Tritium Fuel Cycle

Description

For the deuterium-tritium fusion fuel cycle, the large quantities of tritium required must be produced in the fusion facility itself, and is most efficiently done with lithium bearing materials. Tritium is produced by fusion neutrons interacting with Li-7 and Li-6 isotopes. Surrounding the plasma are blanket structures that contain these lithium materials, either solid (e.g. Li_4SiO_4) or liquid (e.g. liquid metal $Li_{15.7}Pb_{84.3}$). Blankets have multiple simultaneous functions, such as absorbing neutron heating, provide neutron shielding, and breeding tritium, but here the focus is on the tritium aspects. Since tritium is radioactive and easily bonds with water or other biological molecules, it must be strictly controlled, with a facility releasing only very low quantities < 1 g/year [1-4]. This is in spite of generating and handling ~ 10's of kg's annually in a Fusion Nuclear Science Facility (FNSF) or ~ 100's of kg's annually in a commercial power plant, and injecting and exhausting ~10 times these amounts into and out of the plasma chamber. Tritium, being an isotope of hydrogen, can easily migrate through systems. Most importantly it can move right through solid materials by entering their matrix and diffusing. Since the temperatures associated with the fusion core (e.g. blankets, divertors), near core (vacuum vessel, cryostat) and even apparatus beyond the core (e.g. tritium extraction, heat exchanger) range from 300-700 ^oC, tritium diffusion is rapid and will lead to tritium moving throughout these zones. Highly precise behavior predictions, control, and accounting are required to maintain the plant tritium releases to the lowest tolerable levels. Description of the various issues related to tritium breeding, extraction, processing and handling are detailed below.

1) <u>Breeder Materials Behavior in the Fusion Core</u>

In the fusion core, where the tritium breeder resides, tritium will be produced either in a liquid or solid breeder material [5]. The primary liquid candidate is Li-Pb [6], which has lead as a neutron multiplier, and generally has an enriched Li-6 fraction relative to Li-7 (natural Li is 93% Li-7 and 7% Li-6). Primary solid candidates are Li_2TiO_3 or Li_4SiO_4, which are in the form of pebble beds [7] or cellular ceramics [8]. These require an additional neutron multiplier, such as $Be_{12}Ti$ [9], and can also require Li-6 enrichment. Tritium produced in the liquid mostly stays in the liquid but will also diffuse and enter surrounding materials as it flows through the blanket and out to an extraction apparatus and heat exchanger. For the solids, the tritium must diffuse out of the solid into the open pore spaces and then into a gas stream that takes it to an extraction apparatus.

The study of liquid metals in a magnetic field is complex, and understanding the breeder flow behavior in a magnetic field under heating, high temperatures, corrosion and mass transport, and gas production and transmutation has not been established. Liquid metal science challenges break into three main areas, with strong coupling among them: MHD thermo-fluid phenomena, liquid metal interaction chemistry and mass transport, and the electrical/thermal insulator required for the liquid metal breeder to be feasible [10]. The fluid flow structure of the liquid metal in a magnetic field will have 3D and non-steady features since this type of flow will not be fully developed, and is subject to a range of flow instabilities. In addition, the asymmetry of the heating and magnetic field lead to asymmetries in the flow and interactions with the conduit walls. The corrosion of a conduit wall can be up to 10x higher due to the magnetic field and its orientation, relative to the wall and flow direction [11,12]. The high operating temperatures aggravate these mechanisms. The flow channel insert material [13] provides its own challenges, since it must provide both electrical and thermal insulation while minimizing its own interaction with the liquid metal. These phenomena are made more complex by the presence of ionizing gamma and neutron radiation in the fusion environment. The liquid metal facility at UCLA, Maple, running Li-Pb is a critical inroad to developing the knowledge base for liquid metal breeders.

Even in the case of solid breeder materials, our understanding of their behavior is quite limited. High operating temperatures are required to guarantee tritium release, while excessively high temperature will lead to sintering (coalescence of the solid, removing porosity). Processes like these are aggravated by neutron irradiation. The solid breeder material will interact with its steel container, and it will be consumed as neutrons transmute lithium into He and tritium. The other constituents (e.g. Si, O, Ti) will undergo transmutations as well. The associated neutron multiplier, a beryllium compound, will also undergo transmutation/consumption as Be transforms into helium and neutrons. These solid breeders remain in the blanket of the fusion core for extended periods and their evolution is critical to maintaining a viable blanket, while liquid breeders are continuously flowing into and out of the blanket.

2) <u>Tritium Extraction from the Breeder or Purge Stream</u>

For liquid breeders, the breeder flows out of the blanket and fusion core to a tritium extraction apparatus. The most recent examination [14] of this is targeting group 5 elements on the periodic chart, which have high permeability for hydrogen and can serve as vacuum permeation windows, with potential to remove 80% of the tritium in a single pass according to simulations. The liquid metal flows past these window materials and tritium and deuterium adsorb onto the window and then move through the material to a vacuum where it is taken to processing. Challenges for this approach lie in the high temperatures required for fusion and possible low levels of impurities that can degrade the window material. Industrial hydrogen purifiers have been produced for lower temperature operation [15], but also show that oxidation can be controlled, and interlayers may stabilize the window materials at higher temperatures [16]. This approach has not been demonstrated even on small scale experimentally, and requires a dedicated activity to establish its feasibility.

For solid breeders a purge gas (usually helium) is used to gather the tritium diffusing out of the solid and transport it out of the blanket and fusion core. The tritium must be removed from the helium gas stream. Again, a vacuum permeation window may be the best option to isolate the tritium (and deuterium) from other impurities in the gas stream. Getters are well established for removing hydrogen, and many other materials, from gas streams, but they may not be sufficiently selective to isolate tritium. Tritium will also have to be recovered from all helium cooling flows from the fusion core, and if liquid metal plasma facing components are considered, then tritium must be extracted from these fluids, which may not be the same as the breeder fluid, requiring different methods.

3) <u>Tritium Behavior in the Fusion Core, Near Core, and Tritium Intensive Apparatus</u>

Tritium will migrate throughout the fusion core from its production in the breeder, and its introduction into the plasma chamber by the fueling system. Although the behavior of tritium in a fusion system is governed by physical chemistry at a basic level, the actual environment aggravates and complicates this tremendously. The experimental data on various tritium properties used to calculate its behavior (e. g. diffusivity, solubility, and surface dissociation and recombination rate coefficients) have extremely large variations, due to practical system variations, such as the condition of a surface, or inherent difficulties in measuring very small amounts of non-radioactive hydrogen isotopes. The resulting impact on the amount of tritium that could be lost can be 50x, based on simulations to explore this impact [17]. The neutron irradiation environment will significantly aggravate properties, and likely generate synergies that must be understood to the extent possible, such as enhanced trapping of tritium in solid material due to damage or even the nanostructured particles introduced to enhance the material's radiation resistance. Multiple materials are present and in contact in the fusion core, and tritium will migrate through them and across these interfaces. Reliable tritium permeation barriers are not available in spite of decades of research to produce them [18], largely due to unknowns regarding their performance in neutron and gamma radiation and high temperature environments and over long times. Simultaneously, large amounts of tritium are injected into and exhausted from the plasma chamber (burn fraction will be only a few percent, at best), and some will be implanted in the first wall and divertor materials, will be trapped by eroded and re-deposited material, or adhere to dust. A comprehensive knowledge, and predictive capability, of the tritium behavior over a wide range of materials and environmental conditions is required to allow safe fusion systems to operate. Much of this uncertainty is associated with experiments that do not characterize the material or material surfaces sufficiently, experiments that operate at uncharacteristic hydrogen pressures, experiments that do not simulate the prototypical environments, and the ultimate aggravation that will be presented by irradiation on all these mechanisms. The US has had the deepest and most respected hydrogen research anywhere in the world, and is reflected by recent work by Causey [18] that identifies the many flaws and difficulties (complex material physics) in experiments performed over the years. The unique tritium-capable facilities at INL have been utilized by Japan for several years as part of collaborations, and the US is also uniquely qualified to pursue the complex issues associated with tritium in a fusion facility from the vast experience at LANL and SRNL (engaged in ITER tritium system design and operation).

4) <u>The Plasma Fueling and Exhaust of Tritium (and Deuterium) to/from Plasma Chamber</u>

A fusion facility will require that tritium and deuterium are injected into the plasma chamber to sustain the burning conditions, and that the unburnt fuel, reaction byproducts (He) and impurities are removed. Studies indicate [19,20] that the amount of injected fuel that is actually burned can be relatively low, and is determined by complex particle physics in the plasma, scrape-off layer (SOL) and divertor. Fueling the plasma can only be accomplished by pellet injection, since the efficiency of SOL gas penetrating the plasma is extremely low in ITER and future devices (minimizing recycling). In addition, the plasma and atomic physics in the divertor affect the accumulation of species there, and these things conspire to produce low tritium (and deuterium) consumption. The residence time of helium in the plasma, which leads to fuel dilution, and the residence time of tritium in the plasma, which leads to higher fuel burnup, are correlated. Significantly better understanding of the particle physics in a burning plasma device is required to maximize fuel burnup in a self-consistent way with exhaust, core plasma purity and pumping capability. If the tritium burnup fraction (fraction of injected fuel that is consumed in fusion reactions) is 10%, then the fueling/exhaust system is cycling 10x the amount of tritium consumed. This requires a significant inventory of tritium to be sustained in the fueling cycle. In the case of liquid metal plasma facing components, depending on the liquid metal, it can act as a getter (no recycling) or very similar to a solid (high recycling) and can influence the tritium fuel cycle.

5) Tritium Processing in the Fusion Facility

Once neutral particles are exhausted from the plasma chamber by pumping, they must be recovered, the hydrogen separated, hydrogen isotopes separated, and then sent back to the fueling system. It is possible to streamline this process if the hydrogen isotopes do not need to be separated before fueling, but this would compromise one's ability to precisely adjust the D and T injection levels, and would require careful measurement of isotope mix. ITER has provided a tremendous leap in tritium processing due its higher inventory and processing flow rates [21-24]. A key technology and safety challenge for fusion reactors is the quantity of tritium fuel being processed (2-3 kg for ITER tritium plant) and the rate at which this tritium must be processed (maximum 200 Pa-m^3/s for ITER) while at the same time minimizing tritium release to the environment during operation and under accident conditions. As illustrated in Day et al. [23], these challenges only grow in magnitude for a demonstration reactor (DEMO), where the inventory and processing rate are anticipated to increase by a factor of ~4 above ITER for a 2 GW fusion power device. The majority (~80%) of this tritium resides in the fuel processing plant's cryogenic isotope separation system (ISS) (~60%) and on the reactor's vacuum vessel (VV) cryopumps (20%). In addition it is uncertain if cryopumps will prove to be an effective VV pumping option for a steady state fusion reactor like DEMO. This uncertainty relates to possible reliability concerns for cryopumps given their transient mode of operation, i.e. cycled fuel loading and unloading modes.

A solution called the "Direct Internal Recycling" (DIR) approach has been proposed that has the potential for reducing the DEMO tritium processing plant size to that of ITER's, or 75% smaller. A key technology proposed for the DIR approach is called a "superpermeable" metal foil pump (MFP) [25-28]. The MFP is a steady state, high-temperature vacuum pump that works by directly extracting the unburnt hydrogen fuels from the plasma exhaust, instead of condensing them. Because this extracted fuel is free from plasma exhaust impurities, it can be sent directly to the reactor's fueling system for reinjection into the plasma instead of to the fuel processing plant. Another approach identified [29] is to continuously remove the hydrogen ice in the cryopump by scrapping the cryopanel, recovering the hydrogen which is then sent to the fueling system. This is based on the fact that cryopumps can differentially adsorb different materials, so that hydrogen can be isolated to specific panels.

Benefit

A fusion facility cannot function without a closed tritium fuel cycle, and this represents a fundamental feasibility issue for fusion power production. Tritium provides a difficult species for control, accounting, and safety, yet it is critical to the fusion fuel cycle. In order for fusion to realize its maximum potential for safe operation and benign environmental impacts, high fidelity understanding of all processes involving tritium is required. The tritium fuel cycle has a very broad footprint on any fusion facility, the breeding of tritium in the blanket surrounding the plasma, tritium burn fraction in the plasma, extraction efficiencies from the breeder and coolant streams, tritium processing time from plasma exhaust to fueling, tritium losses from and inventories in the fusion core, near-core and ex-core subsystems, and many more constitute a complex and interacting system. This is an essential capability for a fusion power plant, and so advances in these areas would bring a power plant to reality more quickly [30-31].

Current Status

Virtually all of the technologies related to the tritium fuel cycle are at low technical readiness, with widely varying parameters that describe tritium's migration through materials, across interfaces, and its retention in bulk solids and liquids, and retention and behavior in plasma facing materials. Extraction of tritium from breeder materials is still highly uncertain, and the development of tritium barriers has been largely unsuccessful. ITER has provided a strong step in tritium processing and the fueling/exhaust tritium loop, with higher amounts of tritium required in the future (relative to ITER). Breeder material behavior and interactions are still at a low level of understanding.

Programmatic Context

The ITER Test Blanket Module (TBM) program will be the first context where the full, integrated tritium fuel cycle environment is present. Although the amount of bred tritium to be handled is not significant, due to a small testing area and low plasma duty cycle, the tritium transport processes involved in the TBM program are representative phenomena of a DEMO FW/blanket tritium fuel cycle, including a D/T neutral ion flux implantation and consequent transport/permeation under prototypical tokamak plasma facing surface and operating conditions. However, the TBM program does not fall under the larger international ITER agreement; data from ITER testing will only be shared through "partnership." An approach could be for the US to seek supporting partnerships with two or three ITER TBM Leaders to gain access to and experience with substantial R&D results, nuclear design, instrumentation and control, safety and licensing processes, and integrated TBM testing in ITER H/D/T phases that will include all tokamak normal and off-normal operation conditions except significant neutron fluence. In such a partnership collaboration, for example, the US can contribute critical property data such as recombination coefficients, tritium diffusivity in PbLi, MHD mixed convection on tritium transport, etc., through small scale laboratory tritium experiments. It should be noted that there are very few applications for these technologies outside of fusion. A few areas of possible cooperation with Generation IV fission reactors are in lithium isotope separation (for higher concentrations of Li-6) and in tritium extraction techniques. Overall, international collaboration on the various aspects of the tritium fuel cycle and the accompanying areas of fusion nuclear materials, plasma facing materials, fusion nuclear science, and enabling technologies requires serious consideration.

Possible 15-year U.S. Research Agenda

Tritium/deuterium migration data is needed in appropriate materials, temperatures, partial pressures, surface conditions, multi-material environments, and plasma facing environments. These experiments move from basic to more integrated as the actual component and its environment are made prototypical, albeit without neutrons. Some testing with neutron irradiated samples is also required. Both solid and liquid metal breeder studies are required to understand their behavior and interactions, and requirements (e.g. insulator for LMs). Both non-nuclear and nuclear test can and should be pursued. Liquid metal loops are needed and ultimately an apparatus in HFIR could be developed. Tritium extraction schemes require R&D to establish their feasibility and optimization for the fusion environment. [32] Prototypical fluids are needed, with impurities, and at prototypical conditions is required. Exploration of approaches to fueling, exhaust and recovery processing are critical to making the tritium cycle more efficient. The tritium fuel cycle research will converge and culminate with a multi-function integrated tritium-breeding blanket that must endure the multi-physics environment of a fusion core. Apart from the tritium fuel cycle research thrust, the fusion nuclear materials and plasma facing materials areas would impact this development directly.

Research Directions Beyond 15-year Horizon

A Fusion Nuclear Science Facility (FNSF), ranging from a volumetric neutron source [33] to a FNSF [34] to a pilot plant [35] (net electricity), is the target beyond the 15-year time frame, and requires that the tritium related issues are explored and understood to a level sufficient to pursue such a device. All behavior cannot be established before the FNSF, particularly in the complex integrated environment of a fusion core, and the device will continue defining the fuel cycle requirements for a power plant. The FNSF (or similar) is the only component-level, integrated fusion facility that has been proposed internationally prior to DEMO, indicating a US strategy based on a break-in fusion nuclear step, followed by a US demonstration power plant.

Critical Objections and Advocates Response

This research can be delayed until we are ready to build a fusion nuclear device. This is generally untrue, since the R&D in the various tritium topics will require basic science and progressive experimentation toward prototypical conditions. This provides the technical basis to pursue a fusion nuclear device, and requires several years to complete. Postponing this R&D will generate a ~ 15-year delay to any decision to move on toward next steps in fusion. Feasibility demonstrations are needed to establish the credibility of approaches, and allow innovation and optimization to generate attractive solutions for the long term. Not to mention the licensing issues which could delay the construction of a power plant even if all technical obstacles are resolved.

We must choose among specific blanket concepts to make this research focused, be able to afford it, and have a reasonable timeframe. This is probably accurate; however strategies can be designed to avoid carrying several differing blanket concepts simply because they are immature in their technical readiness. Blanket concepts can be chosen by their simulated performance in a power plant, where attractive thermal conversion efficiency, tritium fuel self-sufficiency, simplicity, and long-term relevance are optimized. An attractive blanket concept can be "backed-down" by changing the most vulnerable aspects, the breeder for example. The Dual Coolant Lead-Lithium [36] blanket can be chosen as primary, with the Helium Cooled Lead Lithium [37] as first back-down where liquid metal effects are weakened while overall performance is compromised somewhat. Next a solid breeder can be envisioned as the next back-down, to eliminate the liquid metal breeder (and its complications) altogether, say with a Helium Cooled Pebble Bed [38] or Cellular Breeder concept. Due to the similar structural material and primary helium coolant, these blanket concepts can be carried more efficiently in the US program, or can involve international collaborators.

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