Toward the Ultimate Goal of Tritium Self-Sufficiency: Technical Issues and Requirements Imposed on ARIES Fusion Power Plants

L.A. El-Guebaly and S. Malang

April 2008
(revised August 2008)

UWFDM-1330

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

Toward the Ultimate Goal of Tritium Self-Sufficiency: Technical Issues and Requirements Imposed on ARIES Fusion Power Plants

L.A. El-Guebaly and S. Malang[1]

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

April 2008
(revised August 2008)

UWFDM-1330

[1]Consultant, Fliederweg 3, D-76351 Linkenheim-Hochstetten, Germany

Table of Contents

Abstract

Due to the lack of external tritium sources, all fusion power plants must demonstrate a closed tritium fuel cycle, meaning the blanket should supply the tritium needed for plasma operation. The tritium breeding ratio (TBR) must exceed unity by a certain margin. The key question is: how large is this margin and how high should the calculated TBR be? The TBR requirement is design and breeder-dependent. Fusion designs developed to date displayed a wide variety of calculated TBR, ranging from 1.05 to 1.8. At present, the ARIES requirement is 1.1 for the calculated overall TBR of LiPb systems. The Net TBR during plant operation could be around 1.01. The 9% breeding margin accounts for deficiencies in the design elements (nuclear data evaluation, neutronics code validation, and 3-D modeling tools). Such a low Net TBR of 1.01 is potentially achievable in advanced designs employing advanced physics and technology. Despite the complexity of the various scenarios developed so far to define the TBR requirement, it seems likely that a dedicated R&D effort will reduce the difference between the calculated TBR and Net TBR, and also understand the technical reasons for the major disagreements in the TBR requirement for designs employing various breeders. A generic breeding issue encountered in all fusion designs is whether any fusion design will over-breed or under-breed during plant operation. A shortage of tritium supply affects the operability and availability of the entire plant. On the other hand, fusion should not generate a surplus of tritium exceeding the needs for plasma fueling and for start-up of a new power plant, in order to avoid potential tritium storage problems and also for licensing considerations. To achieve the required Net TBR with sufficient precision, an online control of tritium breeding is highly recommended for all fusion designs. This can easily be achieved for liquid breeders through online adjustment of Li enrichment. A dedicated breeding-related R&D program is desperately needed in the U.S. to relax the breeding requirement, verify the ARIES requirement of 1.01 Net TBR, and bridge the gap between near-term fusion experiments and Demo or power plants.

Introduction

Since the early 1970s, the vast majority of fusion power plant studies and experiments have employed the D-T fuel cycle – the easiest way to reach ignition. Deuterium (D) exists in nature, comprising 0.015 at% of hydrogen, and is found in abundance in seawater. Tritium (T) (one of 65 naturally occurring radioisotopes) is formed in the atmosphere at a very low concentration by reactions initiated by cosmic rays. T decays by β^- emission with a half-life of 12.3 years. All D-T fueled power plants developed to date required breeding T in a blanket surrounding the plasma via the n + ^{6}Li $\rightarrow$ T + He reaction. As a result, many Li-based liquid and solid breeders were developed over the past 40-50 years to help breed T [1]. Examples include Li, LiPb, and Flibe for liquid breeders and Li_4SiO_4, Li_2TiO_3, and Li_2ZrO_3 for solid breeders. Numerous non-fusion facilities and weapon programs in the U.S. and abroad can produce kilograms of T annually. For instance, the CANDU fission reactors [2] have been recovering T from their heavy water moderator at a rate of ~1.7 kg/y. However, the annual T consumption of any fusion power plant operating at the GW fusion power level is so high (55.6 kg per full power year (FPY) per GW), meaning that all fusion power plants must breed their own T as external sources of T are insufficient, impractical, and/or inaccessible. The calculated TBR (ratio of T produced in blanket to T consumed in plasma) should be fairly accurate as an uncertainty as small as 1% translates into 1-2 kg of T per FPY for 2-3 GW fusion power. This has a significant impact that is important for a shortage of T as well as for a surplus of T.

A generic breeding issue encountered in all fusion designs is whether any fusion design will over-breed or under-breed during plant operation [3]. A shortage of T supply affects the operability and availability of the entire plant. On the other hand, fusion should not generate a surplus of T (more T than actually needed for plasma fueling and for start-up of a new power plant) in order to avoid potential T storage problems and also for licensing considerations. The real question is not whether any blanket design could eliminate the danger of T shortage or large surplus – it should – but whether it offers a practical solution for breeding adjustment (up or down) during plant operation. Noteworthy is that it is less risky to produce more T with the understanding that an online adjustment of TBR is a necessary requirement for all fusion power plants. This can be achieved for liquid breeders with fine-tuning of the ^{6}Li enrichment, as will be discussed later in Section 4.

Since the beginning of the first large-scale power plant designs in the early 1970s and continuing to the present, neutronic experts have been calculating the overall TBR for numerous designs using 3-D neutronic models that include all elements affecting the breeding significantly (such as FW, structure, divertor, penetration, etc.). There is a unanimous agreement between fusion designers that the calculated overall TBR should exceed unity by a margin. The magnitude of this margin is design and breeder-dependent. It accounts for well-known deficiencies in the nuclear data libraries, limitations of the 3-D computational models, and additional T that must be bred in excess of T consumed in plasma. The first two items tend to lower the breeding such that, during plant operation, the Net TBR is less than the calculated TBR, but still exceeds unity by a relatively smaller margin (~1%).

The historic evolution of the calculated TBR is displayed in Fig. 1 for numerous magnetic fusion energy (MFE) and inertial fusion energy (IFE) designs developed over the past 40 years in the U.S. Since the inception of the ARIES project [4] in the late 1980s, the majority of ARIES designs employed liquid breeders (Li or LiPb) as shown in Table 1. All ARIES breeding blankets must provide a calculated overall TBR of 1.1 (or more) with a Net TBR of ~1.01. For the LiPb system in particular, a rough estimate of 9% breeding margin accounts for known deficiencies in nuclear data and 3-D modeling as well as uncertainties in the design definition. This margin will decrease with time as nuclear data evaluation improves, more sophisticated 3-D neutronics modeling tools develop, and detailed engineering designs become available. Note that several U.S. projects (HAPL [5] @ NRL, Demo [6] @ UCLA, and IFE [7] @ LLNL) along with some European [8,9] and Japanese [10] studies accord with the 1.1 ARIES calculated TBR as a minimum design requirement.

The Net TBR during plant operation could be as low as 1.01, much lower than the calculated TBR. As will be explained shortly in Section 2.4, for advanced designs with high fractional T burn-up in the plasma (> 10%), the Net TBR depends mainly on the T inventory and required doubling time (defined as the time needed to supply a new power plant with a start-up T). Early generations of fusion plants may require a Net TBR > 1.01 for shorter doubling time while a mature fusion system may call for 1.002 < Net TBR < 1.01. Moreover, fusion plants may not operate in a uniform manner, generating more (or less) T during operation according to the need for variable doubling time, the need for higher/lower breeding over a certain time period (with the same integral amount of T over blanket lifetime), the availability of T from the detritiation system, and/or the Li burn-up of ceramic breeders. This statement strongly supports the argument that an online adjustment of breeding is highly desirable for all fusion power plants.

An extensive research and development (R&D) program is needed to reduce the gap between the calculated TBR (~1.1) and Net TBR (~1.01). Section 2 covers in detail the breakdown of the breeding margin (calculated TBR – Net TBR), Section 3 describes the sensitivity of TBR to the various fusion concepts and design options, Section 4 outlines a feasible scheme for the online adjustment of breeding, and Section 5 documents the R&D programs needed to demonstrate T self-sufficiency, highlighting the ongoing and future activities for solid and liquid breeders.

Table 1. Calculated TBR for ARIES MFE designs [4]

ARIES Designs		Blanket Concept	Calculated TBR	^{6}Li Enrichment
1992	ARIES-II	Li/V	1.12	Nat. Li
1992	ARIES-IV	Li_2O/Be/He/SiC	1.12	Nat. Li
1994	SPPS	Li/V	1.1	Nat. Li
1996	ARIES-RS	Li/V	1.1	Nat. Li
1999	ARIES-ST	LiPb/He/FS	1.1	60%*
2000	ARIES-AT	LiPb/SiC	1.1	90%
2006	ARIES-CS	LiPb/He/FS	1.1	70%#

* Enrichment dropped from 90% to 60% in response to FW design modification that came very late in the design process calling for less FW structure.

Offers flexibility to increase TBR during operation, if needed.

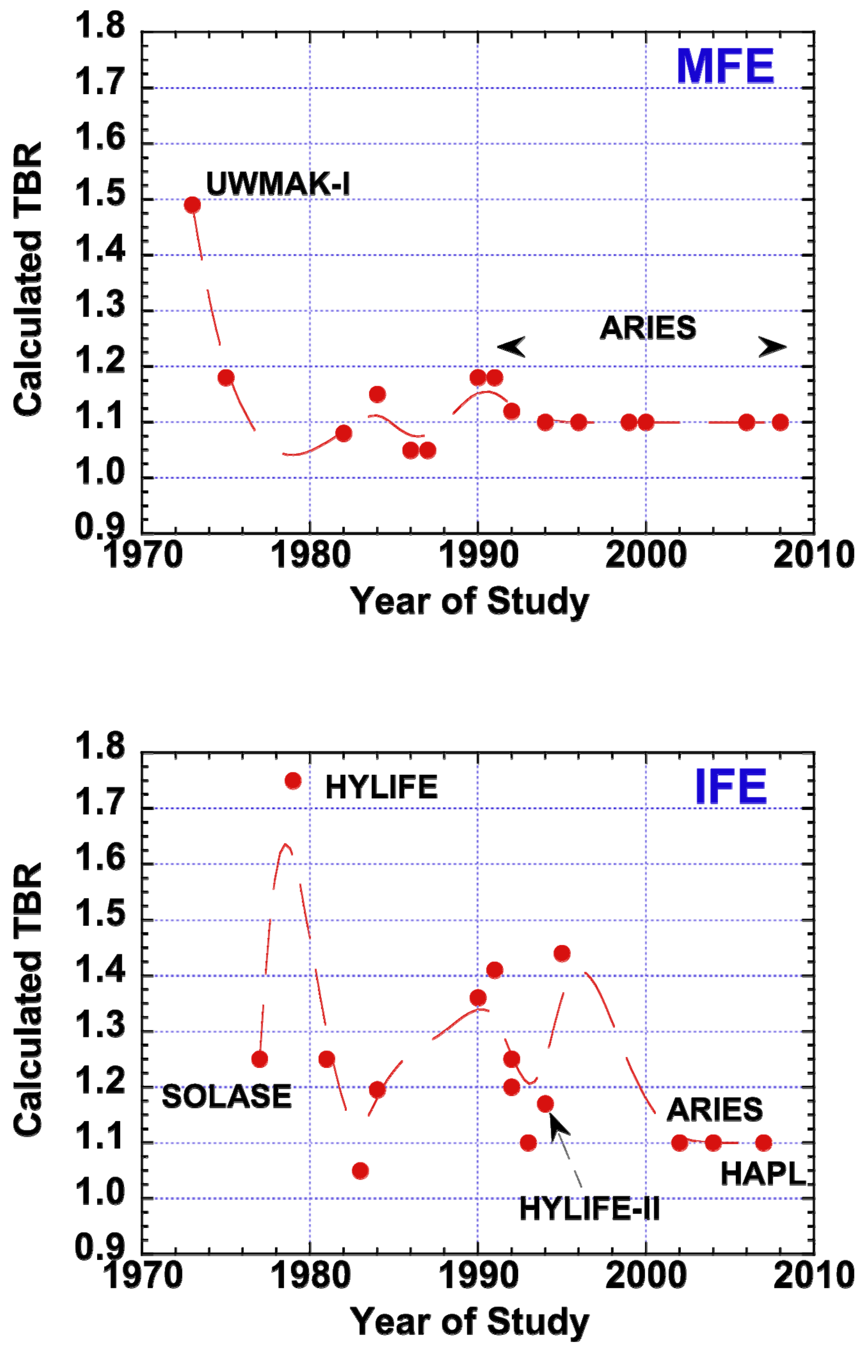

Fig. 1. Evolution of calculated TBR for U.S. MFE and IFE fusion power plant designs. The dashed curve is included just to guide the eye.

2. Breakdown of Tritium Breeding Margin

Essentially, the breeding margin (calculated TBR – Net TBR) can be divided into four distinct categories as illustrated in Fig. 2 and listed below in descending order:

- Margin that accounts for known deficiencies in nuclear data (6-10%)
- Margin that accounts for known deficiencies in modeling (3-7%)
- Margin that accounts for unknown uncertainties in design elements (0-3%)
- Margin that accounts for T bred in excess of T consumed in plasma (1-2%).

The values between brackets indicate the expected contribution of each category to the breeding margin of the LiPb system in particular. The reason for the wide range is that some data are incomplete and/or some values depend on the sophistication of the modeling tools. Building upon what has been accomplished over the past few decades and based on our extensive experience in blanket designs, educated guesses have been made whenever data are scarce or missing. The lower end of the range represents the adopted reference values for the ARIES LiPb design system utilizing the newly developed CAD/MCNPX approach [11] to calculate the overall TBR and including in the 3-D model all the engineering details that affect the breeding significantly.

Detailed numerical models to evaluate the breeding margin have been developed by UCLA over the past two decades [12-16] and more recently presented at the ISFNT-8 symposium in 2007 [17]. The 1990s numerical uncertainties/sensitivity analyses [13,14] predicted 2-6% change to the calculated TBR due to uncertainties in nuclear data. More recently, experimental results indicated larger uncertainties (10-20%) particularly for solid breeders (SB), as discussed shortly. References 12 and 15-17 emphasized the dependence of the required Net TBR (~1.05) on the physics and technology parameters of fusion designs. Section 2.4 confirms this statement and presents a simple but accurate approach developed specifically for advanced ARIES designs, describing and quantifying the essential elements comprising the ARIES Net TBR (~1.01).

2.1 Breeding margin that accounts for known deficiencies in nuclear data

Tritium production within any blanket is highly sensitive to the neutron energy spectrum that is controlled by the nuclear data evaluation. This evaluation is quite complex and involves several data processing systems for numerous cross-sections and isotopes (e.g., 20-30 isotopes in ARIES blankets).

During the past 50 years, numerous organizations in the U.S. and abroad developed nuclear data libraries for fusion applications. Such efforts faced many challenges including compilation, measurement, and processing of basic nuclear cross-section data for all nuclides and isotopes encountered in fusion designs. Cross-section measurements usually include secondary neutron energy and angular distributions that are difficult to measure. The FENDL library [18], developed by IAEA in the early 1990s, is widely used worldwide because of the high fidelity of reliability and quality assurance. The data were carefully selected from several national libraries (developed mainly in the U.S., Japan, and Europe) and tailored to meet the needs of the fusion

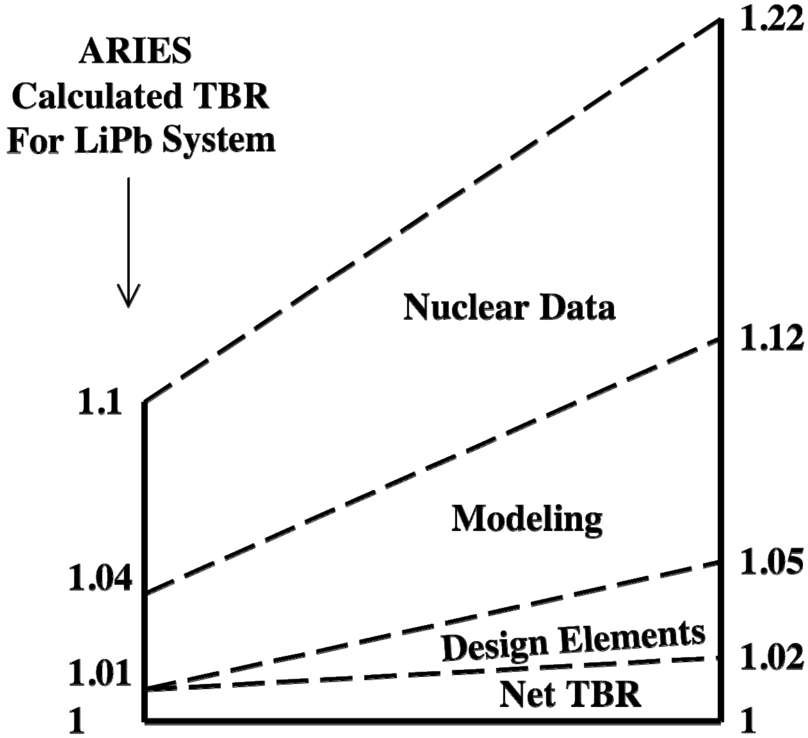

Fig. 2. Calculated TBR for a range of breeding margins.

community. The latest FENDL-2.1 version was released in 2004. Despite the high fidelity in the IAEA evaluation, the FENDL-2.1 version is far from perfect. Currently, there is a strong request for a new version based on feedback from the fusion neutronics community. Issuing a new version of the nuclear data will take years of an extensive experimental program combined with data re-evaluation, then data validation.

In the 1980s and 1990s, the impact of uncertainties in nuclear data evaluation on the calculated TBR was assessed numerically for several breeders [13,14]. Integral validation of the data libraries and 3-D codes (like MCNPX [19]) used in the fusion neutronics area is an inevitable step as feedback from integral nuclear testing in a 14 MeV neutron environment leads to significant improvements to cross section evaluations. At present, there is no fusion experimental program to validate nuclear data in the U.S. but three integral experiments are currently under operation in Japan, Italy, and Germany. These experiments are dedicated to T production rate measurements among other tasks. However, such small-scale integral experiments have their own limitations (such as limited neutron source strength and low neutron flux within mockups). Therefore, uncertainties in the T production rate measurements will exist for some time until testing is performed in a real fusion environment (e.g., in ITER [20] or component test facilities (CTF)).

In Japan, a series of integral experiments has been performed over the past 20 years at the JAEA Fusion Neutron Source (FNS) facility [21] using 14 MeV neutrons to validate the nuclear data and compare calculated to measured T production rates for ceramic breeders [22,23]. The program examined many blanket mockups to determine the magnitude and direction (positive or negative) of the change in the T production rate due to uncertainties in the IAEA FENDL-2.1 nuclear data for several ceramic breeders, multipliers, coolants, and structural materials. The most recent results for the Li_2TiO_3/Be/FS blanket indicated the calculations overestimate the T production rate by up to 10-20% [10,23]. More experiments and R&D programs are ongoing to examine advanced blanket concepts using LiPb, Li, and Flibe [22].

In Europe and within the framework of ITER R&D activities, numerous benchmark experiments were carried out at the Frascati Neutron Generator (FNG) [24] (at ENEA in Italy) and at the TUD neutron generator [25] at Dresden Technical University in Germany. Among a variety of nuclear-related activities, these 14 MeV neutron experiments have so far validated nuclear data for fusion relevant materials and measured T production rates for solid breeders. The experiments demonstrated that the T production is predicted within 5-10% uncertainty for solid breeding blankets utilizing Be multiplier [26]. Benchmarking the Li_2TiO_3/Be results showed a good agreement between the FNS and TUD experimental T production rate results, while the ENEA results indicated a 15-20% underestimation [23].

As noticed, none of the Japanese and European experiments have examined LiPb – the preferred breeder in the most recent ARIES designs. According to References 22 and 26, new experiments are underway in Japan and Europe for the helium-cooled LiPb blanket option as part of the ITER Test Blanket Module (TBM) activities. To be completed in a year or two, these experiments will provide more relevant results for the ARIES project regarding the uncertainties in the T production rate for LiPb.

Normally, the recommendations of such R&D program would be used to re-evaluate the nuclear data and data processing systems for breeders, structure, and multipliers through numerous experiments for the individual isotopes. Several iterations continue between data evaluation and experimental validation until a good agreement is reached. Only ITER TBM and CTF will accurately measure the T production in a prototypical fusion environment (neutron spectrum, surrounding components, etc.). In absence of a dedicated U.S. breeding-related R&D program, the best approach for the ARIES project is to continue including an adequate breeding margin (~6%) in the calculated TBR of the LiPb system to account for the nuclear data deficiency until the Japanese and Europeans conduct the LiPb experiments, benchmark, and publish the final results.

2.2 Breeding margin that accounts for known deficiencies in modeling

Calculating the overall TBR for any fusion system requires advanced neutronics tools. The newly developed Computer Aided Design-Monte Carlo Neutron and Photon (CAD-MCNPX) code [11] provides such a capability. It imports a CAD model of the design, eliminating modeling errors and allowing faster design iterations, if needed. Ideally, the 3-D model should include the essential components that impact the breeding significantly. These are the FW, blanket, shield, stabilizing shells, divertor, penetrations, and assembly gaps. The FW, innermost segment of the blanket, and stabilizing shells should be modeled in detail as they control the breeding level significantly (refer to Section 3 for more elaborate discussion). Practically, the 3-

D model cannot represent to a great extent the real geometry, particularly for a complex blanket design. A very detailed blanket is too costly to model. Homogenization overestimates the breeding level and the calculated 3-D TBR should be adjusted accordingly. The margin of error in the calculated TBR due to modeling could range between 3 and 6%, depending on how crude the 3-D model is.

2.3 Breeding margin that accounts for unknown uncertainties in design elements

This section introduces additional concerns that should be further studied to quantify their impact on the Net TBR. Normally, the TBR is calculated for conceptual designs where the major elements that degrade the breeding (such as FW, blanket structure, and penetrations) are to great extent included in the 3-D model. As the design develops further approaching the construction phase, several design changes may negatively affect the breeding, calling for a larger breeding margin during the conceptual phase, if the design allows. These changes include, but are not limited to, adding 1-2 mm W armor on the FW to enhance the plasma performance, more supporting structure for the FW and blanket, thicker SiC insulator for the dual-cooled lead-lithium blanket concept, larger stabilizing shells, sizable penetrations, and wider assembly gaps. References 1, 27 and 17 cover the negative impact of selected design changes on the TBR prediction. Over ~20 years of ARIES studies, no provision was made in the calculated TBR to account for future design changes. This means such design changes will require higher enrichment and/or redesigning the blanket to meet the breeding requirement.

2.4 Breeding margin that accounts for T bred in excess of T consumed in plasma

The amount of T to be bred in excess of the amount consumed in the plasma can be divided into three main categories:

1. T required to provide the start-up inventory for a new fusion power plant:
    a. T build-up in the power core materials (especially in breeder, multiplier, structural materials) and the T recovery system for the blanket
    b. T build-up in the fuel reprocessing system (especially in cryo-panels, getters, molecular sieves)
    c. T build-up in the detritiation systems for coolants, building atmosphere, and vacuum pumping system
    d. T to be stored in getters as reserve to continue plasma operation in case of temporary malfunctions of the T reprocessing system
2. T necessary to compensate for the decay of the total T inventory
3. T lost to the environment (atmosphere, cooling water, etc.)

After the following discussion on the evolution of the T inventory with time, Subsection 2.4.1 will quantify the excess breeding for each category. At the beginning of new power plant operation, a certain amount of T generated in the breeding blanket will not be available for plasma fueling as it will be trapped in all power core materials as well as in the fuel reprocessing system. Even if both inventories have to be made as small as possible for safety reasons, these holdups of T in the plant system decrease the amount of T stored in getter beds. The stored T maintains a minimum amount of reserve T that is necessary to enable continuing plant operation in case of temporary malfunctions in the T processing system. Here, we describe the T situation after the start-up of a new plant, focusing on liquid breeder blankets:

1. At time zero, there is certain T inventory for start-up stored in getter beds ($T_{start\text{-}up}$).
2. The first T generated in the blankets will be trapped in the breeding material and blanket structure before T extraction in the external loop can be started. For a certain time period, the T needed to fuel the plasma has to be taken from the start-up inventory, decreasing the amount of T stored in getters.
3. From the beginning of operation, the plasma exhaust will be fed into the fuel reprocessing system where T will be stored in getters or cryo-panels before it can be extracted for the re-fueling system.
4. The amount of T stored in the fuel re-processing system increases with time and levels off after a certain time (approximately a few weeks until the inventories in all systems reach their equilibrium value). Up to this point in time, the amount of T in the storage (getter beds) decreases from the original start-up inventory down to a minimum value necessary for assuring continuous operation during malfunctions in the fuel reprocessing system (T_{min}).
5. When this equilibrium level is reached, the amount of T extracted during the fuel reprocessing time will be larger than the amount needed for the fueling system. This excess T will be stored in the reserve storage, increasing its inventory continuously.
6. After certain time, the amount of T stored will be equal to the sum of $T_{start\text{-}up}$ and T_{min}, allowing the delivery of $T_{start\text{-}up}$ to a new plant. The time from the beginning of plant operation up to this point in time is called the doubling time. Depending on the relationship between doubling time and blanket lifetime, we have to consider different scenarios:
    - 6.1 Blanket lifetime > doubling time: In this case, the full amount of $T_{start\text{-}up}$ can be delivered to the new plant when the doubling time is reached.
    - 6.2 Blanket lifetime < doubling time: A blanket exchange leads to a temporary shortage of the T in the reserve storage since a small part of the T produced in the blankets is used to provide the T holdup in the blanket structure. This decrease can be compensated for by feeding back the T recovered from detritiation of newly replaced blankets. After this process, the T stored in the reserve storage will start to rise again until it reaches $T_{start\text{-}up} + T_{min}$, allowing the transfer of $T_{start\text{-}up}$ to a new plant when the doubling time is reached.
    - 6.2 Blanket lifetime = doubling time: In this case, there are two possibilities, depending on the time it takes to detritiate the replaced blankets. If this process is faster than the downtime of the plant, the amount of T gained by this detritiation can be fed back to the reserve storage, and the full amount of $T_{sart\text{-}up}$ can be delivered to the new plant immediately upon shutdown of the first plant. If detritiation takes longer than the downtime, the delivery of the full start-up inventory will be delayed slightly. Considering the short delay for the delivery of a small fraction of $T_{start\text{-}up}$, this will have no major impact on the start-up of the new plant.

2.4.1 Rough estimate of T inventories in different components of ARIES power plants

To quantify the different T inventories, we used ARIES-CS [28,29] and ARIES-AT [30,31] designs as examples. Both designs employ $Li_{17}Pb_{83}$ as the breeder/coolant. Throughout the write-up, the main estimates are made for ARIES-CS while the ARIES-AT values are included in parenthesis. The key parameters for this breeding-related assessment are:

	ARIES-CS	ARIES-AT
Net output power (MW_e)	1000	1000
Fusion power (MW)	2436	1759
Burn-up fraction of T in plasma	12.4%	36.4%
T consumption*: in kg/FPY	135	97.8
in kg/day	0.37	0.268

* Based on 55.6 kg T consumed per FPY for 1 GW fusion power.

2.4.1.1 T required for start-up inventory of new fusion power plant:

As a rough estimate, it is assumed that any power plant has to provide the required start-up inventory for a new power plant every 5 years. If we add the T inventories estimated in the subsections below, we arrive at a total amount of 4 kg in ARIES-CS with the reference dual-cooled lead-lithium (DCLL) blanket, 2 kg in ARIES-AT with LiPb/SiC blanket, and 5 kg in ARIES-CS with solid breeder blanket. This means that a start-up T inventory of 4 kg would be sufficient for ARIES-CS with LiPb blankets (2 kg for ARIES-AT) to allow for the build-up of T inventories in all power core materials as well as in the fuel reprocessing system, and enabling one day of continued plant operation in case 1/3 of the fuel reprocessing systems is not available for this period of time.

a- *T build-up in power core materials and T recovery system for blanket:* At the beginning of operation with new blanket modules, the T bred in the breeder material is used first to build up a T inventory in the breeder. A very rough estimate of this inventory is 100 grams in Pb-17Li, 500 grams in ceramic breeder, and 500 grams in beryllium multiplier for the entire plant. The amount of T stored in the FW and structural materials is estimated as <1 kg in total for steel structure and ~0.7 kg in SiC structure (ARIES-AT). Altogether, this leads to a total T inventory in the power core materials of ~1 kg for ARIES-CS, ~0.8 kg for ARIES-AT, and ~2 kg for ceramic breeder blankets.

After 4-5 years of operation, the FW, blanket, and divertor will need to be replaced with new components for radiation damage considerations. It is assumed that a large fraction of the T holdup inventory will be immediately recovered by detritiating the in-vessel components at the end of their service lifetimes. If detritiation takes longer than 50 days (the projected maintenance time), the delivery of a small fraction of the start-up inventory to the new plant will be delayed slightly.

b- T build-up in fuel reprocessing system: With a fractional burn-up rate of T in the plasma of 0.124 for ARIES-CS (0.364 for ARIES-AT), the throughput of T in the fuel reprocessing system is about 3 kg/day (0.74 kg/day). If we assume a reprocessing time of 1 day, the total T inventory in the reprocessing system is assumed to be about 50% of the daily throughput, or 1.5 kg (0.37 kg).

c- T build-up in detritiation systems for coolants, building atmosphere, and vacuum pumping exhaust: There is continuous T permeation into the helium coolants and from the pipes of the primary cooling systems into the building atmosphere. These losses are recovered in various detritiation systems, leading to a certain T inventory. A rough estimate is that this amounts to ~ 0.5 kg under equilibrium conditions.

d- T to be stored in getters to bridge temporary malfunctions of fuel reprocessing system: For safety reasons, the reprocessing system will be subdivided into a number of independent systems, also providing some redundancy. If there are, for example, three systems with a T throughput of 1 kg/day (0.25 kg/day) each, and we assume that one of these systems is not available for one day, we need ~ 1 kg (0.25 kg) T in reserve to allow for unperturbed refueling of the plasma. This amount could be too high because this excess T would increase the total T inventory and may become a safety issue. To avoid this situation, a better option is to install a larger number of reprocessing units than actually needed and operate them at a partial capacity. This means the need for T reserve to bridge any temporary malfunctions could be reduced (or eliminated) by redundancy at a higher capital cost for the T reprocessing system.

2.4.1.2 T necessary to compensate for decay of total T inventory
The T inventory in the plant decays at a rate of 5.47% per year. This means that with a total initial inventory of 4 kg plus an average amount of 2 kg for the start-up of a new plant in ~5 years, ~0.33 kg T will decay in a year (~0.16 kg for ARIES-AT).

2.4.1.3 T lost to environment
For environmental reasons, the entire losses of T have to be limited to 20-100 Ci/day, or 0.002–0.01 g/day. This means the total T losses to the environment must be negligible (< 4 g/y).

2.4.2 Dynamics of T inventory in different plant components
In the example above, the ARIES-CS plant starts operation with a start-up T inventory of 4 kg (2 kg for ARIES-AT). As mentioned before, at the beginning of operation when most of the T bred in the blankets is used to build up the inventories in the blanket materials, no T can be extracted from the T recovery system. There will also be considerable T holdups in the fuel reprocessing system until the T inventory reaches equilibrium. This means that most of the T consumed in the plasma has to be taken from the plant start-up inventory during this period of time. This inventory would be sufficient to subsidize the entire T recovery system for up to ~11 days (~7 days for ARIES-AT). However, with the build-up of the inventories in power core materials, blanket T recovery system, and fuel reprocessing system, an equilibrium will be reached after a

few weeks, and there will still be a buffer inventory left to bridge temporary malfunctions of parts of the fuel reprocessing system. As assumed in Section 2.4.1.1-d, a remaining reserve of 1 kg (0.25 kg for ARIES-AT) would be sufficient to allow for continued plant operation if 1/3 of the fuel reprocessing system is not working for one day. This may appear as a short period, but a larger buffer could lead to a very high vulnerable T inventory, causing a safety concern. An alternate solution to allow for a longer operating time, despite some malfunctions, would be to have a redundant fuel reprocessing system and/or to start the plant with a considerably higher TBR.

2.4.3 Net TBR

When an equilibrium T inventory is reached in all components, the only losses to be compensated for are the decay of T (0.33 kg/FPY for ARIES-CS and 0.16 kg/FPY for ARIES-AT, see Section 2.4.1.2) and the permissible losses to the environment (< 4 g/y, see Section 2.4.1.3). The required excess breeding ratio above unity to compensate for these losses would be DELTA-TBR = 0.33/135 = 0.0024 (0.16/97.8 = 0.0016 for ARIES-AT). If ARIES-CS could operate with a Net TBR of 1.01, the excess T production would amount to (0.01-0.0024) * 135 kg/FPY= 1.026 kg/FPY, corresponding to a doubling time of 4/1.026 = 3.9 y. Note that a Net TBR of 1.05 would lead to an excess T breeding of (0.05-0.0024) * 135 kg/FPY = 6.43 kg/FPY or a doubling time of ~0.6 y, much faster than realistically required.

Overall, it can be concluded from this example that, ideally, the Net TBR rate must be at least 1.0024 to ensure T self-sufficiency for ARIES-CS. If it is 1.01, the start-up inventory for the next power plant is gained with a doubling time of 3.9 y. This doubling time decreases to ~1.7 y with a Net TBR = 1.02. Figure 3 displays the sensitivity of Net TBR to doubling time for both ARIES-CS and ARIES-AT.

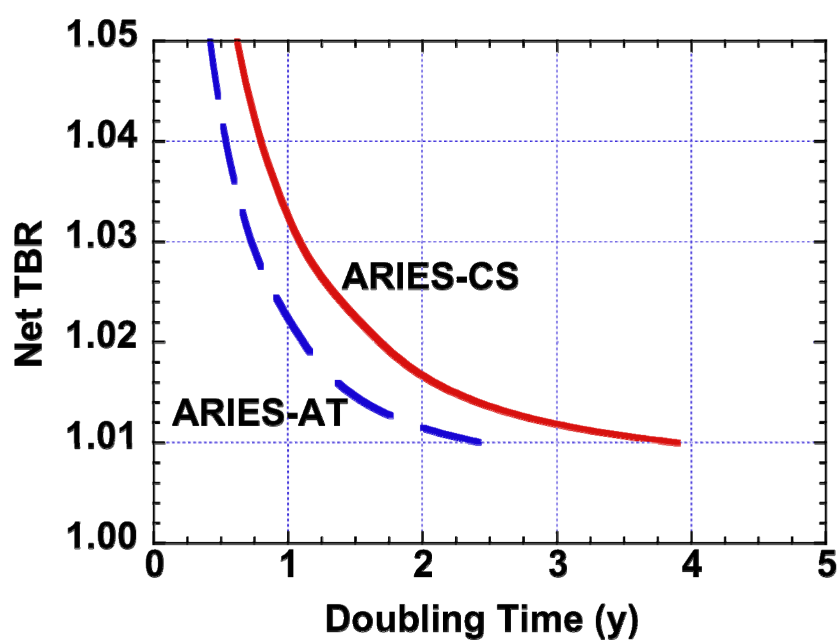

Figure 3. Sensitivity of Net TBR to doubling time.

2.4.4 Summary

The key findings of Section 2 are summarized below:

	ARIES-CS	ARIES-AT
T burn-up rate (kg/day)	0.37	0.268
Fractional burn-up (%)	12.4	36.4
T throughput (kg/day)	3	0.74
T holdups in LiPb breeder (kg)	0.1	0.1
T holdups in structure (kg)	~1	~0.7
T inventory in reprocessing system (kg)	1.5	0.37
T build-up outside FPC (kg)	0.5	0.5
Stored T for malfunctions (kg)	1	0.25
T decay (kg/y)	0.33	0.16
T losses to environment (g/y)	< 4	< 4
Start-up inventory (kg)	4	2

Advanced physics and technology help keep the Net TBR around 1.01. Based on the discussion above, the most essential requirements include:

- T burn-up fraction in plasma exceeding 10%
- T and α particles recycled at high rates
- Low T inventory in all subsystems
- High reliability and short repair time (< 1 day) for T processing system
- Two or more T processing system
- Extremely low T losses to environment (< 4 g/y).

In reality, the Net TBR will not be verified until after the operation of a Demo plant with fully integrated blanket and T extraction and processing systems. We will certainly know what to design for before building the first generation of fusion power plants and any existing blanket design will be redesigned accordingly. Therefore, all blanket designs should be flexible and be able to accept a few necessary changes in order to deliver a Net TBR of 1.01. For liquid breeders, the most practical solution is to adjust the ^{6}Li enrichment online, as will be explained Section 4 for an over-breeding blanket (Net TBR > 1.01). For solid (or ceramic) breeders, a practical solution is to adjust the ^{6}Li enrichment after the first blanket change-out or replace a few breeding modules by shielding components. In case of under-breeding blanket (Net TBR <

1.01), major design changes are anticipated in order to raise the TBR, unless the reference blanket parameters are determined for ^{6}Li enrichment < 90%. These major changes include thickening the blanket, replacing the W stabilizing shells of ARIES-AT by Al or Cu shells, lowering the structural content within the blanket, adding a beryllium multiplier to the blanket, increasing the plasma aspect ratio, or operating tokamaks in a single-null mode. Some of these changes may not be feasible during operation. It is therefore less risky to design an over-breeding blanket (with Net TBR of 1.01 - 1.02) and in the meantime develop a feasible scheme to adjust the breeding shortly after plant operation. Supporting calculations should be performed prior to plant operation in order to assess the impact of the preferred change(s) on breeding as well as on related design parameters (thermal, shielding, configuration, costing, etc.).

3. Impact of Fusion Concepts and Design Options on Breeding Capacity

The attainable TBR is design dependent – some fusion concepts and designs offer more breeding than others. The breeding capacity of any blanket system varies widely according to the preferred fusion concept [27]. For instance:

- IFE system provides > 98% blanket coverage with no restrictions on blanket size ⇒ higher breeding compared to MFE
- In tokamaks:
    - Inboard (IB) and outboard (OB) blankets provide all T breeding
    - Massive divertor structure precludes breeding behind divertor system
    - Constraints on inboard blanket thickness affect breeding
    - Homogenization of FW and blanket overestimates TBR by up to 6% [17]
    - Penetrations on outboard midplane degrade breeding by 2-3%
- Spherical tokamaks limit breeding on inboard, forcing outboard-only blanket to provide all required T [32]
- Stellarator blankets surround the plasma entirely and offer the advantage of breeding behind thin divertor plates/baffles [29]
- Linear systems (like tandem mirrors) offer negligible end losses, maximizing the breeding.

The key elements that determine the breeding level for MFE systems include:

- Blanket concept:
    - Breeder (liquid breeders (Fig. 4-A), solid breeders (Fig. 4-B))
    - Structure (SiC/SiC, C/C, V, FS, Ti, W, TZM alloys) [1,17]
    - Coolant (He, H_2O, liquid breeders)
    - Neutron multiplier (Be, Pb, Be_2C, BeO) [1]
    - MHD electric insulator (Fig. 5) [33]
- FW and Blanket configuration:
    - IB and OB neutron coverage fraction (Fig. 6)
    - Thickness (Fig. 7) and structural content [1,17]
- Penetrations:
    - Divertor, plasma heating/control, diagnostics, laser/HIB beam-lines
    - Number, size, location.

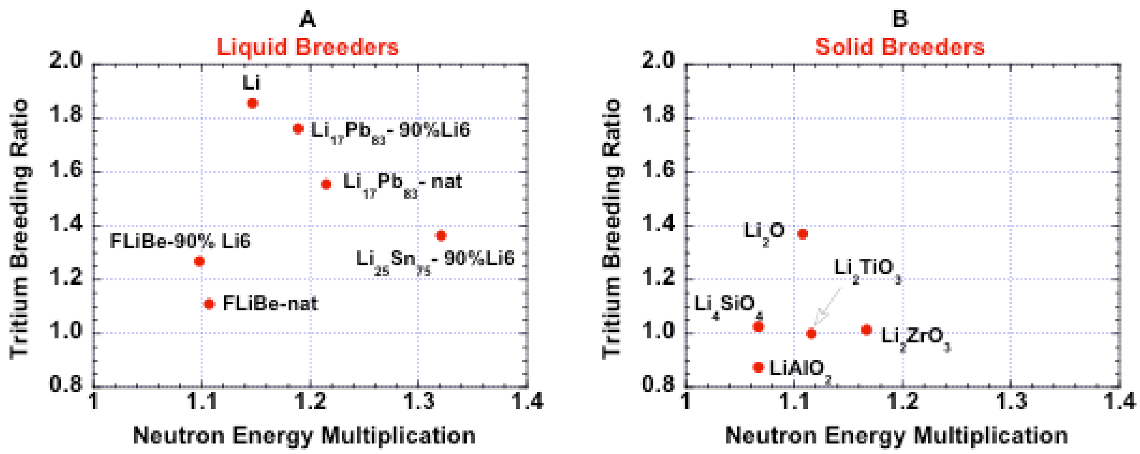

Fig. 4. Local TBR vs. energy multiplication for selected liquid and solid breeders (100% dense materials at room temperature evaluated with FENDL-2.1 data [18]). Natural $Li_{25}Sn_{75}$ (not shown) provides a very low TBR (0.45). Note that some breeders may contain a neutron multiplier (e.g., Pb in LiPb and Be in Flibe).

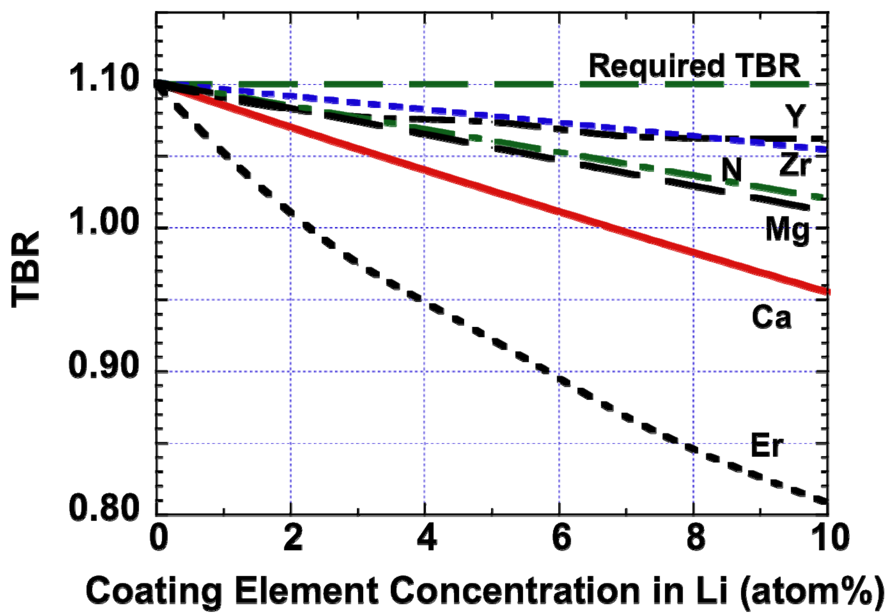

Fig. 5. Relative effect of coating materials on breeding of Li/V system [33].

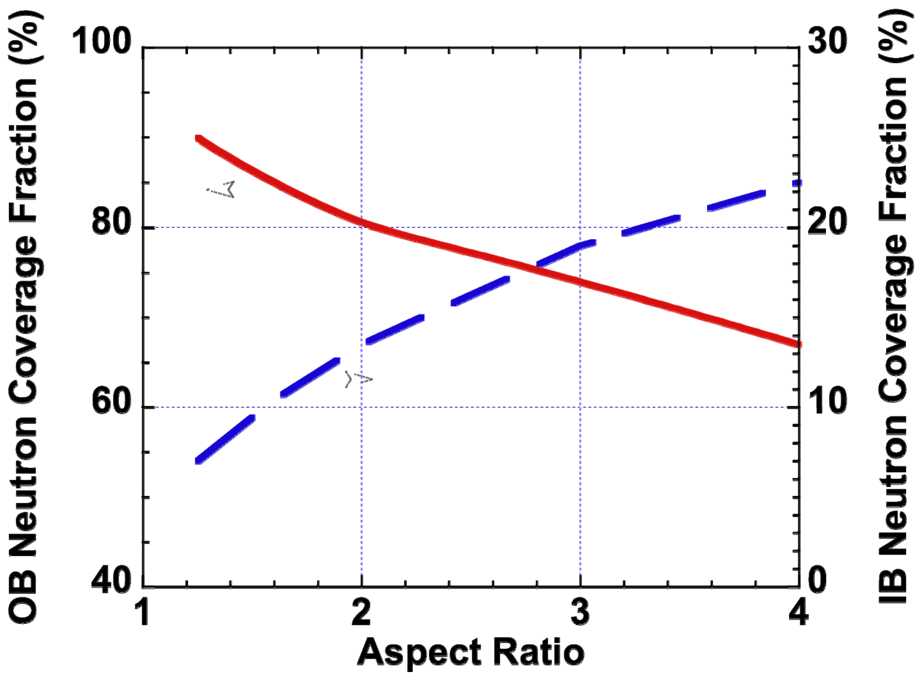

Fig. 6. IB and OB coverage fractions (defined as fraction of source neutrons incident directly on FW) as a function of aspect ratio [32]. This plot confirms the importance of OB for breeding, particularly for devices with low aspect ratio.

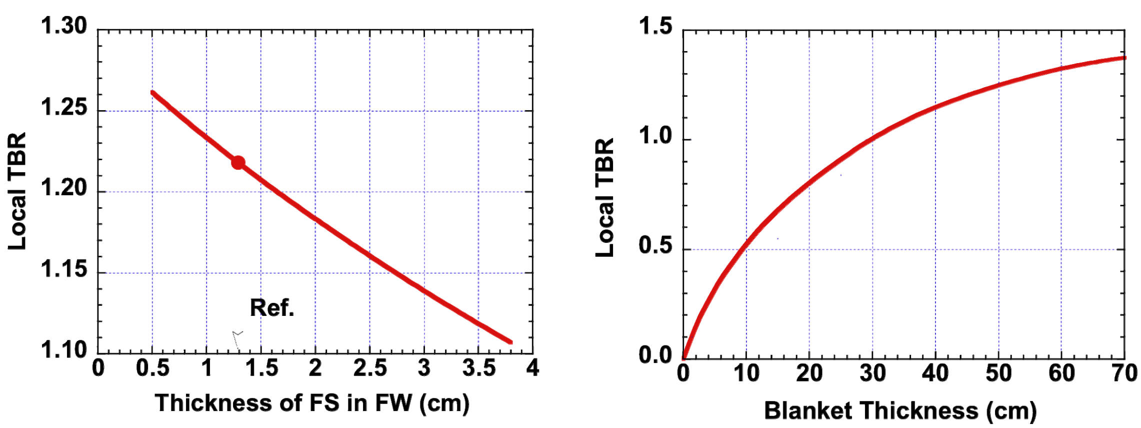

Fig. 7. Sensitivity of TBR to FW and blanket thickness of ARIES-CS [29].

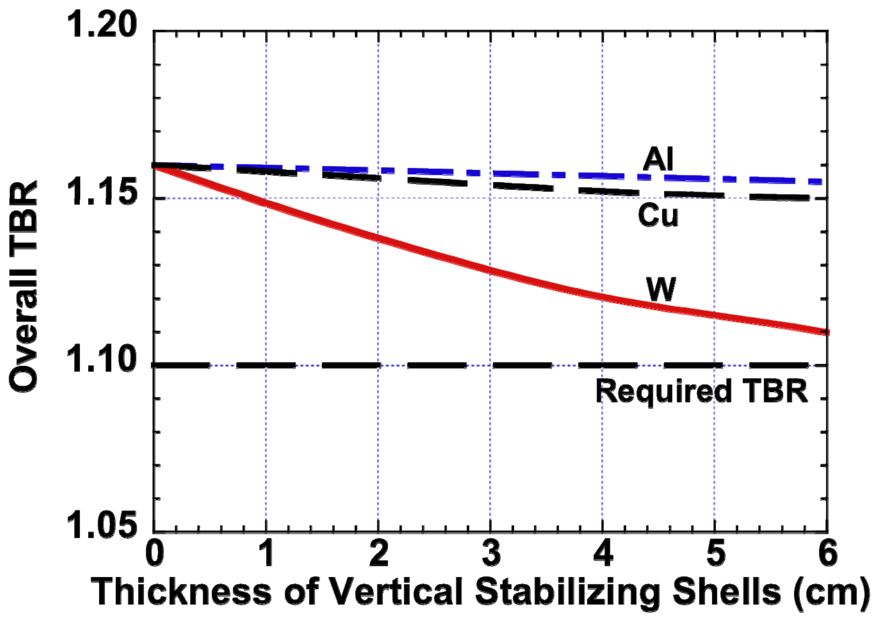

Fig. 8. Degradation of ARIES-AT TBR with thickness of stabilizing shell placed between OB blanket segments [31].

Several material and design choices could remarkably enhance MFE and IFE breeding levels. For instance:

- Li and LiPb offer highest breeding w/o neutron multiplier (Fig. 4)
- Beryllium is the best multiplier and enhances TBR significantly [1]
- FW structure degrades breeding more rapidly than blanket structure. Adding 1 cm FS to FW decreases TBR by 4-5% (Fig. 7)
- Li enrichment enhances breeding of LiPb, LiSn, and SB with multiplier (Fig. 4)
- Water coolant reduces TBR by up to 7% [17]
- In IFE, any breeder could breed without multiplier due to the absence of large penetrations [27]
- In MFE, Flibe, LiSn, and SBs need multiplier if blanket contains 10-20% structure. Blanket concepts employing free-fall breeders could meet breeding requirements without multiplier if structural content remains below 1-2%
- In tokamaks:
    - Unlike W, Al and Cu stabilizing shells degrade breeding slightly (Fig. 8) if placed between OB blanket segments [31]
    - Off-midplane penetrations result in less degradation to breeding
    - Single null configuration offers 3-5% more breeding than double null.

Lithium enrichment has the most impact on breeding, particularly for LiPb (Fig. 4). The calculated TBRs for ARIES-CS [29] and ARIES-AT [31] are shown in Figs. 9 and 10, respectively, as a function of the ^{6}Li enrichment of LiPb. The design requirement of 1.1 calculated TBR is indicated on the figures and suggests a ^{6}Li enrichment of 70% for ARIES-CS and 90% for ARIES-AT. ARIES-CS offers the possibility of raising or lowering the breeding as

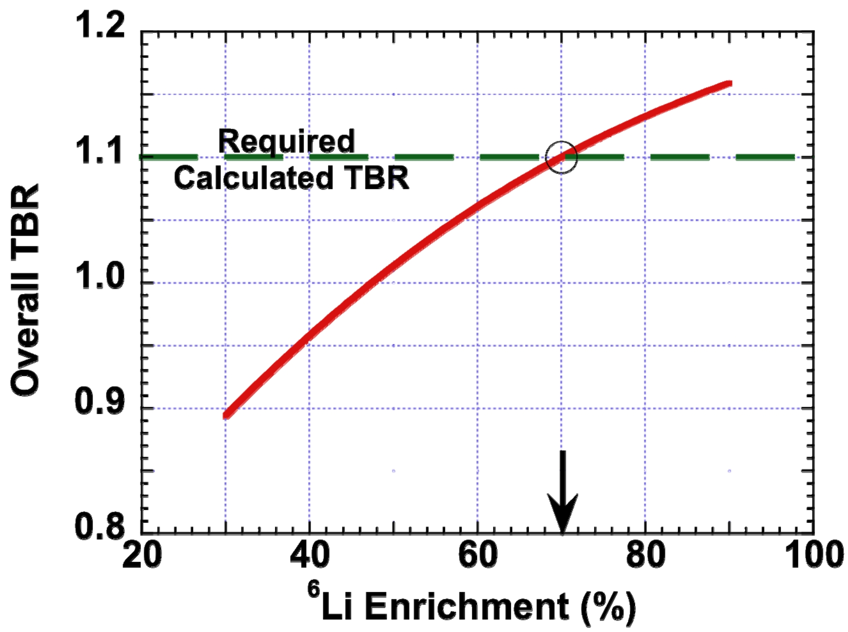

Fig. 9. Sensitivity of TBR to ^{6}Li enrichment for ARIES-CS LiPb/He/FS blanket [29].

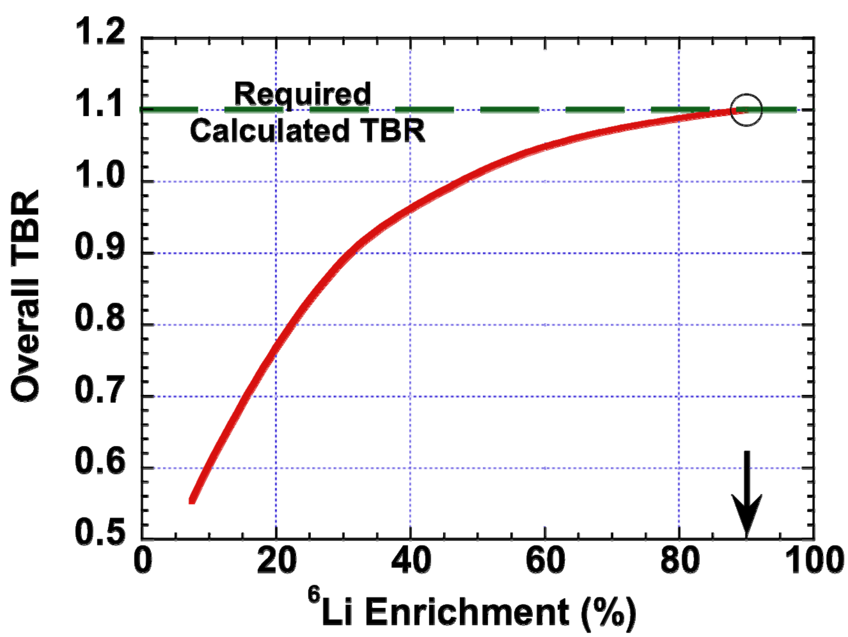

Fig. 10. Sensitivity of TBR to ^{6}Li enrichment for ARIES-AT LiPb/SiC blanket [31].

needed after plant operation using the ^{6}Li enrichment as a knob. On the other hand, ARIES-AT can only lower the breeding, offering no extra margin for higher breeding if deemed necessary. To avoid such a problem, future designs should operate with ^{6}Li enrichment below 90%, if feasible.

4. Online Adjustment of TBR via ^{6}Li Enrichment

4.1 Why do we need online adjustment of TBR with high accuracy?

In most discussions on the required TBR, emphasis is placed on maximizing the breeding rate of the blanket to assure T self-sufficiency. Very rarely discussed is the issue of too much breeding that could cause problems such as the storage of the excess T generated during the blanket lifetime and could become a safety issue, particularly in the absence of means to adjust the TBR during plant operation. In principle, such adjustment is feasible with liquid breeder blankets since the ^{6}Li concentration can be varied online over a wide range of interest. This is impossible for ceramic breeder blankets where an adjustment of the breeding rate would involve replacing some breeding by shielding modules or require some type of “trim rods” as used in fission reactors. To our knowledge, such a method has never been considered until now.

In order to illustrate if over-breeding can become a serious problem, the annual amount of T burned in a power plant has to be considered. With the values explained in more detail in Section 2, the following T balance can be made for both ARIES-CS and -AT power plants:

	ARIES-CS	**ARIES-AT**
T burn-up rate (kg/FPY)	135	97.8
Start-up inventory (kg)	4	2
T decay (kg/y)	0.33	0.16
Excess T breeding (kg/FPY):		
with Net TBR = 1.01	1.35	0.98
with Net TBR = 1.02	2.7	1.96
with Net TBR = 1.05	6.75	4.89

The surplus of T generated over a period of 5 years (after subtracting the desired start-up inventory for a new plant (with a doubling time of 5 years) and the amount of T decayed) would be significant, depending on the Net TBR (see Fig. 11). For comparison, the total amount of T accumulated from all CANDU reactors [2] at their peak, anticipated for 2025, amounts to ~30 kg [34]. This example shows clearly that the net T breeding averaged over the lifetime of the blanket has to be controlled with accuracy better than 1% to ensure T self-sufficiency without a storage problem for the surplus of T.

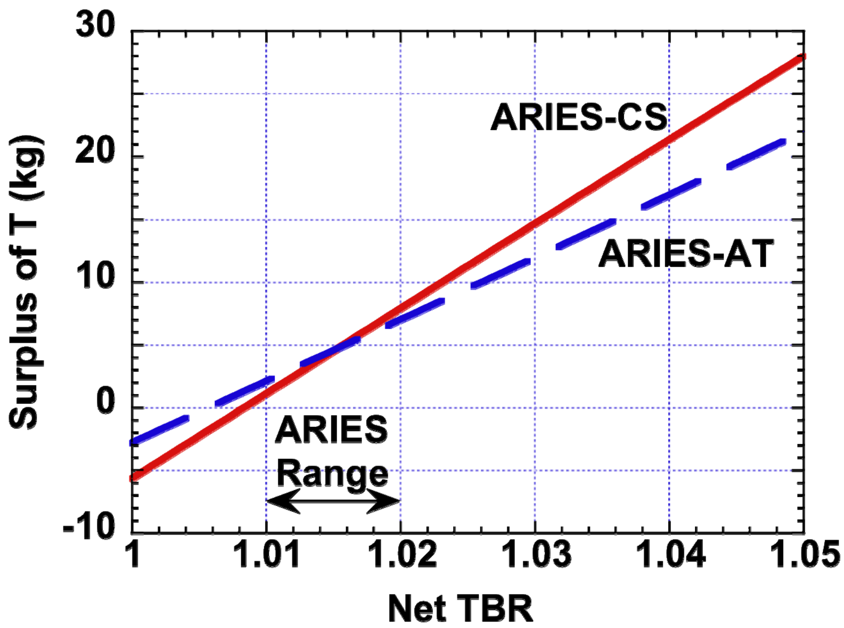

Fig. 11. Impact of Net TBR on T surplus.

Surplus of T (kg):	ARIES-CS	ARIES-AT
with Net TBR = 1.00	-5.65	-2.8
with Net TBR = 1.01	1.1	2.1
with Net TBR = 1.02	7.9	7.0
with Net TBR = 1.05	28	22

4.2 Possible methods for online adjustment of TBR of Pb-17Li blanket

In the past, most blanket and power plant studies utilized 90% enriched ^{6}Li in the eutectic lead lithium alloy to maximize the breeding. Only recently, in the ARIES-CS study [29], the design point has been moved to 70% enrichment in order to provide some extra margin to increase the TBR online, if needed. In ARIES-AT, the reference design point is 90% enriched ^{6}Li with a calculated TBR of 1.1 [31]. The sensitivity of the calculated TBR to the ^{6}Li enrichment is shown in Fig. 10 for the ARIES-AT LiPb/SiC blanket. To illustrate the online TBR adjustment scheme, we examined an extreme case of over-breeding where the 1.1 calculated TBR should be equal to the 1.01 Net TBR. In other words, the blanket delivers much more breeding during operation than actually needed for plasma fueling. To fix this over-breeding problem, the ^{6}Li enrichment of LiPb can be reduced to ~50% in order for ARIES-AT to provide a TBR of 1.01 (refer to Fig. 10). How can this be accomplished in practice? This is explained in the following simple example.

Assuming a total LiPb breeder volume in the entire plant of ~500 m^3, we have:

Total Pb-17Li mass ~5000 tons
Li content in $Li_{17}Pb_{83}$ with 90% enrichment (0.6 wt%) ~30 tons
Starting 90% ^{6}Li enrichment ⇒ ^{6}Li content in 90% enriched LiPb ~26.6 tons
Required 50% ^{6}Li enrichment ⇒ ^{6}Li content in 50% enriched LiPb ~14.7 tons
Required Li adjustment: 11.9 tons of ^{6}Li need to be replaced by ^{7}Li.

Two practical methods are feasible for TBR adjustment through combining two Pb-17Li eutectics with different enrichments:

a) Replace 2226 tons of the 90% ^{6}Li enriched Pb-17Li by 2226 tons of Pb-17Li with 100% ^{7}Li. For practical reasons, the amount of replaced Pb-17Li would be higher if ^{7}Li enrichment is < 100%.

b) Remove 13.5 tons of Li with 90% ^{6}Li enrichment from the eutectic alloy and replace it with 13.5 tons of ^{7}Li.

Which method is more suitable for the online adjustment of the ^{6}Li enrichment?

- Method-a is straightforward but requires an additional storage volume for the eutectic alloy (~200 m^3 in this example).
- Method-b does not require such large storage, but needs a practical method to remove about half of the Li with 90% ^{6}Li enrichment from the eutectic and feed it back with 100% ^{7}Li. The first step of separating the Li from the eutectic alloy is feasible by oxidizing the Li content in the alloy. The second step has already been investigated for the online replenishment of the ^{6}Li burn-up by adding small amounts of Li-rich Pb-Li alloy (LiPb, Li_3Pb, or Li_7Pb_2) to the eutectic alloy [35,36].

This example pertains to an extreme case where the reduction of the Net TBR is assumed to be from 1.1 to 1.01. By the time a Demo blanket is manufactured and built and with the support of a dedicated breeding-related R&D program, the margin between the calculated and Net TBRs will be < 9% (probably in the order of 2-3%). This would reduce the amount of Li to be exchanged in the above example by 3-4 fold.

5. Tritium Breeding-Related R&D Programs

A large gap exists between the near-term fusion experiments (that generate a few grams of T) and Demo or power plants (that produce ~100 kg of T/y). More R&D programs can close the gap and reduce the large uncertainties in the TBR prediction. Both analytical and lab-based studies are necessary to validate the analytical predictions of T production rates and to demonstrate T generation, recovery, storage, and fuel cycle that eventually lead to T self-sufficiency for fusion power plants. Active breeding-related R&D programs have been ongoing for decades in Europe and Japan. The U.S. had strong collaborative activities with Japan in the

1990s [14,15]. However, the U.S. neutronics program has been halted in the late 1990s for budgetary reasons.

Several small-scale T breeding-related experiments are currently operational worldwide and provide information. For example:

- The 14 MeV neutron integral experiments (such as the Fusion Neutron Source at JAEA in Japan [21], the TUD experiment in Germany [25], and the Frascati Neutron Generator at ENEA Frascati in Italy [26]) help validate the predictive capabilities (nuclear data and codes) used to calculate the Net TBR. Thus far, activities have been focused on solid/ceramic breeders. Ongoing experiment activities in Japan [22] and Europe [26] plan to generate the needed data for the LiPb breeder in the near future.
- In the U.S., T extraction and processing knowledge base can be tested at the Hydrogen Technology Research Laboratory at SRNL [37] or at the Safety and Tritium Applied Research (STAR) laboratory at INL [38].
- Testing blankets in ITER [20] will provide initial screening data of blanket concepts that have the potential for achieving T self-sufficiency. However, because of the limited fluence, pulse length, and testing space, measuring T production rate in the ITER Test Blanket Modules (generating ~2 g/y) will not be adequate to conclude whether T self-sufficiency can be achieved in Demo with a given blanket concept. However, testing blankets in ITER (where TBM have their own integrated systems for T breeding, T processing, and heat extraction) would help:
    - Clarify the technological conditions for attaining T self-sufficiency in future devices
    - Provide accurate T-related operational information (e.g., T inventory, T mean residence time, reliability, and efficiency of T processing system)
    - Determine the minimum required TBR for CTF, Demo, and power plants
    - Provide knowledge base for blanket design parameters (e.g., FW thickness, structure content, and coolant conditions) that impact TBR.
- Knowledge base for several plasma physics and technology-related conditions impacting T self-sufficiency will need to be addressed. These include:
    - Feasibility of plasma operation in a high plasma edge-recycling mode to increase T burn-up fraction to assist T self-sufficiency
    - Impact of Demo-relevant plasma fueling, heating, and diagnostics and current drive systems
    - Impact of nearby conducting shells for plasma control.
- Knowledge base to reduce uncertainties in predicting TBR to a few percent by:
    - Improving fusion nuclear data and data processing systems
    - Continue developing the CAD-MCNPX interface [11] for accurate 3-D neutronics modeling.
- A CTF providing large fluence, continuous (or long burn) operation, and full testing blanket sector would enable research in engineering design and reliability improvement on the most promising blanket concepts, resulting hopefully from the ITER screening process. Such research would aim to establish the knowledge base for one or more blanket concepts for Demo with assured basis for achieving T self-sufficiency. Also, a CTF would enable research under Demo-relevant conditions into the plasma physics and fusion technology properties required for T self-sufficiency. A CTF would further

provide critical data to validate the predictive capabilities (data processing systems and codes) needed for accurate determination of TBR for Demo.

Small-scale lab testing facilities, fission reactors, and code development will be very helpful to supplement ITER testing. Near-term R&D programs needed to fill the gaps and enable operation of CTF, Demo, and power plants involve efforts to:

- Improve TBR prediction:
  - With better evaluation, resolve deficiencies in nuclear data that significantly impact uncertainties in calculated TBR
  - Broaden the scope of nuclear data and code validation through integral experiments covering all relevant blanket concepts (e.g., LiPb/FS/SiC blanket)
  - Reduce uncertainty in calculated TBR attributed to approximations in modeling. The capability of using detailed engineering CAD drawings coupled directly to neutronics codes needs to be developed further.
- Improve the prediction of the minimum required TBR:
  - Continue developing fuel cycle dynamics model that accurately predicts the minimum required TBR based on T behavior, transport, and inventories in all subsystems (plasma facing components, blankets, plasma exhaust, and T processing)
  - R&D to accurately determine the T inventory holdup in all in-vessel components
  - R&D to increase the efficiency and improve the performance of T processing and extraction systems
  - Explore plasma operating scenarios with high plasma-edge recycling mode and high T fractional burn-up.
- Develop elements that help maximize TBR:
  - Thin SiC electrical/thermal insulator for LiPb/FS blanket concept
  - Low-neutron absorbing materials for:
    - Plasma fueling, heating, and current drive systems
    - Passive coils and shells that stabilize advanced plasma modes
    - Impurity control system (e.g., divertor).

6. Conclusions

For many researchers involved in fusion power plant development, the issue of T self-sufficiency is of particular concern because of the danger of placing the plant at risk due to T fuel shortage and because of potential problems handling the surplus of T. Thus, the calculated TBR must exceed unity by a margin. This design and breeder-dependent margin could be divided into four distinct categories: margin that accounts for known deficiencies in nuclear data (6-10% for LiPb system), margin that accounts for known deficiencies in modeling (3-7%), margin that accounts for unknown uncertainties in design elements (0-3%), and margin that accounts for T bred in excess of T consumed in plasma in order to compensate for T losses and the amount needed for future power plants (1-2% for advanced designs). The 10% lower end of the sum of all margins has been driving the minimum level necessary for reliable breeding in ARIES LiPb systems, meaning a calculated overall TBR of 1.1. The Net TBR during plant operation could be as low as 1.01, much lower than the 1.1 calculated TBR. Such a low Net TBR of 1.01 is practically achievable in fusion designs employing advanced physics and technology where the T fractional burn-up exceeds 10%, the T inventory is minimal, and the T extraction and reprocessing system are highly reliable.

To achieve a Net TBR of 1.01 (or lower) with sufficient precision, despite the rather large uncertainties in design elements, an online control of tritium breeding is mandatory for all fusion designs. This can easily be accomplished with liquid breeding materials by adjusting the ^{6}Li enrichment during plant operation. In reality, achieving a Net TBR around 1.01 during plant operation will not be verified without building and operating a component test facility or a Demo with fully integrated blanket and T recovery and reprocessing systems.

The difference between the calculated TBR and Net TBR will decrease with time. Future developments of highly accurate nuclear data evaluation, code validation using experiments with a 14 MeV neutron source, state-of-the-art CAD/MCNPX modeling tools, and detailed engineering designs will all help reduce this difference to the 2-3% range. The greatest single obstacle is the lack of a dedicated breeding-related R&D program in the U.S. Such a program is desperately needed not only to relax the breeding requirement, but also to bridge the gap between the near-term fusion experiments (that generate a few grams of T) and Demo or power plants (that produce 100-120 kg of T/y).

In summary, the following points can be made:

- The Net TBR in any fusion power plant must be very close to unity in order to ensure a sufficient tritium supply without generating an excessive surplus of T.
- The 3-D calculated TBR must be greater than the Net TBR.
- The breeding margin (calculated TBR - 1) is breeder and design-dependent. It primarily accounts for know deficiencies in the calculated TBR and 3-D modeling, unknown uncertainties in design elements, possible malfunctions during plant operation, and start-up T supply for a new power plant.
- A dedicated R&D program will significantly reduce the breeding margin. However, the remaining uncertainty could still be large enough for operating a Demo plant. Therefore, an online adjustment of the breeding rate is a must requirement for Demo and power plants.
- Such an online adjustment is feasible for liquid breeder blankets through adjusting the ^{6}Li enrichment of the breeder, but difficult to envision for solid breeder blankets.

Acknowledgments

This work was performed under the auspices of the U.S. Department of Energy (contract # DE-FG02-98ER 54462). The useful comments by the ARIES team members are gratefully acknowledged. Also, the authors are very appreciative to many colleagues for their advices, comments, and contributions: Drs/Prof. K. Schultz (GA), M. Sawan (UW), M. Abdou, M. Youssef (UCLA), S. Sato, T. Nishitani (JAEA), P. Batistony (ENEA), and U. Fischer (FZK).

References

1. L.A. El-Guebaly, Breeding potential of candidate breeders for the U.S. Demo plant, Proceedings of 16th IEEE Symposium on Fusion Engineering, Champaign, IL, Vol. 2, (Oct. 1995) 1198-1202.
2. CANDU reactors: http://www.candu.org/candu_reactors.html.
3. L.A. El-Guebaly, Overview of ARIES-RS neutronics and radiation shielding: key issues and main conclusions, Fusion Engineering and Design 38 (1997) 139-158.
4. The ARIES project: http://aries.ucsd.edu/ARIES/.
5. M.E. Sawan, C.S. Aplin, G. Sviatoslavsky, I.N. Sviatoslavsky, A.R. Raffray, Neutronics analysis of a self-cooled blanket for a laser fusion plant with magnetic diversion, Fusion Science and Technology 52 (4) (2007) 771-775.
6. M.E. Sawan, M.Z. Youssef, Three-dimensional neutronics assessment of dual coolant molten salt blankets with comparison to one-dimensional results, Fusion Engineering and Design 81 (2006) 505-511.
7. S. Reyes, R.C. Schmitt, J.F. Latkowski, J. Sanz, Liquid wall options for tritium-lean fast ignition inertial fusion energy power plants, Fusion Engineering and Design 63-64 (2002) 635-640.
8. D. Maisonnier et al. (ed), A conceptual study of commercial fusion power plants. Final Report of the European Fusion Power Plant Conceptual Study (PPCS), Report EFDA-RP-RE-5.0 (2005).
9. A. Li Puma , J.L. Berton, B. Brañas, L. Bühler, J. Doncel, U. Fischer et al., Breeding blanket design and systems integration for a helium-cooled lithium–lead fusion power plant, Fusion Engineering and Design 81 (2006) 469-476.
10. S. Sato, K. Ochiai, M. Wada, M. Yamauchi, H. Iida, T. Nishitani, C. Konno, Monte Carlo analyses of blanket neutronics experiments at FNS with latest nuclear data libraries, Proceedings of International Conference on Nuclear Data for Science and Technology, Nice, France, April 22 -27, 2007.
11. DAG-MCNPX User Manual. Available at: http://cgi.cae.wisc.edu/~fti/dagmc.cgi/wiki?p=UsersGuide.
12. M. Abdou, E. Vold, C. Gung, M. Youssef, K. Shin, Deuterium-tritium fuel self-sufficiency in fusion reactors, Fusion Technology 9 (1986) 250-285.
13. M.Z. Youssef, M.A. Abdou, Uncertainties in prediction of tritium breeding in candidate blanket designs due to present uncertainties in nuclear data base, Fusion Technology 9 (1986) 286-307.
14. M.Z. Youssef, A. Kumar, M. Abdou, The prediction capability for tritium production and other reaction rates in various systems configurations for a series of the USDOE/JAERI collaborative fusion blanket experiments, Fusion Engineering and Design 18 (1991) 407-418.
15. M. Abdou, H. Maekawa, Y. Oyama et al., Japan Atomic Energy Research Institute-United States integral neutronics experiments and analysis for tritium breeding nuclear heating and induced radioactivity, Fusion Technology 28 (1995) 5-38.

16. W. Kuan, M. Abdou, A new approach for assessing the required tritium breeding ratio and start-up inventory in fusion reactor, Fusion Technology 35 (1999) 309-353.

17. M.E. Sawan and M.A. Abdou, Physics and technology conditions for attaining tritium self-sufficiency for the DT fuel cycle, Fusion Engineering and Design 81 (2006) 1131-1144.

18. The FENDL-2.1 175 neutron 42-gamma group coupled cross-section library is available at: http://www.iaea.org/.

19. MCNPX Version 2.5.0 User's Manual, LA-CP-05-0369, Los Alamos National Laboratories, Los Alamos, New Mexico.

20. The ITER project: http://www.iter.org/.

21. The Fusion Neuron Source at JAEA, Japan: http://fnshp.tokai-sc.jaea.go.jp/english/index_e.html.

22. Mikio Enoeda, Masato Akiba, Satoru Tanaka, Akihiko Shimizu, Akira Hasegawa, Satoshi Konishi, Akihiko Kimura, Akira Kohyama, Akio Sagara, Takeo Muroga, Overview of design and R&D of test blankets in Japan, Fusion Engineering and Design 81 (2006) 415-424.

23. S. Sato, Y. Verzilov, K. Ochiai, M. Nakao, M. Wada, N. Kubota et al., Progress in the blanket neutronics experiments at JAERI/FNS, Fusion Engineering and Design 81 (2006) 1183-1193.

24. The Frascati Neutron Generator at ENEA, Italy: http://www.fusione.enea.it/LABS/FNG/detail_en.html.

25. The TUD (Technische Universität Dresden) neutron generator at Dresden, Germany: http://www.fzd.de/db/Cms?pNid=321.

26. P. Batistoni, U. Fischer, K. Ochiai, L. Petrizzi, K. Seidel, M. Youssef, Neutronics and nuclear data issues in ITER and their validation, Fusion Engineering and Design, to be published.

27. Laila El-Guebaly, Tritium breeding issues for MFE and IFE, Presented at ARIES Tritium Town Meeting (March 6-7, 2001). Available at: http://aries.ucsd.edu/LIB/MEETINGS/0103-ARIES-TTM/.

28. F. Najmabadi, A.R. Raffray, the ARIES-CS Team, The ARIES-CS compact stellarator fusion power plant, Fusion Science and Technology, to be published.

29. L. El-Guebaly, P. Wilson, D. Henderson, M. Sawan, G. Sviatoslavsky, T. Tautges et al., Designing ARIES-CS compact radial build and nuclear system: neutronics, shielding, and activation, Fusion Science and Technology, to be published.

30. F. Najmabadi, A. Abdou, L. Bromberg, T. Brown, V.C. Chan, M.C. Chu, F. Dahlgren, L. El-Guebaly et al., The ARIES-AT advanced tokamak, advanced technology fusion power plant, Fusion Engineering and Design 80 (2006) 3-23.

31. L.A. El-Guebaly, Nuclear performance assessment of ARIES-AT, Fusion Engineering and Design 80 (2006) 99-110.

32. L.A. El-Guebaly, ARIES-ST nuclear analysis and shield design, Fusion Engineering and Design 65 (2003) 263-284.

33. L. El-Guebaly, Potential coatings for Li/V System: nuclear performance and design issues, Fusion Engineering and Design 81 (2006) 1327-1331.

34. S. Willms, Tritium supply considerations, Presented at FESAC Fusion Development Paths Workshop (January 14, 2003). Available at: http://fire.pppl.gov/fesac_dp_ts_willms.pdf.

35. H. Feuerstein, D.A. Wirjantoro, L. Hoerner, S. Horn, Eutectic mixture Pb-17Li - in-situ production and Li-adjustment, Fusion Technology 2 (1994) 1257-1260.

36. P. Hubberstey, M.J. Capaldi, F. Barbier, Replenishment of lithium lost from Pb-17Li, Fusion Technology 2 (1996) 1475-1478.

37. The Savannah River National Laboratory: http://srnl.doe.gov/.

38. Safety and Tritium Applied Research at INL: http://nuclear.inl.gov/fusionsafety/experiments/star.shtml.