Fuel Cycle for an Inertial Fusion Energy Reactor: Isotope Separation and Breeder Blankets

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Submitted to the 2022 IFE community workshop to be held Feb 22–24, 2022 under the topic:
Fuel Cycle

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1.0 Introduction

The tritium fuel cycle has been studied exhaustively in the magnetic confinement fusion (MCF) arena. [1,2,3] Technologies have been developed to handle, monitor, recover from chemically bound species, enrich and store tritium at near-reactor relevant throughputs. [4] Critical components have been tested on large tokamaks or in tritium handling facilities. [5] A significant portion of that technology is transferable to systems applicable to inertial fusion energy (IFE). However, the operating conditions differ substantially from the magnetic case and as such impose conditions on the IFE fuel cycle components not found in MCF case and as a result research directed to IFE specific topics is needed.

The fueling circuit consists of an injector system and infrastructure to recover effluent from the reactor. Pellet injection in MCF is an attractive method to deliver DT ice deep into a tokamak plasma. Targets deployed in IFE reactors will need specific designs to optimize the burn fraction which can be as high as 1/3. This may entail composite layers of varying elements. Target concepts such as wetted foams will consist of liquid DT embedded in a low-density CH foam are also promising. MCF reactors will operate in vacuum with the primary constituents being the hydrogen isotopes. Some designs of IFE reactors will operate at moderate vacuums (a few torr) with the primary components being neon or xenon to help moderate the blast wave and the particle assault on the first wall. MCF reactors must contend with dust generated when the plasma interacts with the divertor. IFE reactors will need to separate and detritiate residual target debris from the volatile hydrogenic species in the effluent streams.

A generic fuel cycle for an IFE reactor is provided in Figure 1. The use of DT ice layered inside a thin-walled plastic shell is implicit in this design as a representative example. Foam filled liquid DT targets and more complicated target designs (such as those employing hohlraums) will need, depending on the details, more extensive debris collection and treatment subsystems. The fuel cycle comprises two independent circuits: one circuit to fuel the reactor and the second to breed tritium.

Reactor effluent is separated into two streams: volatile components that are cryopumped as the gas leaves the reactor while particulate debris is gravity fed into a collector and oxidized to separate absorbed hydrogen from carbon species. The cryo-separator rejects helium ash to the environment, diverts neon/xenon for re-use and discharges hydrogen isotopes to the isotope separator via a permeator. The isotope separator rejects hydrogen to the environment and directs the deuterium and tritium to the Capsule Factory and the Target Filling System.

The breeding blanket circuit has two primary functions: to extract heat from the reactor and to breed tritium. The reactor is surrounded by a molten salt bath to capture and moderate the fusion neutrons as a precursor to tritium breeding. The molten salt is pumped from the reactor through a heat exchanger, an impurity removal subsystem to purify the molten salt, a tritium extraction module and is returned to the containment vessel surrounding the reactor.

The molar flow rates are estimated for the primary species: H, D, T, C, O, He, and Xe expected in a 380 MWe IFE reactor that uses DT ice targets encapsulated in thin plastic shells. The 20 mg tritium targets are injected at a frequency of 0.5 Hz. The burn fraction is assumed to be 25%. The fusion power conversion to electrical energy is taken to be 30%. The assumed plant duty factor is 90%.

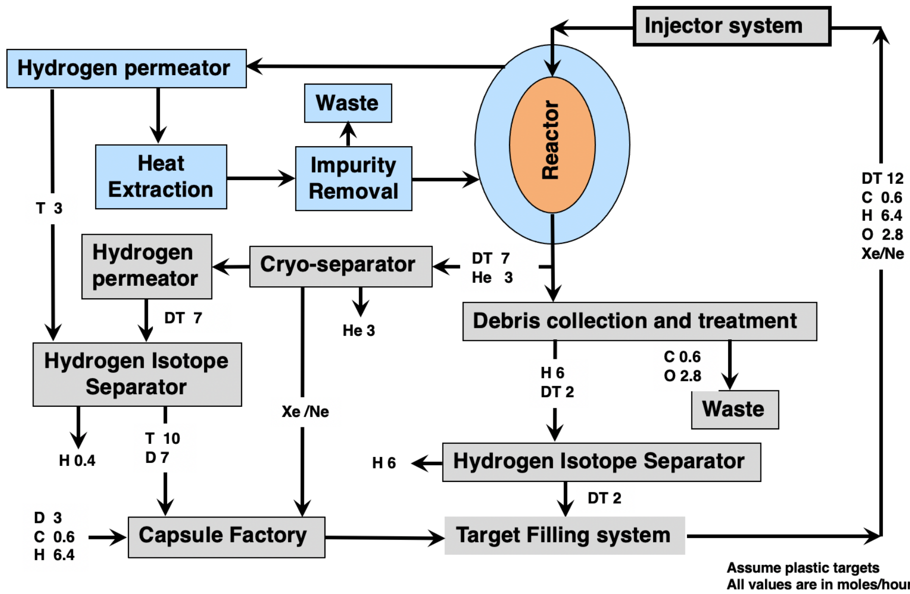

Figure 1: Fuel Cycle for a 380 MWe Inertial Fusion Energy Reactor assuming plastic targets.

2.0 Isotope Separation

2.1 Executive Summary

Developments in the art of isotope separation over the past decade offer new approaches that simplify tritium handling and improve system robustness. The Thermal Cycle Absorption Process (TCAP) configuration offers compactness, modest operational tritium inventories, high product purity, low activity raffinate and simplified operations. R&D to expand the operational space of this isotopic separation approach is needed. This section describes the state of the art and the advances that have led to the development of the TCAP.

2.2 State-of-the-Art

Several technologies are applicable to hydrogen isotope separation. These include: thermal diffusion, cryogenic distillation, displacement chromatography, elution chromatography, and thermal cycling absorption. Thermal diffusion is a slow process requiring a battery of columns and has limited enrichment capabilities. [6,7] Cryogenic distillation can achieve high purity tritium (~ 98%) but requires large hydrogen and tritium inventories and several columns to achieve the high levels of separation of the six species: H_2, HD, HT, D_2, DT, and T_2. [8] Chromatography offers a more compact approach to achieve high product purities (T_2) with smaller working inventories but requires a process loop comprising several pumps and intermediate storage devices. [9,10] Thermal cycling absorption is a relatively new concept by comparison to the other approaches. [11]

Over the past ten years significant progress has been made in improving the efficiency of the thermal cycling approach. [12,13] The addition of a cryogenic molecular sieve column to compliment an existing palladium on kieselguhr offered for the first time additive isotopic separation with atomic hydrogen separation on the palladium and molecular hydrogen separation on the molecular sieve column both favoring the concentration of the heavier species in the same direction. Thermal cycling absorption offers four advantages: simple processing loop configuration, no moving parts with the exception of one valve, and low working inventories capable of accepting a broad range of input concentrations.

2.3 Research opportunities

Presently one thermal cycling absorption system is operating at the University of Rochester to discard protium (H) and to separate deuterium from tritium. This six-liter capacity isotope separation system uses a closed loop configuration that has generated tritium purities up to 99.85% with the primary contaminant being deuterium. [12] Higher product purities appear achievable. A larger system, capable of processing up to ten standard liters of hydrogen per hour has been commissioned and will be installed into a 30-gram tritium facility in 2022. The isotope separator's function will be to increase the tritium-to-deuterium from about 85% to 98%. [14] A similar system will be constructed to reclaim and purify tritium from the SPARC tokamak effluent stream. The purification system throughput for the prototype power generating tokamak, ARC, will need to increase by a factor of three to 30 standard liters/hour. [15] Inspection of Figure 1 indicates that hydrogen isotope throughputs on the order of 192 to 240 standard liters per hour will be required to meet a 400 MWe IFE reactor's needs.

The status of the Thermal Cycle Absorption Process (TCAP) in its current configuration is in its infancy. R&D is required: to demonstrate high throughputs without compromising product purity, to tailor the working tritium inventory for specific applications, and to suppress raffinate activities to negligible values.

3.0 Blanket Technology

3.1 Executive Summary

A U.S. blanket strategic framework has been established in an effort to increase the Technology Readiness Level of the breeder blanket subsystems for Magnetic Confinment Fusion. Several national laboratories currently participate in this blanket R&D program. Some of this effort can be leveraged into the IFE program. However, this R&D program does not include two recent developments: the concept of using an immersion blanket to surround the reactor and the availability of a volumetric source of fusion neutrons. The use of an immersion blanket offers several advantages to IFE reactors. Combination of these two recent developments will help side-step many of the issues that plague the MFE solid breeder program and can help focus the IFE breeder program on increasing the technology readiness level of tritium production.

3.2 Main Line Approaches

ORNL hosted a workshop January 2019 to identify the critical R&D requirements in the blanket and fuel cycle areas. The fusion nuclear science and fusion materials community identified several high-priority tasks in the blanket arena. These included:

A. Tritium extraction from lead lithium (PbLi)
B. Solid breeder material examination and characterization
C. PbLi compatibility with RAFM steel and SiC materials, and
D. Simulations of liquid PbLi behavior

Each of these four high-priority areas were funded soon after the workshop with the objective of increasing their Technology Readiness Level (TRL).

Some progress in each of these areas has been reported. In area A, tritium extraction from lead lithium, INL is constructing a loop to extract tritium from helium purged through PbLi. SNL supports the diagnostic efforts. ORNL is providing the group five metal permeation membranes and characterizing structural materials such as RAFM and ODS steels. PNNL is examining alternative permeation membrane materials. In area B, solid breeder material examination and characterization, ORNL has studied the properties Li_2TiO_3 as a breeding material. PNNL is leveraging its materials science strengths in fission reactors to investigate the ion irradiation of Li_4SiO_4 in particular. In area C, PbLi compatibility with RAFM steel and SiC materials, ORNL has been operating a loop to expose structural materials to PbLi. UCLA has undertaken the modelling of fluid flow, heat and mass transfer of liquid metal loops. Finally, in area D, simulations of liquid PbLi behavior, UCLA has undertaken the modelling of flows in blankets under a variety of perturbations such as flow transitions and magnetic fields.

The 2019 workshop and subsequent progress by the various institutions discussed in the previous section spawned the Blanket and Fuel Cycle Strategic Framework workshop in February 2021 with the intent of coordinating the research necessary to establish the technical basis for a fusion plant as a package rather than continuing with piecemeal R&D efforts. A top-level structure for the strategic framework was developed in this workshop. As a community, U.S. researchers prefer helium primary cooling with RAFM structures as a baseline blanket design. Water cooling has been eliminated because of accidental contact with lithium compounds, activation and tritiated water handling. [16]

There is far less consensus over the selection of the breeder material within the U.S. blanket community with significant influence from the international community. Both liquid and solid options remain on the table. International fusion programs favor a $Pb_{84}Li_{16}$ eutectic for the liquid breeder option. Japan has explored using molten FLiBe but dropped it in favor of FLiNaBe. [17, 18] International solid breeder options focus on lithium metal oxides. [19] ITER will pursue four blanket concepts: water cooled lead lithium, water cooled ceramic breeder, and two helium cooled ceramic breeder concepts. [20]

3.3 Alternate Approaches

In contrast to the main line approach espoused by the U.S. blanket community, Commonwealth Fusion systems has proposed a liquid immersion blanket using FLiBe as the breeding material. [21] This approach touts increasing the breeding ratio from ~1.05 in solid breeders to more than 1.3, reducing activated waste and side-stepping material damage that will arise in currently proposed breeder configurations. The increased breeding ratio arises from reducing the quantity of structural material and concomitantly increase the volume of molten salt in the same space and from judicious placement of neutron multipliers within the FLiBe. The key technical risks center on tritium extraction from the molten salt, maintaining the purity of the salt, suppressing hydrogen permeation loses from the circuit, and quantifying the chemistry between the loop containment materials and the molten salt. An underlying risk in this and other approaches that use FLiBe is the ability to manufacture the molten salt in sufficient quantities.

3.4 New Opportunities

In July 2019, staff from Phoenix LLC and SHINE Medical Technologies LLC achieved a world record in DT neutron production. A deuteron beam was accelerated and dumped into a tritium gas target to generate $4.6*10^{13}$ fusion neutrons per second for 132 hours. The neutron fluence from this test was $2.16*10^{19}$ 14.1 MeV neutrons. The volumetric, cylindrical, 1m long, neutron source represents an ideal configuration for holistic breeding tests to investigate breeding ratios, the impact of neutron moderation and reflection and tritium extraction technologies. This new resource should be accessed to increase the TRL of blanket and loop designs, material compatibility with the molten salt and tritium extraction approaches.

There are at least two options for tritium breeding in a molten salt immersion blanket. In one, the Li in the FLiBe salt is enriched in Li-6 to directly produce tritium in the salt. Irradiation studies at MIT suggest that the primary chemical product will be TF [22]. Techniques have been studied for tritium removal from FLiBe via graphite for both TF and T_2. [23] A second option for tritium production in molten salt is suspension of lithium ceramic particles in the FLiBe. In this option, salt enriched in Li-7 can minimize tritium production directly in the salt (some production of tritium from Li-7 is inevitable in a fusion spectrum where there is a significant neutron flux with energies above the 2.47 MeV threshold). By suspending lithium ceramic particles in the salt, it might be possible to increase the overall lithium density of the immersion blanket to increase tritium production. Tritium production in fission reactors suggests that tritium release from ternary lithium oxides can be maximized, by maximizing the Li:O ratio and minimizing the O:T ratio in the irradiated material. [24] This suggests focusing on Li-rich ternary oxides such as Li_5AlO_4 or Li_8ZrO_6, which have been investigated in Japan for fabricability and thermodynamic stability. [25, 26] The theoretical Li density in these two ceramics is 0.625 and 0.694 g/cm^3, respectively, compared to the Li density in FLiBe of 0.27 g/cm^3. Even with significant porosity to facilitate rapid tritium release, these ceramics will have a higher Li density than the FLiBe, offering the potential for increased blanket tritium production compared to FLiBe. By utilizing lithium ceramics that quickly release their tritium inventory, the tritium recovery system for particles suspended in FLiBe could be similar to one proposed for tritium production in FLiBe.

R&D is required to evaluate the irradiation performance and materials compatibility of lithium ceramics suspended in FLiBe for tritium production:

- Microstructural evolution and thermodynamic stability of Li-rich ternary oxides with irradiation damage and Li burnup
- Tritium retention/release characteristics of Li-rich ternary oxides
- Leaching of Li from Li-rich ternary oxides while suspended in FLiBe and exposed to fusion neutrons
- Chemical compatibility of FLiBe with Li-rich ternary oxides
- Radiation effects and chemical compatibility of structural materials for FLiBe containment

Many of these R&D topics could be studied, at least initially, using ion irradiation as a surrogate for fusion neutron irradiation. This might be particularly useful for initial separate-effects studies to elucidate various radiation damage mechanisms. However, integral tests in a fusion neutron flux with simultaneous immersion of the lithium ceramics and structural materials in FLiBe will be required to address all of the relevant technical risks.

Radiation damage modeling could provide complementary benefits to the experimental R&D. Classical molecular dynamics simulations of energetic displacement cascades and defect accumulation could be used to understand atomic-level defect production, phase changes, and mechanical behavior of Li-rich ternary oxides. Empirical potentials based are available for some of the relevant ceramics and could be developed for the others to model these systems. The simulations would shed light on atomic-level and transient phenomena that provide insight into the microstructures seen in experimental studies and could help select materials and irradiation conditions for future R&D.

The concept of a breeding blanket that uses a flowing metal wall in IFE reactors is particularly attractive. Activation of materials outside the reactor wall is reduced to a minimum. Reactor replacement at end of life can be designed to be a 'plug-and-play' device. Penetration designs can accommodate ports for drivers and minimize perturbations in liquid metal flows. Reactor wall lifetimes can be enhanced by the continuously flowing metal. The tritium permeation direction can be tailored to favor diffusing into the reactor instead of into the reactor hall.

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