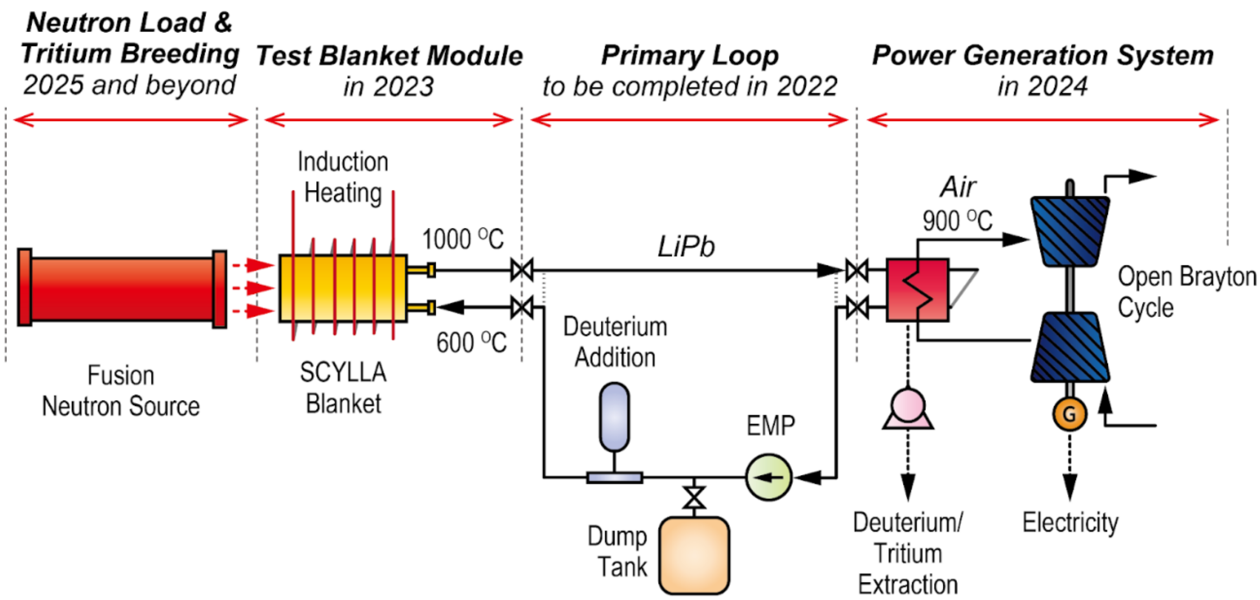

Fig. 4. Design of the integrated testing facility (UNITY) for breeding blanket functions including electricity generation and tritium recovery. Currently at the initial stage of construction in Japan.

that 10–15 kW per mock-up-sized module can be achieved based on the proposed upgrades to the previous SiC_f/SiC IHX module. This translates to about 100 000 modules in the IHX for a 1.5-GW reactor, equivalent to 32 m^3 volume of the flow elements excluding pipework. The method is applicable to a commercial fusion reactor, but progress must be made to provide a low cost and low footprint (tall structure heat exchanger). This engineering challenge needs to be addressed before reactors using a blanket can produce electricity in continuous operation.

The solubility of tritium is an important aspect for both the structural and the breeder material. Hydrogen isotopes have a low solubility in LiPb and can be extracted easily [8]. The measured solubility has, however, uncertainties on its temperature dependence, and further studies would decrease uncertainties on quantitative details for a specific reactor design. Solubility in the structural material influences the tritium inventory of the plant.

Tritium extraction from LiPb has been successfully demonstrated, and several different methods exist, including gas–liquid contactor [49], permeator against vacuum (PAV) [50], [51], [52], and vacuum sieve tray (VST) [9], [53]. The latter two can reach high extraction efficiencies of above 80%. VST is being pursued by KF, and it has been calculated that a tritium extraction efficiency at a temperature of 500 °C of 45% can be achieved with a 1-m high sieve tray. Currently, no fusion reactor breeds its own tritium, and no tritium extraction system has been built to support the breeding quantity of a breeding blanket during reactor operation, and thus, these exist as key challenges to be solved by the KF Program.

Finally, LiPb purification is necessary to remove radioactive isotopes or their precursors from the liquid to meet the proposed nuclear-related material quality assurance [54]. The use of cold traps on LiPb blanket systems will remove such impurities, to reduce the concentration of radioactive isotopes to an acceptable level, including bismuth, which under neutron irradiation produces polonium [55].

IV. Development Programme

A. Work to Date

Almost all aspects discussed in Section III for SiC_f/SiC (neutron irradiation, thermal conductivity, hydrogen isotope permeation, corrosion resistance, tungsten compatibility, and manufacturability) and for LiPb (heat extraction, T solubility, T extraction, and impurities) have been investigated at Kyoto University. Design studies have been carried out for the blanket structure, LiPb loop, heat exchanger, and VST, and models of each have been built and tested at small (laboratory) scale. Calculations for heat transfer and tritium breeding have been performed. Currently, the shielding performance and requirements for several fusion reactor configurations are being assessed. Radioisotope studies are ongoing to determine the quantity of radioactive waste produced by the blanket system, especially in regard to necessary interface materials impurities in the breeder, structural, and shielding materials.

B. Future Work—Integrated Testing Facility in Japan

The engineering challenges to take the technical readiness level of SCYLLA© from 2 to 7 to be ready to integrate it into the first-generation demonstration plants, which are expected to be constructed over the course of the next two decades. These challenges require an integrated approach to solve them, and KF has launched the construction of its UNique Integrated Testing facilitY (UNITY) in Japan in cooperation with major Japanese engineering partners. Planned to be completed in 2024, this facility will allow a small-scale SCYLLA© blanket module to be tested in similar conditions seen in the first fusion demonstration plants. This includes all features from heat extraction to electricity generation (see Fig. 4).

KF expects that the construction of this facility will bring a suite of essential technical capabilities to KF, including the following.

1) SiC_f/SiC R&D: KF will develop a cost-effective method to scale the production of SiC_f/SiC blanket components. At least two grades of SiC_f/SiC material will be made available for this purpose.
2) LiPb R&D: MHD effects under a strong magnetic field will be tested using high temperature superconducting (HTS) magnets (see the following).
3) Tritium recovery R&D: Scalability of tritium extraction from LiPb will be tested with deuterium as an analog in the first phase (~2024), based on VST technology (see Sections III-B and IV-A).
4) Pumping R&D: Pumping power requirements will be assessed for several types of pumps, with specific focus on performance in view of MHD effects (see the following).

Specific attention should be given to MHD effects, which have the potential to cause a pressure drop in the liquid LiPb flow increasing the required pumping power, which is detrimental to the performance of the blanket system. Channels made of electrically insulating material can reduce MHD effects. Theoretical predictions at UCLA have shown that SiC foam-based flow channel inserts can reduce the pressure drop up to 100 times compared to reduced-activation steel, but experiments have shown the ingress of LiPb within the foam, whereby LiPb infiltration SiC inserts negate the expected reduction in pressure loss [56]. To avoid this risk, the SCYLLA© blanket will comprise fully dense SiC_f/SiC components, which intrinsically implies no LiPb ingress. Furthermore, the radiation-induced change of electrical conductivity for SiC_f/SiC is weakly dependent on the neutron dose at the operational temperature regime of SCYLLA© (1000 °C) [41]. However, the effects of MHD on the flowing LiPb and the resulting potential impact on heat transfer, particularly surrounding the cooling of the first wall, must be investigated further as a key challenge, through both experimental research on UNITY and simultaneously through theoretical modeling and simulation.

V. Summary and Conclusion

We argue that SCLL-type blankets are an optimal choice for commercially viable fusion reactors. This conclusion is drawn from material properties under high heat and neutron irradiation, performance of cooling and tritium breeding, and recent advances in technology that make this design feasible. We provide an overview of the currently available data for the properties of the basic materials, SiC_f/SiC, and liquid LiPb and highlight the unresolved challenges to be overcome for a commercially viable breeding blanket.

An SCLL blanket is under development at KF, SCYLLA©, and key R&D work is in progress across several areas and in collaboration with public institutions, including Kyoto University, the National Institute for Fusion Science (NIFS), and the National Institutes for Quantum Science and Technology (QST), as well as industry. Foremost, a new LiPb loop that has the potential to demonstrate the first power generation from a fusion reactor blanket is discussed, with construction starting in 2022 as part of a larger integrated testing facility (UNITY) for blanket development.

Importantly, key technologies from SCYLLA©, such as SiC_f/SiC pipework, tritium extraction by VST, and heat exchanger design, will be valuable components to be applied for many reactors configuration and not just those with conventional blanket structure (e.g., liquid first wall designs). Variations and compatibility of such technologies with other liquid breeders, including molten salts, are also under active consideration by KF.

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