Study on Tritium Production Property

by D-T and D-D Neutrons of LiPb Blanket for Fusion Reactor

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Contents

1. Introduction

1. 1. Energy for Humankind

Energy is conserved. Energy is not being created and destroyed completely. Energy exists in different form everywhere from around us to at the outside of the universe. All activity of humankind need the energy whether the activity for productive or consumptive purposes. Let's think about the energy for comfort and convenience of living for us. Now, we can find and see easily the energy around me in the student room. The monitor displays what we input some words for this paragraph. The computer works what we ordered. The printer prints out what we wanted to read some articles. There are sockets on the wall and fluorescent lamps on the ceiling. All I mentioned things consumed electrical energy. We who are living in the modern society are consuming a lot of energy for keeping industrial activities such as operating factories and providing a service for the public. The Industrial Revolution was a period from the 18^{th} to the 19^{th} century where major changes in agriculture, manufacturing, transport, mining and technology starting in the United Kingdom, then subsequently spreading throughout Europe, North America and the world finally. After the Industrial Revolution, consumption of energy is increasing tremendously for more comfortable and convenient society to humankind with world population growth and development of high technologies. Although the population growth in more developed region is not high, but its reduction rate is decreased due to the development in medical technology. In addition, the population in less and least developed region has increased greatly with low reduction rate and relatively high growth rate. Recent world population growth from 1950 to 2100 [ref 1] is shown in figure 1-1. In order to verify the connection between the population and energy, "world energy demand" from 1800 to 2000 [ref 2] is shown in figure 1-2. Figure 1-1 shows estimated world population from 1950 to 2010 and being expected world population with the medium variant to reach 10.9 billion persons by 2100. That is, 3.7 billion more than in 2013 and 1.3 billion more than in 2050. As figure 1-2 indicates, world energy demand had risen a hundredfold from 1800 to 2000. World population growth accompanies increasing of energy demand.

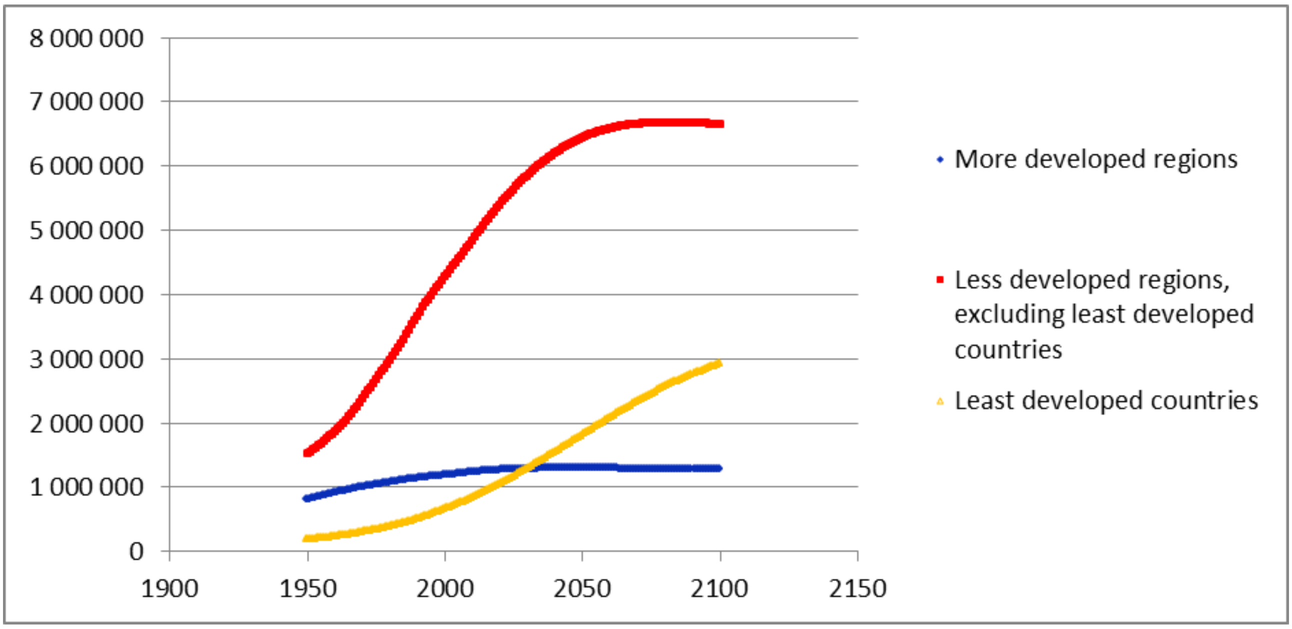

Fig. 1-1. World population growth from 1950 to 2100 (Medium variant)

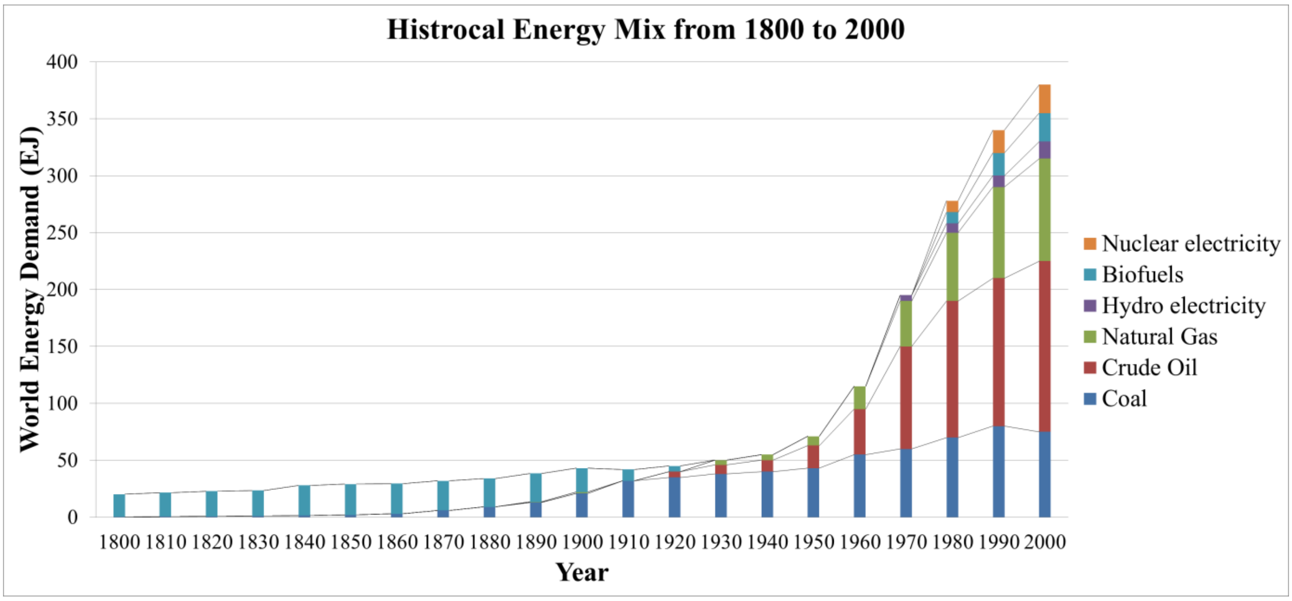

Fig. 1-2. World energy demand from 1800 to 2003

Regionally, more developed and least regions such as European Union, United States and other OECD countries are in high demand in primary energy demand until now and it will keep continuing. However, primary energy demand in less developed countries such as Middle East, India, China and other non-OECD countries will increase gradually more with their economic development in the near future as figure 1-3 represents [ref 3].

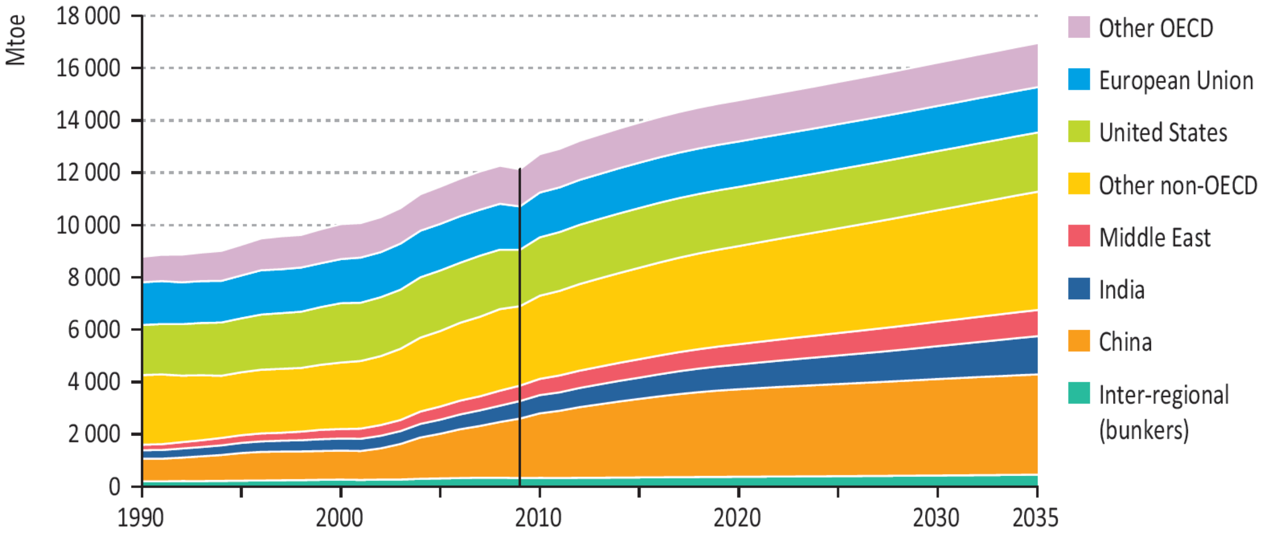

Fig. 1-3. World primary energy demand by region in the new policies scenario

1. 2. Supply Problem of Current Energy Sources; Fossil fuel, Hydro and Nuclear Fission

There are several types of energy we consume since 1980 [ref 3] as shown in figure 1-4. Fossil fuel energy such as oil, coal and gas accounts for about three quarters. Fossil fuels are formed from the remains of dead organisms such as plants and animals. Typically they are includes three types of their state as follows: coal as a solid type, petroleum (oil) as a liquid type and natural gas as a gas type. After that biomass and waste occupy 10% of world primary energy. The biomass that humans have harnessed since the time when people began burning wood to make fire is an eligible source of renewable energy in most of more developed regions nowadays. Development of related science and technologies and expecting forward-looking sources of energy are firm criteria for nuclear energy since the early 20 century. Even though other renewable energies including solar, wind, geothermal and hydro still have a small amount of total energy demand, people can decrease dependence on burning fossil fuel with expansion act of renewable energy.

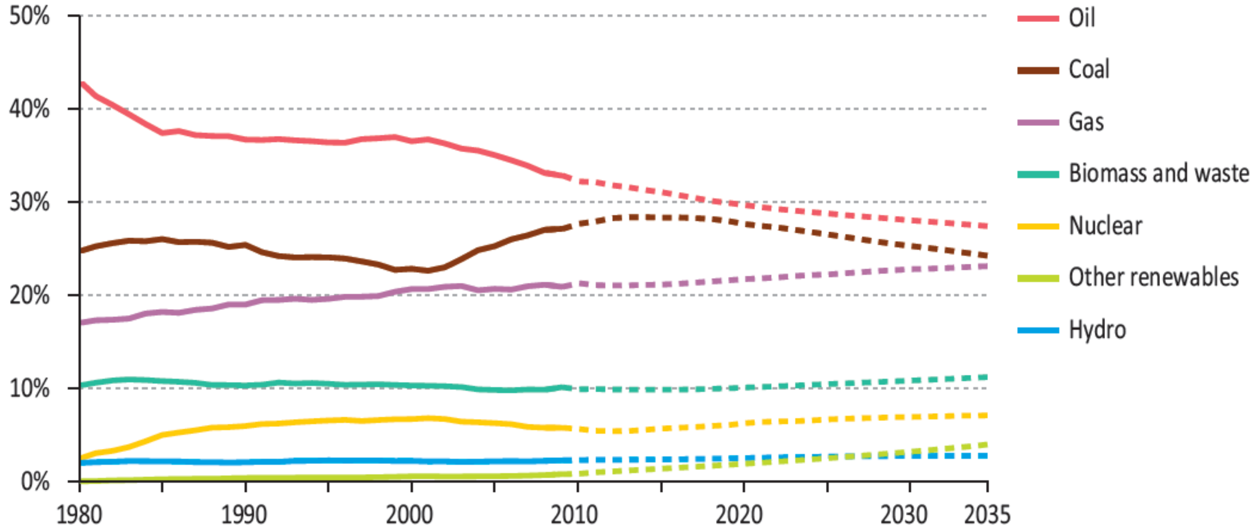

Fig. 1-4. Shares of world primary energy demand in the new policies scenario

As a simple overview of the current world energy situation consider the end uses of energy. Energy demand in the power sector, in particular electricity generation, is the easiest way for understanding relation between energy and human act above all else. Figure 1-5 represents electricity generation by fuel in 1990 and 2035 [ref 4]. As we can see, there are a lot of changes to generate electricity by new type of fuels such as solar, geothermal and wind power, and expansion uses of renewable energies instead of uses by coal and oils in 2035. Nevertheless, fossil fuel to burn and generate electricity still is apportioned more than a half of total. Unlike world primary energy demand, hydro power and nuclear power are occupied most of the remains in both the past and the future.

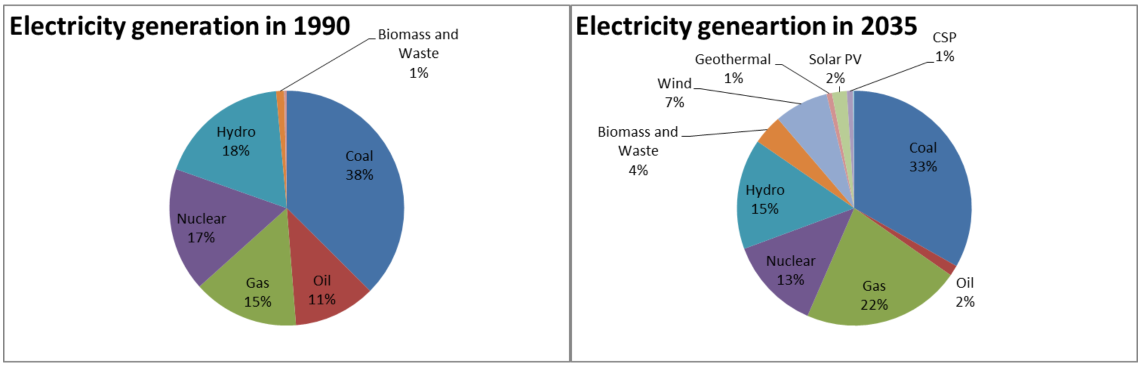

Fig. 1-5. Electricity generation from each energy source in 1990 and in 2035

Hydro-electric power (water power) is derived from the energy of falling water or running water. It offers a number of advantages. Primarily, it is a clean and renewable energy source and produced no harmful air pollutants, few greenhouse gases emission except emission by related materials for its construction. Also its plants have low operational costs and a long life span after installing. Hydropower is a unique source of electricity because water can be stored, then drawn to provide power as needed. It can be used for both base load and peak demand. Consequently, hydro contributes to a stable and reliable electricity grid. Among hydropower's disadvantages are its high initial capital costs, and the possibility that drought can limit its capability to provide power. It takes a great deal of money and resources to build large hydro plants and dams. Like all energy projects, hydropower affects the environment. For example, hydropower reservoirs modify natural habitats. Planned measures such as reforesting land, establishing wetlands, and restocking fish stocks can minimize these impacts.

Nuclear power is derived from the energy of nuclear fission reaction of heavy elements, notably uranium and plutonium. Fissile materials are natural U-235 and the man-made isotopes Pu-239 and U-233. The total energy from fission reaction of U-235, after all of the particles from decay have been released, is about 200 MeV [ref 5]. However, only about 190 MeV of the fission energy is effectively available, because there are countless part of energy yield of the fission process such as other emissions of gamma rays and beta particle. Nevertheless the loss, only 1.3 g of U-235 is used per megawatt-day of useful thermal energy released. To produce the same energy by the use of fossil fuels, millions of times as much weight would be required. In the decade of the 1950s several nuclear fission reactor concepts were tested and dropped. Then, nuclear power has generated useful heat and electricity since 1970s. Before the Fukushima Daiichi accident on 11th March in 2011, nuclear power provided about 5.7% of the world's energy and 13% of the world's electricity. According to recent report by the IAEA (International Atomic Energy Agency), there are 437 nuclear power reactors in 31 countries although certain reactors are not generating electricity [ref 3]. Assurance of nuclear energy, in particular the prevention of radiation hazard, is more strongly required after the Fukushima Daiichi accident. Recent serious nuclear power plant accidents include the Fukushima Daiichi nuclear disaster (2011), Chernobyl disaster (1986) and Three Mile Island accident (1979). The Three Mile Island accident was a significant turning point global development of nuclear power. After the event, the number of reactors under construction in the U.S. declined every year and a lot of similar reactors on order were canceled. Totally 51 nuclear reactors in the US were canceled from 1980 to 1984. Changing social attitudes towards hazard and risk of nuclear energy was accompanied by the Chernobyl disaster. According to the report by IEA (International Energy Agency), "low nuclear case" in world energy scenario the situation is to be especially considered after the Fukushima Daiichi nuclear disaster [ref 3]. Global nuclear market is downsizing, in practice, showing signs of stagnation or even decline to keep operation of current plants and construct newly. Energy consumers are concerned about high level radioactive materials from nuclear power plants and have antipathy towards the risky energy source.

Current energy sources to be mentioned above have many assignments accumulated as follows: some delivery troubles by inequitable distribution of resources, the possible monopoly by specific countries, harmful effects on global environment or some technologic problems. In this situation, discovering and securing of other sustainable energy sources can be regarded as an obligatory energy promise for posterity.

1.3. Considerable New Energy Source; Possibility and Risks of Nuclear Fusion Energy

All of the activities of human beings depend on energy. Also, an adequate long-term supply of energy is therefore essential for man's survival. There are difficult energy problems facing the world that will probably become worse in the future. Obviously there is no single solution. Therefore, people who are living in the present should secure selectable energy sources. As a solution, however, many researchers make every effort to be possible options about alternative energy sources, especially non-fossil fuels such as renewable energy sources for the future humankind. Moreover, rapid increase in energy consumption in the developing world will be the key driver of growth for the global energy market.

Fusion energy is one of the alternative energy sources with eligibility criteria such as its resources, supply chain stability, safety, environmental feasibility, social receptivity of processing and disposal of radioactive waste. When two light nuclear particles combine or fuse together, energy is released because the product nuclei have less mass than the original particles. Nuclear fusion reaction takes place in the sun and in other stars through the so-called carbon cycle, a complicated chain of events involving hydrogen and isotopes of the elements carbon, oxygen, and nitrogen. The cycle is extremely slow, however, and is not suitable for terrestrial application. The most promising reactions on the earth make use of the isotope deuterium (abbreviated D). It is present in hydrogen as in water with abundance only 0.015%, however since earth has enormous amounts of water, the fuel available is almost inexhaustible. The relevant reactions are as follows:

$$D + D \rightarrow n + {}^{3}He + 3.27\ MeV \qquad (1\text{-}1)$$

$$D + D \rightarrow p + T + 4.03\ MeV \qquad (1\text{-}2)$$

$$D + {}^{3}He \rightarrow p + \alpha + 18.3\ MeV \qquad (2)$$

$$D + T \rightarrow \alpha + n + 17.6\ MeV \qquad (3)$$

The fusion of two deuterons (reaction formula 1-1 and 1-2) results in two processes of nearly equal likelihood. The other reactions (reaction formula 2 and 3) yield more energy but involve the artificial isotopes tritium, T, and the rare isotope helium-3, ${}^{3}He$. Also the reaction rate as a function or particle energy is lower for the D-D case than for the D-T case. The cross section of D-T reaction is large and the energy yield is favorable. The ideal ignition temperature for the D-T reaction in only 4.4 keV in contrast with 48 keV for the D-D reaction, making the achievement of practical fusion with the former far easier, even though it is hard to occur nuclear fission reaction of uranium-235. The world's fusion energy research is focused on the D-T reaction, the ease of initiation being of more importance than the problem of breeding tritium. As expected, there is one drawback that is the artificial isotope tritium is required. Tritium can be produced by neutron absorption in lithium. The two reactions of tritium production as follows:

$$n + {}^{6}Li \rightarrow \alpha + T + 4.8\ \mathrm{MeV} \qquad (4)$$

$$n + {}^{7}Li \rightarrow T + \alpha + n - 2.5\ \mathrm{MeV} \qquad (5)$$

The neutron can come from the D-T fusion process itself, in a breeding cycle similar to that in fission reactors. Lithium or lithium-based materials can be used as a breeding blanket. In the long run, use of the D-T reaction is limited by the availability of lithium. All things considered, the lithium for tritium generation and the tritium for initial operation mode are the real key substances to achieve a practical D-T fusion reactor, according to related important tasks; (a) a qualified breeding blanket design using adequate lithium without the dissipation of the lithium for tritium breeding performance, (b) a securing method of required tritium for initial loading of D-T reaction.

Fusion energy has no carbon emissions and uses the isotopes of hydrogen for fuel. It has low possibility of regional conflict with an energy resource. Also, it has excellent inherent safety qualities, among which absence of chain reaction, no production of long-lived and high-level radioactive waste, even though activated plant materials still need to be disposed. Then the worst accident would not be able to occur like the fission power plants. The inherent safety features are due to very low fuel inventory in the reactor during operation and the rapid cooling that extinguishes a dysfunction of the fusion reactions. As yet, a practical fusion device has not been developed, and considerable research and development such as plasma stability and its control, plasma confinement and transport, super-conducting coil, recovery/purification/handling of tritium, activated material by neutrons and a divertor and breeding blanket design will be required to reach that goal. In order to employ the fusion as the practical energy resource, the scientific and technical aspects should be solved as the primary aim. Then its qualifications such as economic feasibility, the effect on reduction of carbon dioxide and the risk of its wastes are need to be examined and assessed sufficiently.

1. 4. Limitation of Fusion Reactor on Fuel Supply and Reactor Design

There are several limitations of fusion reactor about its fuel supply and reactor design. Uranium enrichment technology is most important factor to operate nuclear fission reactor possibly or to pay for import enriched uranium. These can be great limitations of fission fuel supply. In the case of fusion reactor, however, there are more complicated restrictions such as the corroboration of self-sufficiency tritium with breeding tritium capability of the blanket, lithium-6 enrichment for TBR>1 and securement of initial loading tritium before the operation of DEMO reactor beyond fusion experimental reactor. The main purpose of this study is to solve the restrictions of fuel supply for fusion reactor by design LiPb blanket with natural lithium, validation of D-D start-up scenario for production of initial tritium with designed LiPb blanket and proposal of D-D neutrons integral experiment on laboratory scale in order to evaluate the scenario. Figure 6 represents fusion blanket research for DEMO and commercial reactor and significance, positioning of this study. As figure 1-6 indicates, neutronics and integral experiment are needed to carry out at the same time or with similar processing period to design and operate of DEMO reactor. In this study, there are three verification methods to clarify the fusion fuel limitations in order to achieve DEMO reactor as follows: neutronics analyses including LiPb blanket design for conviction of its tritium self-sufficiency with natural lithium breeder (chapter 2), numerical analyses of fuel supplement scenario to secure initial loading tritium with neutronics results of blanket performance including tritium bred (chapter 3), and an empirical study including small scale D-D neutron integral experiment design for the supplying scenario (chapter 4).

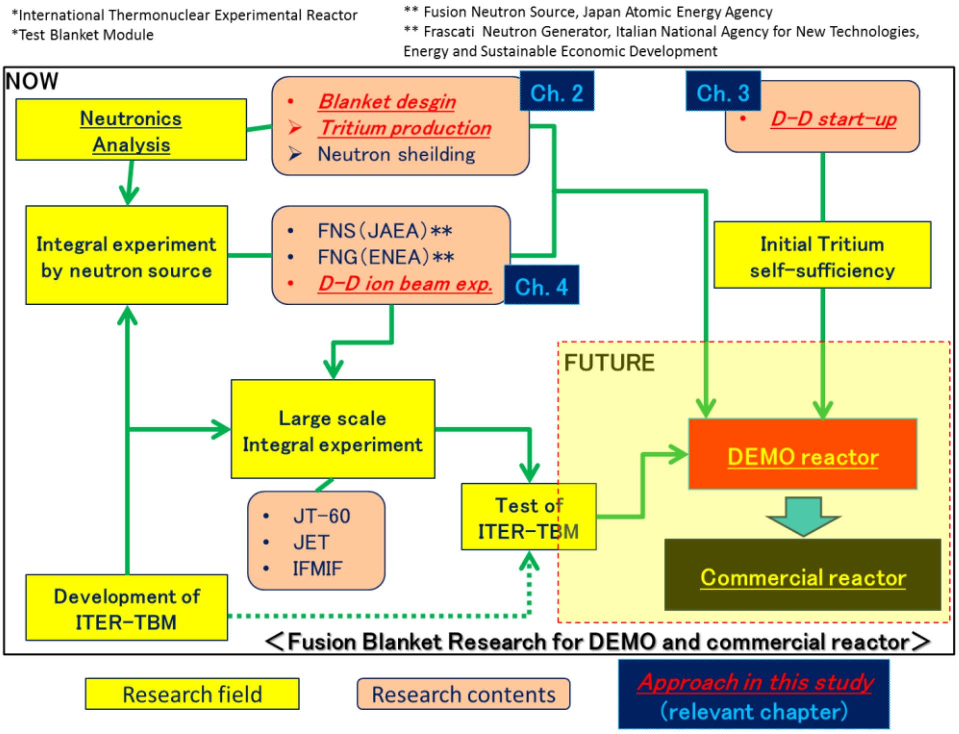

Fig. 1-6. Fusion blanket research and approach methods in this study for DEMO reactor

1. 5. Importance of Fusion Neutronics Research

The fusion blanket research is related neutron behavior deeply since the produced neutrons by D-T reaction carry away 80% of its total energy, is deposited to the blanket. Since it is merely computer analysis, needs to combine with integral experiment for raising reliability of the results. Neutronics of fusion blanket modules is necessary to compare results from the calculations with results from integral experiments. Many simulation results for test blanket module (TBM) of ITER, or related fusion DEMO reactors have been proposed [ref 6-11]. There are fusion development research needs to benchmark the actual scale of the fusion blankets. At the same time, investigations of the advanced concepts are performed at school or laboratory level that also needs to verify the neutronics calculations with small scale experiments.

However, there is still no zero power reactor before success of ITER, it can be performed like a critical assembly in fission research. At present, there are two operating small scale nuclear fusion systems for research purposes, FNG (Frascati Neutron Generator) in Italy [ref 12-14] and FNS (Fusion Neutron Source) in Japan [ref 15-17]. Even though the systems are being performed actively for the current fusion research, it cannot be satisfied total demand for the blanket research. The operation periods of the systems have passed 20 and 30 years respectively, and there is no confident plan to construct new D-T neutron source system yet. The development D-D neutron source on laboratory scale mentioned in the previous section can be had great merit in these circumstances.

1. 6. Significance of Breeding Blanket for Fusion Hybrid Reactor Concept, GNOME

Biomass hybrid fusion reactor (GNOME reactor) that can produce electricity and chemical bio-fuel together was designed for small scale fusion reactor on the sidelines of the DEMO. The concept design of GNOME reactor is shown figure 1-7. Its overall design, pumping power and neutronics were proposed and examined already by K. Ibano et al. [ref 18, 19]. The breeding blanket for GNOME was examined on SiC-LiPb concept and its few performances only. However, it should be assessed with more various conditions due to its importance as a practical fusion reactor.

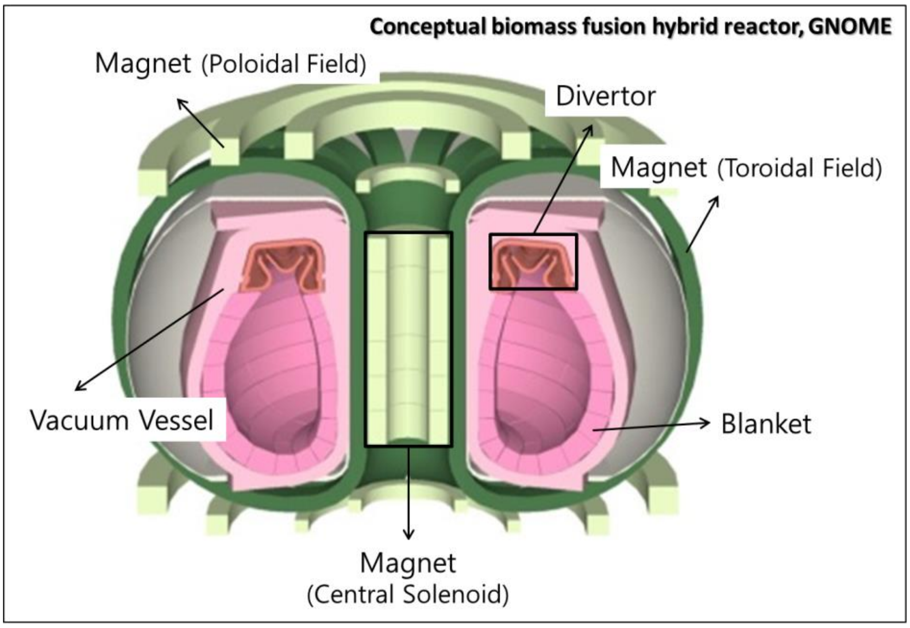

Fig. 1-7. Biomass fusion hybrid reactor, GNOME

The blanket is an important part of the fusion reactor because it extracts the heat for generating power, it shields the magnetic coils from neutrons by nuclear reactions in the plasma and it produces tritium as fuel for D-T reaction. It can be called a multiplayer in fusion reactor. It goes without saying that the tritium breeding performance of the blanket should be managed and verified a lot of related conditions such as; a thickness and material of a first wall (FW), a cooling material, a size of cooling pipe, a type of breeding material, inner structure of the blanket and so on. In other words, the blanket has to be designed and manufactured with optimization of the conditions mentioned above for fulfilling its own roles.

It is particularized that tritium breeding performance of the fundamental SiC-LiPb blanket design, its nuclear heating, effect on TBR by nuclear reflector in the blanket and numerical analysis about expectable tritium amounts of designed blanket module including the optimized conditions. Also, advanced blanket design is included that the betterment of SiC-LiPb blanket module for attaining high TBR (Tritium Breeding Ratio) more, effects of blanket constituents on tritium production/breeding performance in the LiPb blanket with natural lithium/beryllium in chapter 2.

1. 7. Tritium flow and Fuel Supply Scenario for Fusion Power Plant

Although there is the blanket with tritium breeding performance examined in the fusion reactor, the concerns about tritium as a fusion fuel has not completely disappeared. As we mentioned section 1.3, tritium for the fusion reactor needs to be produced artificially in the breeding blanket. Because tritium is naturally scarce and is a radioactive isotope of hydrogen that it has a half-life of 12.3 years and emits beta particle. It is known that tritium inventory being around 20 kg in the oceans and atmosphere. Also non-fusion facilities can produce tritium annually such as CANDU (CANada Deuterium Uranium) fission reactors in around the world. It has been recovering tritium at a rate of ~1.7 kg/year. However, most existing tritium will be burned for ITER operation. Moreover, the annual tritium consumption of fusion power plant operating will be 55.6 kg per full power year per GW. The fusion power plants have to generate their own tritium needed for plasma operation. Once D-T plasma operation is achieved, tritium is produced through interaction between neutrons from D-T reaction and lithium in the blanket. The tritium is fueled to next plasma operation continuously after tritium recovery from the blanket, separation of tritium into molecular hydrogen and tritium purification. It seems that the fusion power plants have an ideal closed tritium fuel cycle. However, there is still remaining tritium fuel problem to solve for initial loading tritium at the beginning of D-T operation.

As one of the solution, this research supports the fuel-operation scenario called **'D-D start-up scenario'** [ref 20, 21]. In order to secure tritium for full D-T operation, literally D-D reactions are used in the scenario: (a) reaction between 2.5 MeV neutrons by D-D reaction and lithium in the blanket (reaction formula 1-1 of section 1.3), (b) tritium by D-D reaction (reaction formula 1-2 of section 1.3), with the mounted blanket that is examined previously tritium breeding performance on the D-T phase. The stored tritium is deployed as the fuel on the initial D-T operation stage. The primary parameters of this scenario are tritium breeding performance by D-T neutron of the blanket and tritium production by D-D neutron together. Also, various operation conditions such as operation starting with small amount of tritium and without initial tritium should be evaluated.

It is discussed that tritium production/breeding performance of improved SiC-LiPb blanket design, analysis of reconstituted the D-D start-up by system dynamics code and verification of time scale for full D-T operation in chapter 3.

1. 8. Requirement of D-D neutronics and its integral experiment

The majority of fusion power plant studied and related experiments have employed the D-T fuel cycle due to it is the easiest way to achieve ignition on the earth. Securing tritium fuel can be determined by breeding in the blanket well known or by shedding new light on the fuel scenario study without tritium consumption using D-D reaction.

Some approach of D-D neutronics is necessary work for verifying the latter as one of the purpose. Also, a fusion reactor is used under intense neutron and other radiations, in particular high-energy neutrons as the D-T reaction products. Lots of verifications about interaction between neutrons and materials should be dealt with imposing on materials and measuring instruments like a neutron detector for the reactor that can suffer damages from the process before the stage of fusion commercialization. Allowable range between simulation and reality can be sought through the D-D neutronics approach including computer analysis and experiments with D-D neutron sources in the world. Even though D-

T neutronics and D-T integral experiments are certain way to evaluate directly for D-T fusion, D-D neutronics and its experiments are more acceptable way for preventing material damage in experimental stage because D-D neutrons have lower energy than D-T. Likewise, carrying out the experiment with same conditions for several times cannot be avoided in order to improve the reliability of the results in any field of science. It is needed to apply an adaptable method with few risks since measurement errors and devices damages can be occurred. In addition, there are various modes in ITER operation including D-D operation. The first plasma is expected to be produced in 2020 and D-T operation is predicted to be started in 2027. As already known, the interval between the first hydrogen plasma and D-T operation has D-D operation with low tritium concentration. Several diagnoses are possible before full-scale operation through the D-D operation mode for discovering symptom unexpected. It can save many steps of unnecessary operation and prevent severe accidents or superfluous materials activation when some troubles are detected.

In order to evaluate the validity of the D-D start-up scenario, chapter 4 describes in detail as follows: the adequacy of D-D neutron integral experiment using a deuterium ion beam to examine the tritium performances experimentally.

1. 9. Lithium for fusion breeding material

Lithium is demanded a lot in myriad of processes including ceramics, glass, metal refining, organic synthesis and batteries, although the tritium breeder for a fusion reactor has been excluded from the uses. These demands are almost fulfilled from buried lithium in the mine. Lithium is deposited three main types: brines, pegmatites, and sedimentary rocks. One of the estimated total lithium resources in the world is at least 38.33 million tons. Also, seawater contains about 230 billion tons of lithium, although the concentration is low (0.1 ~ 0.2 ppm). Selective recovery methods of lithium ion from seawater were developed and applied already. Existing lithium resources can be deemed to be sufficient to secure at this time. As mentioned above, however, lithium can become a trouble in global resources market due to the competition for scarce resources and the price rise by its overfull uses. Besides, it is not only problem about lithium for the fusion reactor in particular. As mentioned in section 7, tritium consumption of fusion power plant operating will be 55.6 kg per full power year per GW. Natural lithium has large mass per mole about 2.3 (6.941/3.016) times than tritium. Therefore, natural lithium to produce tritium per a year will be required about 128 kg. Natural lithium is composed of two stable isotopes, lithium-6 (7.6%) and lithium-7 (92.4%). According to (4), (5) in section 1.3, even though both of the isotopes can be produced tritium, enriched lithium-6 is used as the main available tritium breeders. Because lithium-6 has a large cross section for reacting with thermal neutrons, expelling a tritium. Figure 7 shows cross section of both of lithium isotopes for reacting to produce tritium, (n, t) [ref 22]. As shown in figure 1-8, lithium-7 has a large cross section of order 0.3 b with very high energy neutron with above 5 MeV. However, there is no possibility to occur the (n, t) reaction below the 2.5 MeV, its threshold energy. Lithium-6, on the other hand, has a large (n, t) cross section with thermal neutrons. Since lithium-6 is a scarce resource and an adequate tritium source, it is enriched in most cases. The lithium enrichment is an additional step at an extra expense for the fusion reactor. The use of natural lithium, hence, can be a worthwhile research in tritium breeder field. Thus, tritium breeder including natural lithium is proposed as a kind of trial blanket design and as the improved blanket design with details of neutron multiplier region in chapter 2. It is expected that the use of natural lithium with beryllium alleviate concerns about lithium resources for a prospective fusion blanket concept through this approach.

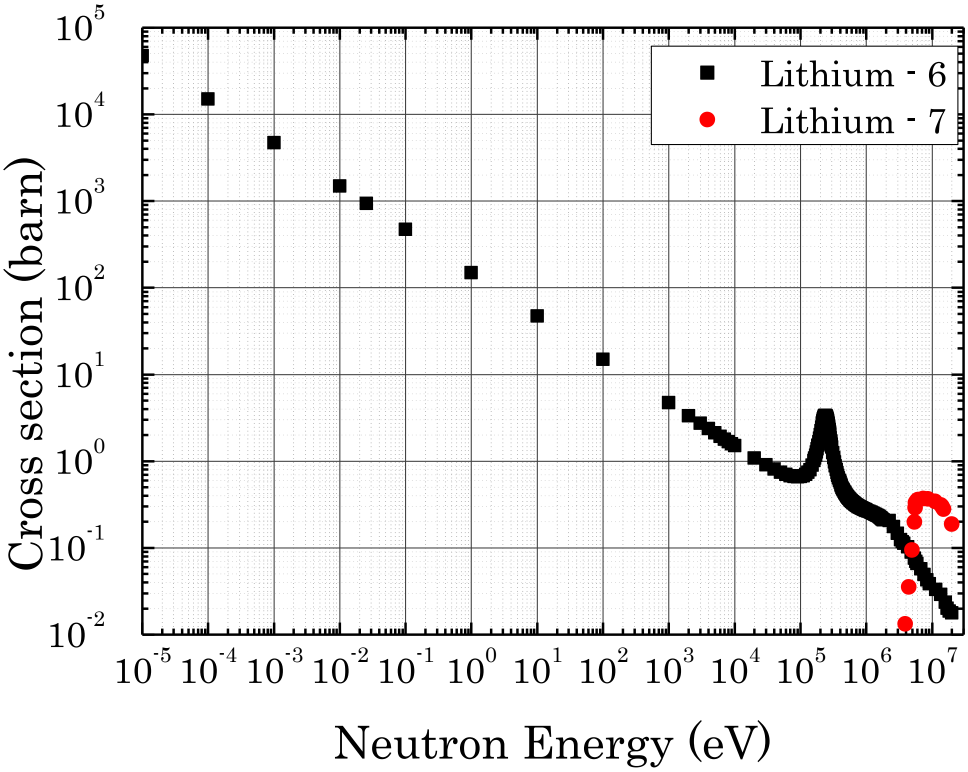

Fig. 1-8. Tritium production cross section, (n, t) of natural lithium

1. 10. Chapter Summary and Goal

Various designs for the fusion reactors have been proposed for the way to achieve commercialization of fusion energy. Typically selected ideas and concepts were taken into consideration for the international project, ITER, as a substantive fusion issue. However, there are still several problems on fusion fuel supply such as a fuel breeding blanket design and initial loading of tritium in particular. In this study, neutronics analysis and system dynamics were carried out for the approach methods to assess the fusion fuel supply in each chapter as follows: Chapter 2 presents details of fundamental LiPb blanket designs with high tritium production/breeding performance and the effects by structural parameters, Chapter 3 presents verification of the D-D start-up scenario employing the LiPb blanket with advanced tritium production/breeding performance, Chapter 4 presents evaluation as D-D integral experimental method for the D-D start-up scenario by the proposal using a deuterium ion beam equipment, Chapter 5 presents the overall conclusion of this study. From the results and discussion, the study on the blanket design and tritium cycle can be expected that it contributes to attain towards sustainable society on fusion energy system without its fuel concerns.

References

[1] World Population Prospects: The 2012 Revision, Highlights and Advanced Tables, United Nations Department of Economic and Social Affairs/Population Division (2013)

[2] Energy Vision 2013, Energy Transitions: Past and Future, World Economic Forum (2013)

[3] World Energy Outlook, International Energy Agency (2011)

[4] World Energy Statistics, International Energy Agency (2012)

[5] Raymond L. Murray, Nuclear Energy, Fifth Edition (2001)

[6] L. GIANCARLI et al., "Test blanket modules in ITER: An overview on proposed designs and required DEMO-relevant materials," *J. Nucl. Mater.*, 367-370, 1271 (2007)

[7] S. ZHENG et al., "Neutronics analysis for the test blanket modules proposed for EAST and ITER," *Nucl. Fusion*, 47, 1053 (2007)

[8] Y. POITEVIN et al., "The European breeding blankets development and the test strategy in ITER," *Fusion Eng. Des.*, 75-79, 741 (2005)

[9] K. TOBITA et al., "Search for reality of solid breeder blanket for DEMO," *Fusion Eng. Des.*, 85, 1342 (2010)

[10] Z. XU et al., "Desgin and validation of breeder unit of helium-cooled pebble bed blanket for demo fusion reactor," *Fusion Eng. Des*, 81, 2233 (2006)

[11] Y. KIM et al., "A neutronic investigation of He-cooled liquid Li-breeder blankets for fusion power reactor," *Fusion Eng. Des*, 75-79, 1067 (2005)

[12] M. Martone et al., The 14 MeV frascati neutron generator, *J. Nucl. Mater.* 212, 1661 (1994)

[13] P. Batistoni et al., Neutronics experiment on a mock-up of the ITER shielding blanket at the Frascati Neutron Generator, *Fusion Eng. Des.*,47, 25 (1999)

[14] Yu A. Kaschuck et al., Fast neutron spectrometry with organic scintillators applied to magnetic fusion experiments, *Nucl. Instr. Meth in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* 476.1, 511 (2002)

[15] T. Nakamura et al., Present status of the fusion neutronics source (FNS), *In Proc. 4th Symp. on Accelerator Sci. and Technol.*, 24 (1982)

[16] H. Maekawa et al., Neutron yield monitors for the fusion neutronics source (FNS)(for 80.. beam line), *Rep. JAERI* (1983)

[17] C. Konno et al., Neutronics experiments for ITER at JAERI/FNS, *Nucl. fus.*, 41(3), 333 (2001)

[18] K. Ibano et al., Design studies of innovatively small fusion reactor based on biomass-fusion hybrid concept: GNOME, *Fusion Eng. Des.*, 86, 2779 (2011)

[19] K. Ibano et al., Neutronics and pumping power analyses on the Tokamak reactor forthe fusion-biomass hybrid concept, *Fusion Eng. Des.*, 88, 2881 (2013)

[20] S. Konishi et al., Possible Scenrio to Start Up DT Fusion Plant without Initial Loading of Tritium, *J. Plasma Fusion Res.*, 76, 1309 (2000)

[21] Y. Asaoka et al., Commissioning of a DT fusion reactor without external supply of tritium, *18th IAEA Fusion Energy Conf.* PDP-08 (2000)

[22] K. Shibata et al., JENDL-4.0: A New Library for Nuclear Science and Engineering, *J. Nucl. Sci. Technol.* 48, 1 (2011)

2. Neutronics Analyses of LiPb Blanket and Evaluation of Tritium Production Capability

2. 0. Brief Description of Chapter 2

This chapter evaluated tritium production and breeding behavior in an experimental LiPb block target and a LiPb blanket module with the neutron transport code MCNP using nuclear cross-section data from FENDL-2.1 libraries in order to establish fundamental the LiPb blanket design. The calculation results suggested that a sufficient tritium breeding ratio (TBR) can be obtained in the SiC-LiPb blanket concept and therefore a proper integral experiment on LiPb block with D-T neutrons can be planned for a small test module. Also, TBR, neutron shielding and nuclear heating in the module were evaluated. The expected tritium with existing fusion neutron generation devices in the world was estimated with the results of calculated TBR in this chapter. Also, more various blanket concepts were verified for advanced blanket design with several FW/coolant combinations and natural LiPb breeder including beryllium. Considered and evaluated blanket concepts are; (a) W armor, F82H first wall, SiC and helium coolant panel, natural LiPb breeder, (b) W armor, SiC first wall, SiC and helium coolant panel, natural LiPb breeder, (c) W armor, F82H first wall, SiC and helium coolant panel, natural LiPb breeder with Be neutron multiplier, (d) W armor, F82H first wall, SiC and helium coolant panel, enriched LiPb breeder, etc. Various water cooled concepts were examined at the same time. Tritium production by D-D neutron and the subsequent breeding were calculated to evaluate the possibility supply initial tritium for fusion reactor within a reasonable period of D-D operation. Even though use of enriched lithium-6 was found to be more effective than a neutron multiplier, partition walls of beryllium stand on different directions in the blanket are needed to obtain a sufficient TBR.

2. 1. Introduction

This chapter reports assessment of the LiPb blanket design parameters using the Monte Carlo method, specifically on the tritium production and breeding. There are several conceptual LiPb blanket designs in the world as application for ITER-TBM and DEMO blanket [ref 1-6]. All LiPb conceptual designs assume enriched LiPb for high tritium production performance. However, LiPb has a possibility to achieve TBR>1 without lithium-

6 enrichment using adequate neutron multiplier such as beryllium composites. The natural LiPb blanket design with neutron multiplier has different characteristics of neutron behavior or its tritium production property.

In this chapter, the related parameters with reflector material which affects the tritium production/breeding in an experiment target and optimization was considered for the considerable integral experiment with existing D-T neutron sources. Also, tritium breeding ratio (TBR), neutron shielding and nuclear heating in the module were obtained. The TBR can be defined as a ratio of produced tritium by nuclear reaction between neutron and lithium in a blanket to consumed tritium by D-T reaction. The D-T neutron integral experiment plan and measurable tritium amount with available neutron sources are evaluated with designed LiPb blanket on actual neutron generation devices [ref 7-9]. Also, due to the spatial restrictions of the model to be actually performed in the integral experiment as shown in figure 2-1, applications of reflectors were examined. A reflector to surround the blanket module can assist 1) neutron shielding and 2) tritium production and breeding in the test module.

In particular, for the early generation of fusion plants in the near future, it should be noted that no commercial source of enriched lithium-6 is available. The purpose of this chapter is to evaluate the TBR with a realistic thickness of armor and structural material, while obtaining sufficient tritium production/breeding performance using natural lithium for revising radial build data confidently without enrichment of lithium-6 as a breeder.

2. 2. Fundamental LiPb Blanket Design

The LiPb blanket was evaluated with the 3-D Monte Carlo neutron transport code MCNP-5.14 [ref 10] and fusion reference library FENDL-2.1 [ref 11]. All of the calculation models were chosen on the rectangular coordinate and the +z direction was the length direction. Figure 2-1 shows the model to investigate the effect of candidate reflector materials (F82H, Tungsten, Graphite, SiC, H_2O, SS316) surrounding the considerable LiPb blanket experimental module.

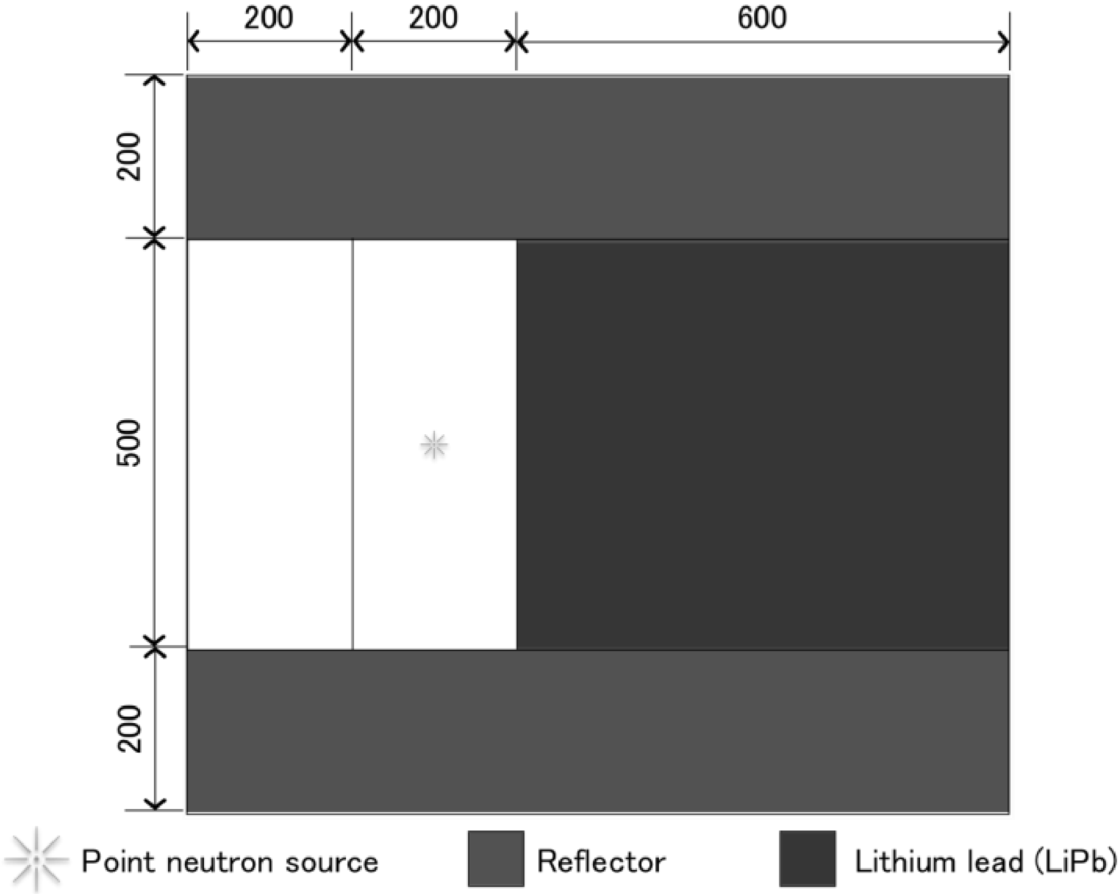

Fig. 2-1. The model with candidate reflector material (dimensions in mm)

Figure 2-2 shows the schematic conceptual LiPb model for considering self-sufficient TBR. The model consists of a first wall, LiPb breeder, SiC walls between each breeder and shield, assuming outboard blanket region. Also, the main parameters to optimize the module design for TBR values over unity are summarized in TABLE I. The parameter table includes reflector materials, LiPb breeder thickness, neutron multiplier, shield material, source neutron type etc.

TABLE I. Parameters for calculation

Breeder	Lithium lead (^{6}Li 90% enrichment)
Multiplier	Beryllium, Lead in LiPb
Reflector	Graphite, SiC, H_2O, F82H, Tungsten, SS316
Shielding material	SS316, Fe/H_2O, SS316/ H_2O, Boron, Boron Carbide
First wall thickness	0.5 cm to 3 cm
Breeder thickness	40 cm to 60 cm
Backside reflector	10 cm, SiC
Source neutron type	14.1 MeV, point (it can be assumed as a neutron generator), volume (it can be assumed as a plasma)

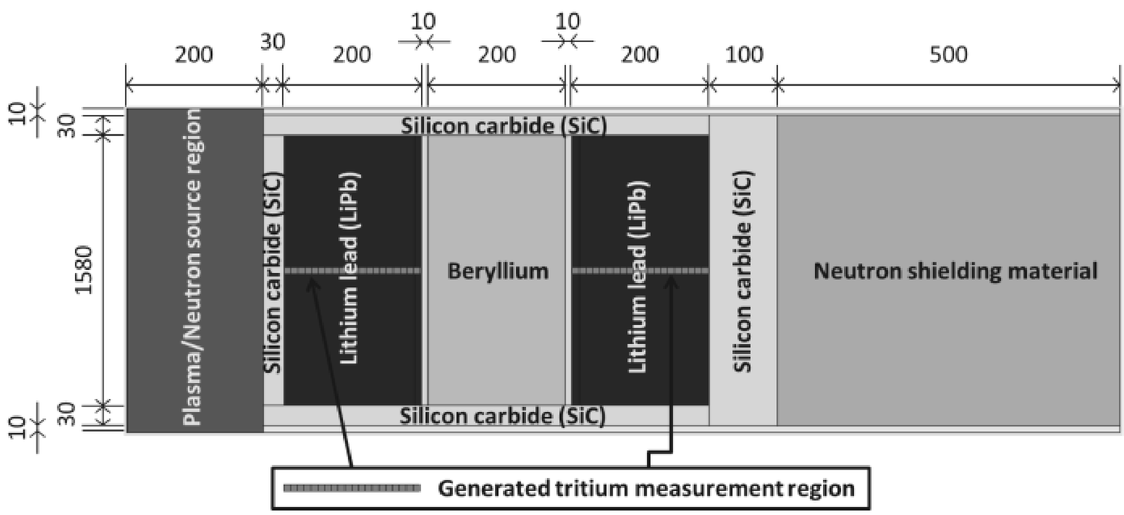

Fig. 2-2. Schematic of the conceptual LiPb blanket module (dimensions in mm)

2. 3. Results and Discussion of Fundamental LiPb blanket design

2. 3. 1. Effect of Reflector Materials

TBRs with various materials as a reflector were calculated for the model shown in figure 2-1. All TBRs in this chapter calculated with more than nps 10^6 for low relative error (<0.10). The nps means number of source particles *n* to be started. As shown in figure 2-3, the TBRs with graphite and SiC are 0.341 and 0.236, respectively. The contribution by the reflector is calculated as 86% and 80% respectively for the graphite and SiC. To obtain a high TBR, beryllium was considered as a reflector material at first, however, its risk for toxicity and cost for machinability would be a problem. Blanket design model with SiC_f/SiC allows up to 850 °C of outlet temperature [ref 12]. It is considered to be attractive to utilize LiPb blanket module with SiC panel for working at the high temperature. Therefore, the SiC-LiPb concept is applied as high temperature blanket module in this work.

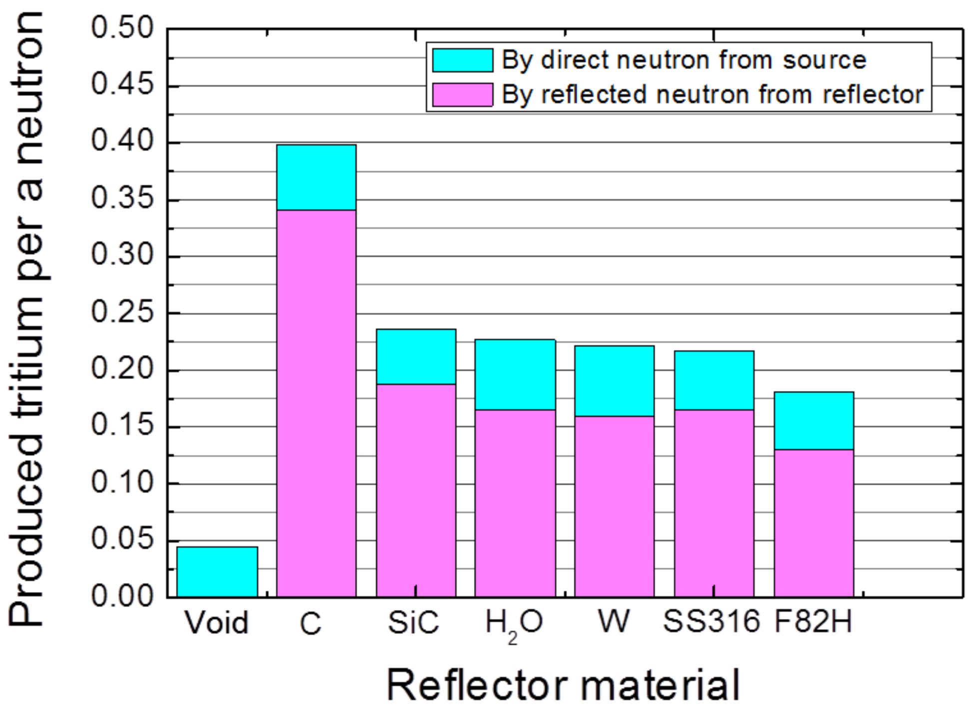

Fig. 2-3. Produced tritium per a neutron by various reflectors in the blanket module

2. 3. 2. First Wall Thickness and Backside Reflector

Although the first wall thickness is not the only parameter to decrease TBR, it is considered that it may have large influence on TBR. Japanese reduced-activation steel F82H, tungsten, vanadium alloys and copper alloys etc. [ref 13, 14] have been nominated for the first wall of the blanket. However, it has been found by the above results of MCNP calculations that SiC may be used as first wall material for its better reflective characteristic compared to F82H and tungsten. Also, SiC/SiC composites have been suggested for the primary candidate structural material already [ref 15].

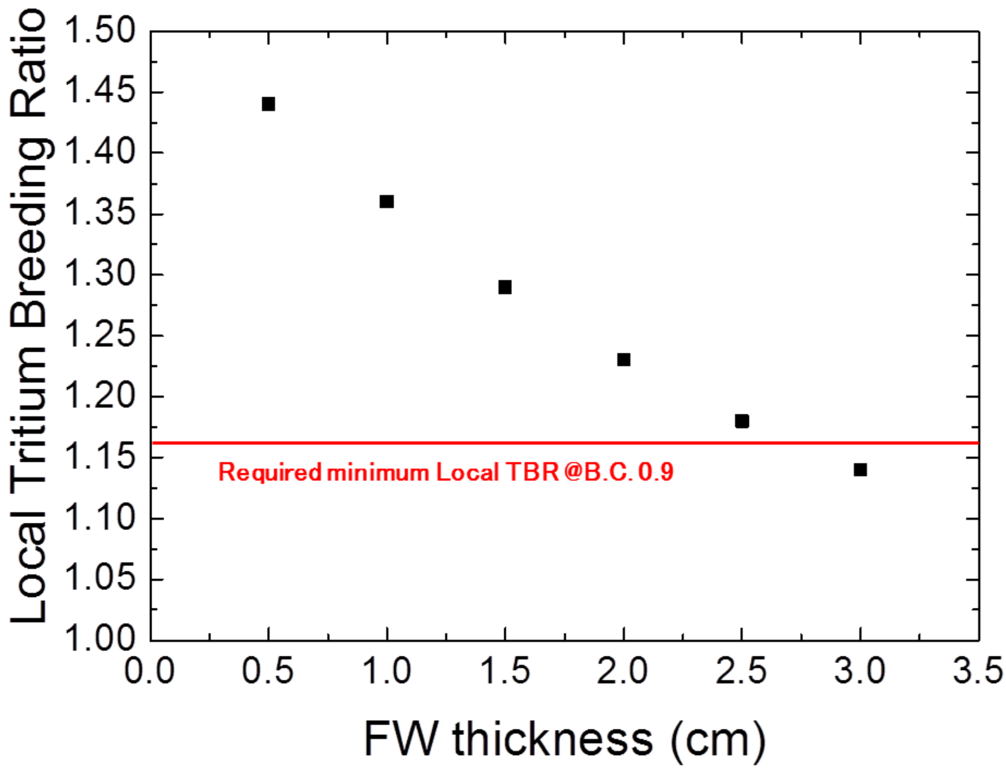

Fig. 2-4. First Wall thickness effect on local TBR

TBR with SiC first wall (in front of first LiPb region) for thickness of 0.5 cm, 1 cm, 1.5 cm, 2 cm, 2.5 cm, and 3 cm with 60 cm of LiPb has been estimated as 1.44, 1.36, 1.29, 1.23, 1.18 and 1.14 as shown in figure 2-4. A red line in figure 2-4 means required minimum local

TBR (TBR_{Local}=1.16) to achieve net TBR 1.05 with large blanket coverage 0.9. Local TBR continue to decrease as FW thickness increases. Because neutrons to produce tritium may be absorbed into the FW region. Even though large blanket coverage is assumed, FW thickness should be less than 3.0 cm for sufficient tritium breeding performance.

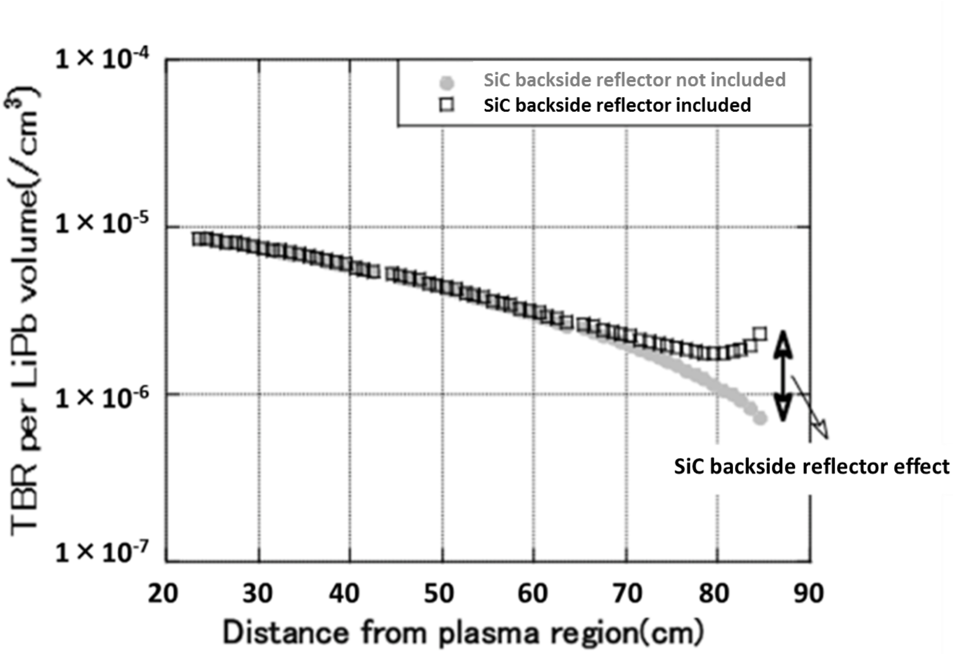

Fig. 2-5. Backside reflector effect on LiPb module

When the local TBR is not satisfied to required value with small differences, local TBR can be increased by backside reflector in LiPb blanket module. The effect of a backside reflector on TBR was also examined. The TBR can be increased by using a SiC backside reflector as shown in figure 2-5. Tritium production and neutron shielding can be aimed at the same time using by backside reflector in 60 cm of LiPb region.

2. 3. 3. TBR of SiC-LiPb Blanket Module

Figure 2-6 shows the depth dependence of TBR calculated for three types of breeder region in the blanket modules: (1) the type 1 includes 40 cm of LiPb breeder without SiC walls (1 cm) between each LiPb region, (2) the type 2 includes 40 cm of LiPb and SiC walls between LiPb regions, (3) the type 3 includes 40 cm LiPb and beryllium multiplier between 44 cm and 64 cm of LiPb region. In order to compare each TBR values correctly, type 1 and 2 have void regions between 44 cm and 64 cm. The total TBR values for the above were 1.04, 1.05 and 1.13, respectively. TBRs of type 2 and 3 increase when approaching the disjointed region (44~64 cm). For type 2, it is by back of the SiC wall of 1st LiPb, while for type 3, it is by back of the SiC/Be region of 1st LiPb. In the deeper region from 65 cm of LiPb, the TBR with beryllium (type 3) rapidly decreases compared with the other types of calculation results. Even though thermalized neutrons are increased by the beryllium multiplier, total neutron flux decreases as the beryllium thickness increases. TBR increased about 8% only by arranging beryllium between LiPb regions. The low increase rate can be predicted by unsuitable beryllium position in tritium breeding region. It should be arranged for neutron multiplication effectively with few neutron absorption into structure materials.

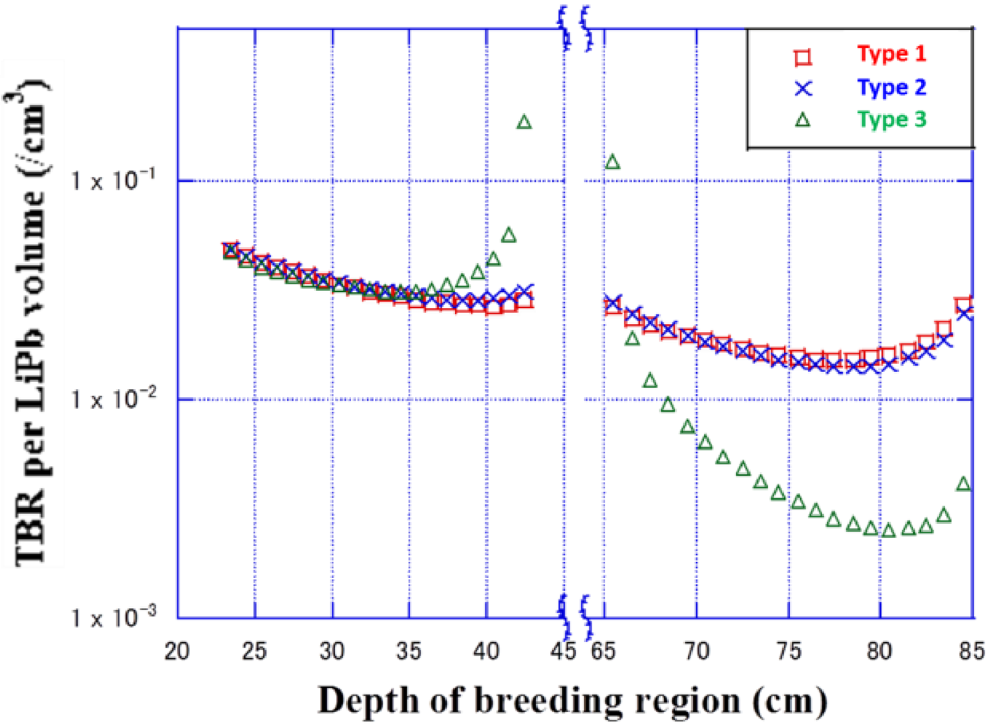

Fig. 2-6. TBR of three types of blanket modules

Through the comparison of figure 2-7 and 2-8, it is considered that neutrons are slowed down by beryllium and are absorbed by the SiC wall behind beryllium. All neutron flux results in this chapter calculated with more than nps $10^8 \sim 10^9$ for low relative error (<0.10). In particular, it is showed the flux difference at low energy neutron region. Since there are slowed-down neutrons increase, tritium production is increased also. Beryllium is worked as a neutron multiplier and a moderator in the fusion research field. Even though the simple design which has beryllium between LiPb regions was evaluated in this section, applying these dualities of beryllium to blanket design is effective method to produce tritium in LiPb blanket. Slowed-down neutrons by beryllium can be absorbed easily into neutron shielding region, LiPb blanket with beryllium affects also selection of shielding material. More advanced LiPb blanket design with beryllium examines in section 2. 4.

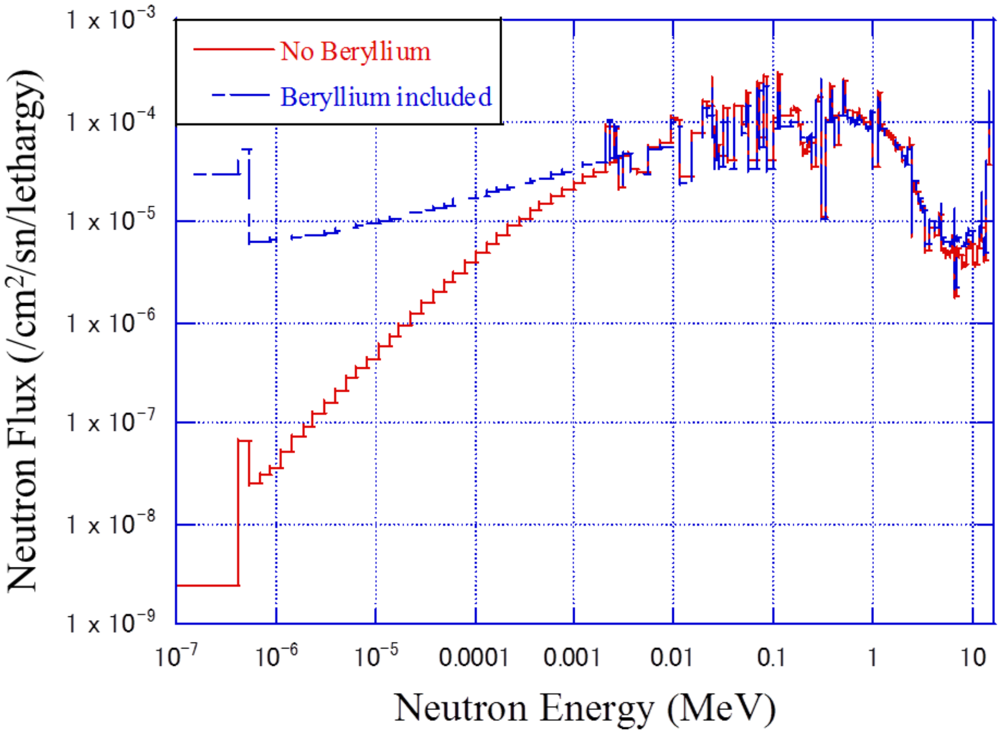

Fig. 2-7. Lethargy neutron spectrum moving through beryllium between SiC walls

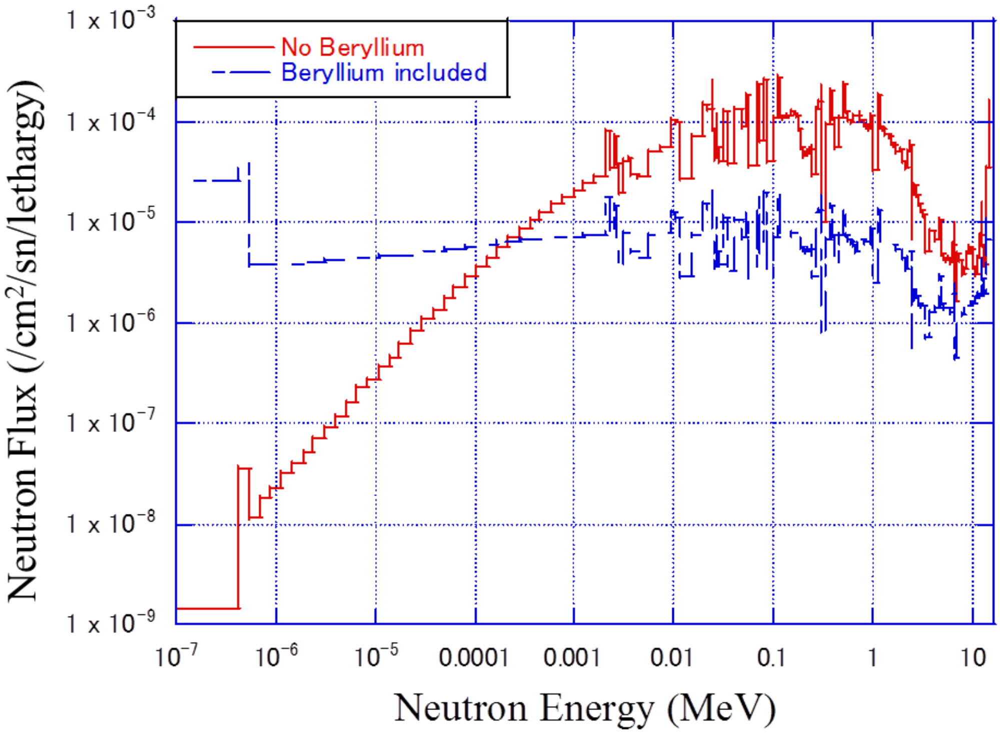

Fig. 2-8. Lethargy neutron spectrum moving through SiC wall between 1st LiPb and beryllium region

In order to clarify TBR change by beryllium thickness between LiPb regions, tritium production was evaluated with 5~30 cm of beryllium. As shown in figure 2-9, 10~20 cm of beryllium is required for high TBR between 20 cm of LiPb regions. Thicker beryllium causes to obstruct sufficient tritium production with absorption of slowed-down neutrons by moderation into SiC or other structure materials.

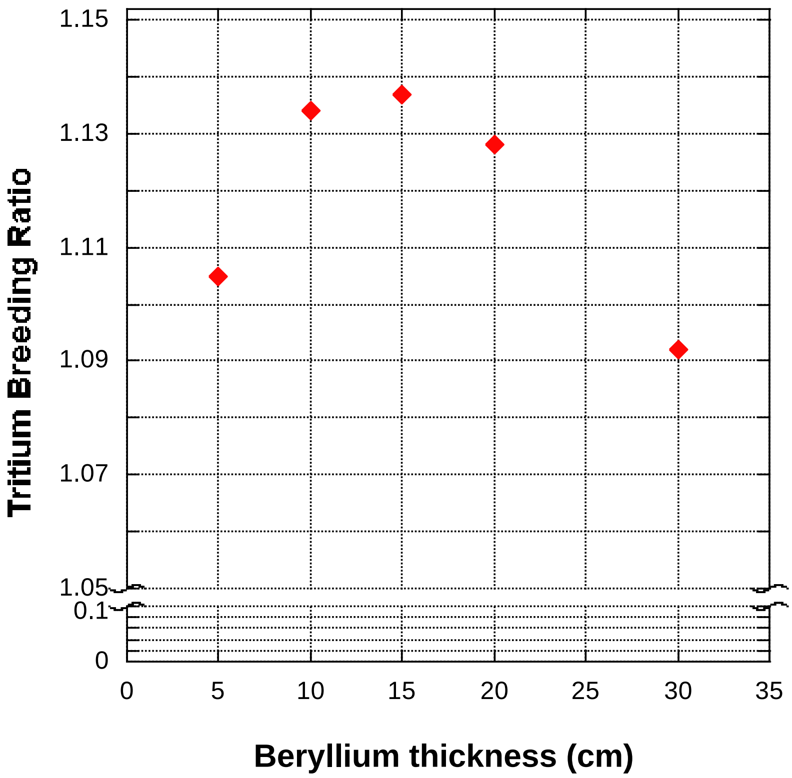

Fig.　2-9. TBR by Beryllium Thickness between LiPb regions

Lead in LiPb can be a neutron multiplier for high energy neutron above 7 MeV. One type of lead isotope, ^{208}Pb is naturally abundant with a long half-life, has also the largest (n, 2n) reaction from 12 MeV to 16 MeV. Since high energy neutron flux decreases in LiPb, the multiplication reaction by lead also decreases, and the lead in LiPb for the present module does not increase the TBR significantly. The (n, 2n) cross-section of ^{208}Pb (ref 16) is shown in figure 2-10. ^{208}Pb in the FENDL-2.1 data libraries have evaluated by ENDF/B-VI.

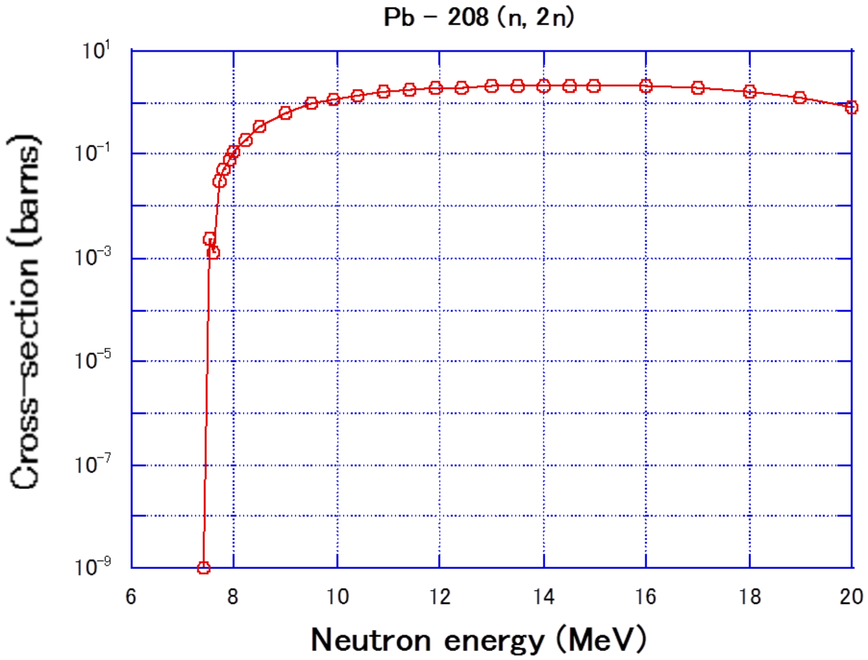

Fig. 2-10. (n, 2n) cross-section for ^{208}Pb (FENDL-2.1 from ENDF/B-VI)

2. 3. 4. Neutron Shielding Material

In order to protect superconducting coils by neutrons from fusion reactions on the inside and outside of the blanket, sufficient neutron shielding ability is required as the fusion blanket property. In ITER, toroidal field coils are Nb_3Sn, and its design limit is reported 17 kW of nuclear heating loads [ref 17]. Although nuclear heating loads by neutrons and gamma rays have to be verified for accurate evaluation of neutron shielding ability, calculated neutron flux is the evaluation index as a simplified evaluation method in this chapter. According to the back calculation from the limit, it is necessary to decrease the neutron flux in the shielding region to four-digit or more (required neutron flux $10^{-4} \rightarrow 10^{-8}$ in figure 2-11). The neutron shielding performance was evaluated with 5 kinds of shielding material candidates. The assumed thickness of neutron shielding region was 50 cm between the backside of SiC wall and the coil. Figure 2-11 shows neutron shield performance of various materials. Only SS316 is not suitable for neutron shield and Fe/H_2O and $SS316/H_2O$ are effective. Also, boron and B_4C have good neutron shield capability, because of the large thermal neutron capture cross section. The multi-layer neutron shield region in the blanket module should be applied for more efficient neutron shielding design, such as 1st layer for absorption of fast neutron region, 2nd layer for neutron moderator and 3rd layer for absorption of thermal neutron.

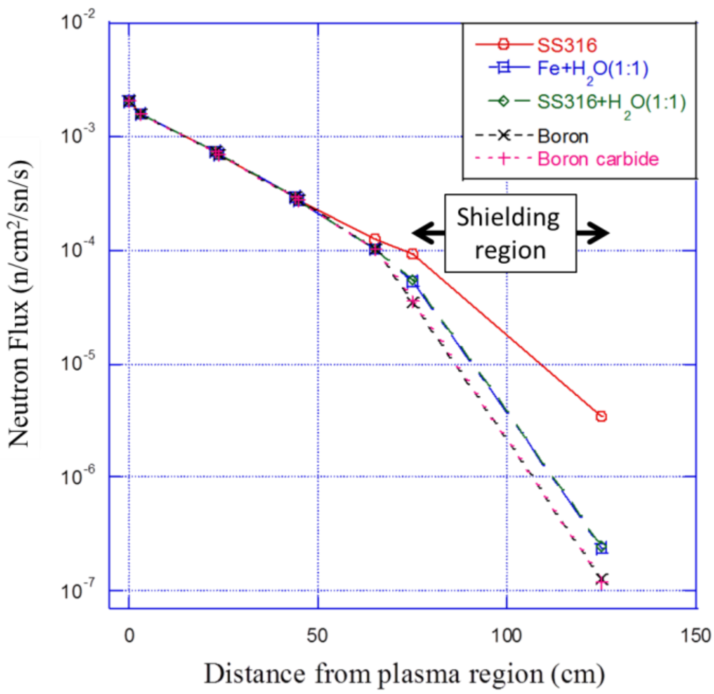

Fig. 2-11. Distribution of total neutron flux through the blanket module

2. 3. 5. Nuclear Heating in the Blanket Module

The nuclear heating from neutron and gamma in the blanket module was calculated for a neutron wall loading of 5 MW/m^2 [ref 18] respectively. The calculated power density values of each component are given in units of W/cm^3 and shown in TABLE III. The SiC-LiPb blanket in the DEMO reactor yields 41 W/cm^3 in first wall, 7 W/cm^3 in LiPb breeder. Furthermore, heat removal performance, temperature distribution of the blanket and temperature rise of helium gas have to be evaluated as a future study.

TABLE II. Power density in the SiC-LiPb blanket

Blanket component	Power Density (W/cm^3) At neutron wall loading 5 MW/m^2 of DEMO reactor
First wall	41.03
Whole Reflectors	11.43
Breeder (LiPb)	7.19
SiC wall between LiPb	1.36

2. 3. 6. Measurable Tritium in Integral Experiment System

For planning of the tritium breeding mock-up experiment of the designed LiPb blanket module at previous section, measurable tritium amount was estimated with parameters of various neutron sources as its first step. The tritium measurement method assumes that liquid scintillation counting (LSC) is used. Hence, in order to obtain an accurate result, the produced tritium amount requires at least 10 Bq for a decent measurement target size. Tritium measurement region (TMR) of the 1st LiPb breeder is shown in figure 2-12. It consists 20 small cubes 10 mm thick (t=10).

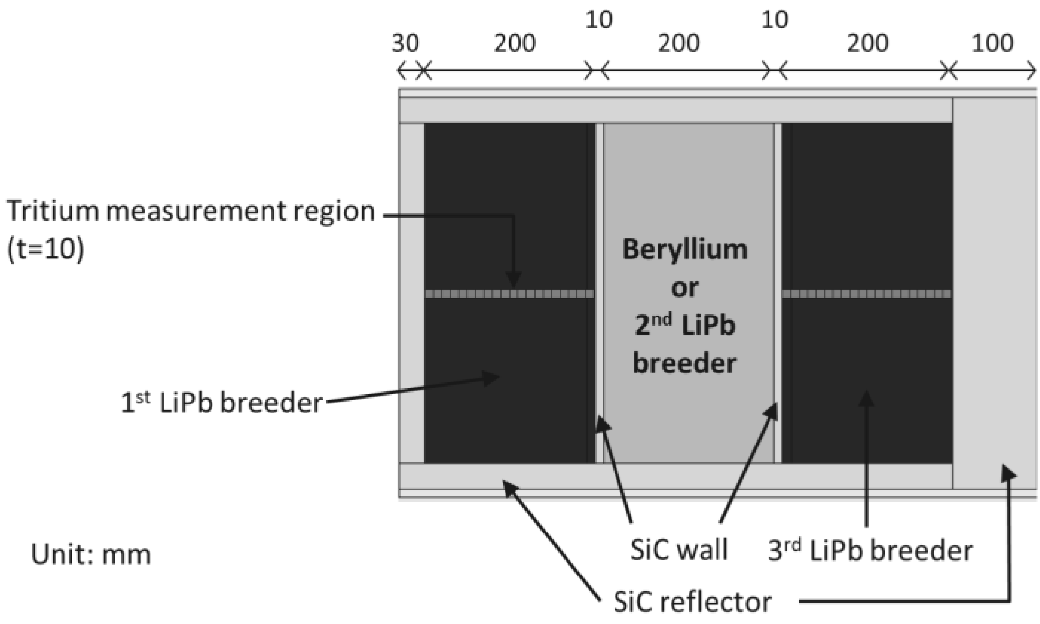

Fig. 2-12. Cross sectional view with three breeder region

The measurable tritium amount with beryllium multiplier (no beryllium multiplier) at the TMR in 1st LiPb and 2nd LiPb will be shown by each neutron generator. Each neutron generator cases and estimated results are shown in figure 2-13.

Operation time (hour)	3		14	
Neutron multiplier	Be	X	Be	X
Case1	90	65	422.5	297.5
Case2	15.1	10.6	70	50
Case3	3.7	2.1	14.8	9.9

Fig. 2-13. Estimated tritium (Bq/cm3) with various neutron generator cases

Each neutron generator cases are as follows:

- Case 1: 14.1 MeV neutrons at Fusion Neutronics Source of JAEA (3×10^{12} n/s)
- Case 2: 14 MeV neutrons at Frascati Neutron Generator of ENEA Energy Center (5×10^{11} n/s)
- Case 3: 14.7 MeV neutrons at IEC of the University of Illinois at Urbana-Champaign (10^{11} n/s)

Above measurable tritium is total amount for 40 small cubes 10 mm thick in all TMR. However, it is able to be measured accurately using LSC in a small cube (1 cm^3) having 10 Bq at least. In case 3, it can hardly obtain measurable tritium production for 3 h operation even with beryllium and 14 h operation without neutron multiplier.

2. 4. Advanced LiPb Blanket Design

In order to investigate the more realistic performance and feasibility of this concept, detailed neutronics analysis of more accurate structure of modules is needed. This section considers the effect of inner structure, inner structural materials, material of the first wall (FW), coolant combinations and armor material, natural lithium-6 in tritium breeding material, etc.

2. 4. 1. Neutronics analysis

In order to obtain tritium production/breeding performance in the blanket, Tritium Breeding Ratio (TBR) is calculated using Monte Carlo neutron transport code, MCNP-5.14 and fusion evaluated data libraries, FENDL-2.1. The TBR is defined as a ratio of bred tritium by nuclear reaction between a neutron and lithium in the blanket to burnt tritium by D-T reaction. Also, a number of Tritium Production per D-D neutron (used symbol TBR_{DD}, it means TP_{DD}) is defined in this research for neutronics of D-D neutrons. The blanket has same height (direction of poloidal coil) and depth (direction of toroidal coil) of ITER-Test Blanket Module (TBM) [ref 19], 1660 mm and 480 mm respectively. Because, ITER-TBM is only settled size of blanket module at this moment except design models for DEMO reactors of each countries. Fundamentally, LiPb breeder is 60 cm (radial) in spite of different concept of each model. The x-z cross section view of the calculation model is shown figure 2-14. All of the module models have reflective boundaries. Therefore, calculated TBRs in the research should be converted with coverage of whole blanket before applying for specific fusion reactor. GNOME reactor in figure 2-14 is a scheme for conceptual design of a biomass fusion hybrid reactor, it was designed and published as a design study already [ref 12]. Also, detailed information of used materials and its density are shown in TABLE IV. 14.1 MeV and 2.45 MeV of volume source are used as the neutron source condition.

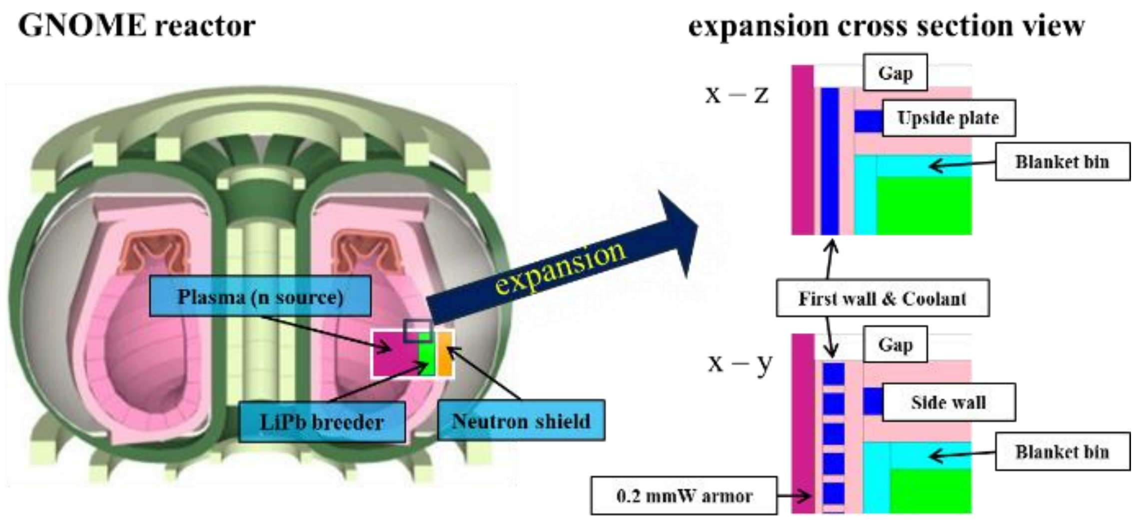

Fig. 2-14. Cross sectional view of calculated blanket model

2. 4. 2. Neutronics approach for various blanket concept

Tritium production/breeding performance has been evaluated on various concept blanket model. Considered parameters affected on TBR are as follows: first wall material, its thickness, coolant, neutron multiplier and enrichment of lithium-6. Also, calculated TBR_{DD} has same parameters in order to apply the value for next chapter about scenario evaluation of D-T fusion plant operation.

TABLE IV. Details of used materials

Mateirials (Symbol)	**Density (g/cm^3)**	**Consisted elements or isotopes**
Water (H_2O)	1.00	66% Hydrogen, 33% Oxygen
Helium (He)	1.786e^{-4}	^{3}He (0.000001%), ^{4}He (99.99999%)
Tungsten (W)	19.25	^{182}W (26.5%), ^{183}W (14.3%), ^{184}W (30.6%), ^{186}W (28.4%)
Lithium lead (LiPb)	9.59	17% Lithium (^{6}Li 7.4%, ^{7}Li 92.6%), 83% Lead
Silicon Carbide (SiC)	3.12	50% Silicon, 50% Carbon
Shielding material	6.6	80% SUS316, 20% water
F82H	7.53	F82H IEA heat
Beryllium (Be)	1.432	^{9}Be, 80% Packing factor

2. 4. 3. TBR obtained for natural LiPb breeder and beryllium neutron multiplier

In the natural LiPb breeder model, TBR was evaluated with beryllium content. In the previous section 2. 3 for early generation blanket, majority is enriched lithium-6 solid breeder with neutron multiplier that achieved high TBR. Enriched tritium breeder will be used for high TBR in liquid blanket as well. However, taking advantage of low pressure because of the self-cooling nature of liquid blanket, LiPb has a possibility to achieve TBR>1 without lithium-6 enrichment. The analysis result will be used for comparing pros and cons of using beryllium as a neutron multiplier.

2. 4. 4. Effect of Partition wall direction on TBR

There are two kinds of inner structure types for the blanket by direction of partition walls included a coolant panel as shown in figure 2-15. For typical examples, Japan Water-Cooling Ceramic Breeder (WCCB) TBM is one of the (a) radial layer type, and EU Helium-Cooling Lithium Lead (HCLL) TBM is one of the (b) poloidal layer type. The difference of inner structure types of the blanket can be affected on decrease tendency of neutron energy, because it determines the direction of collision when neutron is entering to materials. In other words, the difference of inner structure types brings about the difference of tritium production/breeding performance. The effect on TBR can be confirmed when inner partition walls of beryllium stand on different directions in the blanket with same material condition, also studied about which type of the blanket have an advantage from neutron economical point of view.

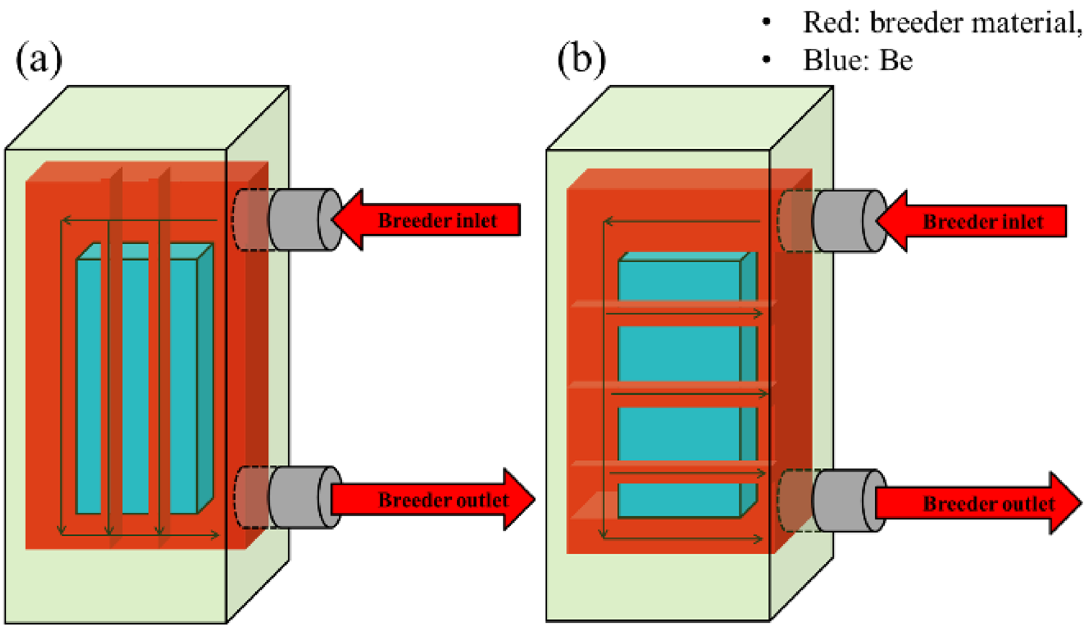

Fig. 2-15. Virtual blanket models with different direction partition walls; (a) radial layer type, (b) poloidal layer type

2. 5. Results and Discussion of Advanced Blanket Design

2. 5. 1. Tritium production/breeding performance of various blanket concepts

Calculated TBRs with various concepts of the blanket are shown in TABLE V. TBR by D-T neutrons (TBR_{DT}) and TBR_{DD} are calculated for most of considerable cases in this study. In both of helium-cooling and water-cooling systems with LiPb breeder blanket, SiC is more suitable as first wall material for high TBR_{DT} and TBR_{DD}. Also, in order to obtain high TBR, effective combination of first wall material and cooling material can be F82H-helium or SiC-water. Although proper other conditions have to be prepared, the possibility of sufficient TBR can be secured using natural LiPb breeder together with beryllium. In cases of the natural LiPb with beryllium, the ratio of beryllium to LiPb is 3:1 for high TBR as shown in figure 2-16. Also, according to the results in figure 2-16, TBR is almost constant in the F82H-water and SiC-water cases even when the ratio of beryllium is increased. In the cases of natural LiPb with beryllium, water is determined by an obstacle that beryllium can act as neutron multiplier. Even though there is only one case (SiC-helium combination) that TBR exceed unity, sufficient TBR can be achieved using natural LiPb with beryllium. When high enriched LiPb is used as a breeder, F82H-helium is a feasible combination to obtain sufficient TBR. The combination of F82H-water in all cases has lower TBR than other combinations such as F82H-helium, SiC-helium and SiC-water, because the combination shields neutrons effectively. Unusual results can be figure out through comparison of SiC-helium and SiC-water without beryllium in TABLE V. SiC-water has large tritium production/breeding performance than SiC-helium. It is regarded that SiC is low Z material, and thus its moderation ability is not enough to get a sufficient tritium production without neutron moderator such as beryllium and water.

TABLE V. Results of calculated TBRs for various concepts

First wall	**Coolant**	**Breeder**	**TBR_{DT} (TBR_{DD})**
F82H	Helium	Natural LiPb	0.683 (0.475)
F82H	Water	Natural LiPb	0.576 (0.419)
SiC	Helium	Natural LiPb	0.853 (0.674)
SiC	Water	Natural LiPb	0.955 (0.785)
F82H	Helium	Natural LiPb with Be	0.831 (0.560)
F82H	Water	Natural LiPb with Be	0.583 (0.404)
SiC	Helium	Natural LiPb with Be	1.041 (0.789)
SiC	Water	Natural LiPb with Be	0.979 (0.769)
F82H	Helium	90% Enriched LiPb	1.204 (0.855)
F82H	Water	90% Enriched LiPb	0.931 (0.684)
SiC	Helium	90% Enriched LiPb	1.134 (0.907)
SiC	Water	90% Enriched LiPb	1.094 (0.902)

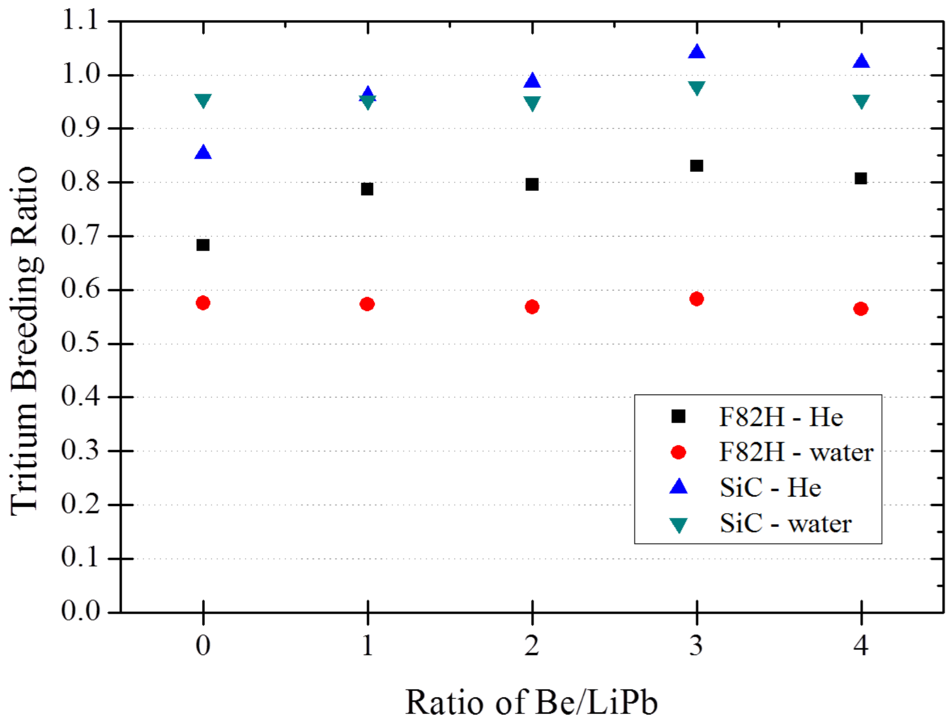

Fig. 2-16. TBRs by ratio of ratio of beryllium to LiPb breeder for each case

2. 5. 2. Tritium production/breeding performance of natural LiPb with different beryllium layer directions

Figure 2-17 shows prepared MCNP models for two types of inner structure of the blanket. TBR and TBR_{DD} of models were calculated with various cases such as different first wall materials and coolant materials. In this calculation, only natural LiPb with beryllium was selected as a breeder and a neutron multiplier. In order to examine the impact of only the difference of the inner structure, two types of inner structure of the blanket have same amount of materials for the TBR calculation. Calculated results are summarized in TABLE VI. As the result of comparison with different direction partition wall, radial layer type had slightly higher TBR_{DT} and TBR_{DD} mostly. However, both types had higher TBR_{DT} and TBR_{DD} than in the cases without partition walls of beryllium. And most of cases had TBR_{DT} and TBR_{DD} exceeded unity except the combination of F82H-water. F82H-water acts a neutron shielding material like the precedent cases in TABLE V. As with above mentioned cases that natural LiPb with beryllium, sufficient TBRs can be achieved without lithium-6 enrichment.

Even though attainable TBR is limited, there can be able to improve the design natural LiPb with beryllium through various beryllium structures set in the further study.

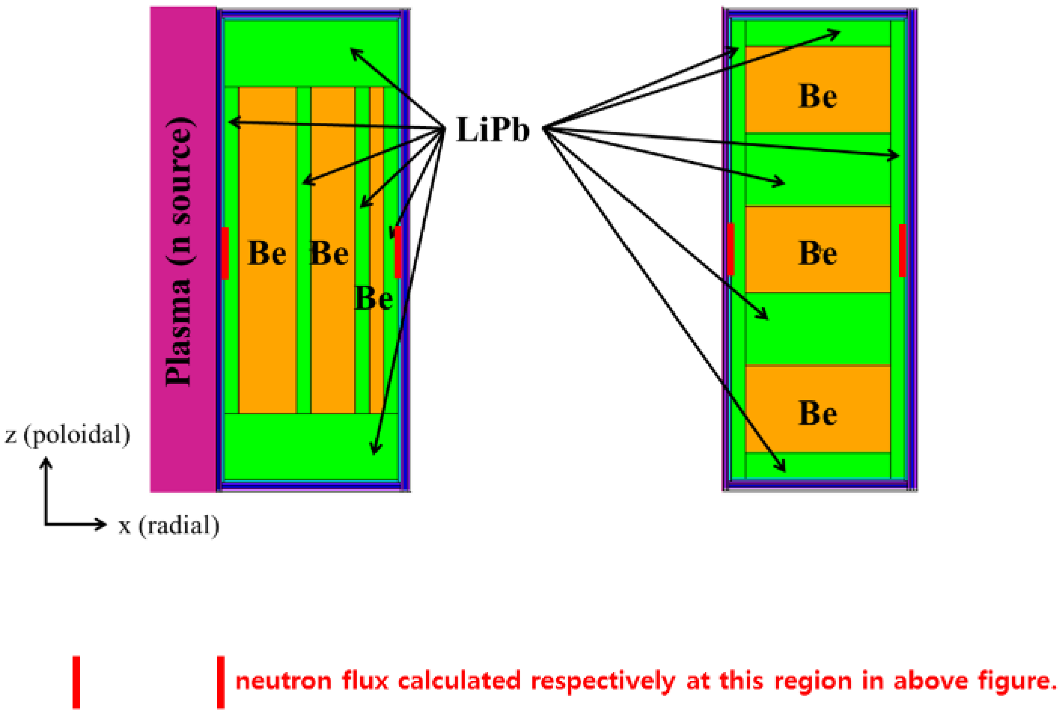

Fig. 2-17. MCNP models for comparison between different direction of partition wall, radial beryllium layer type (left), poloidal beryllium layer type (right)

As the typical result, the neutron flux of F82H-water and SiC-helium are shown in figure 2-18. This result shows that radial beryllium layer type and poloidal beryllium layer type have similar tendency of tritium production property. However, neutron behavior in each layer type has to be evaluated more for accurate understanding of its phenomena.

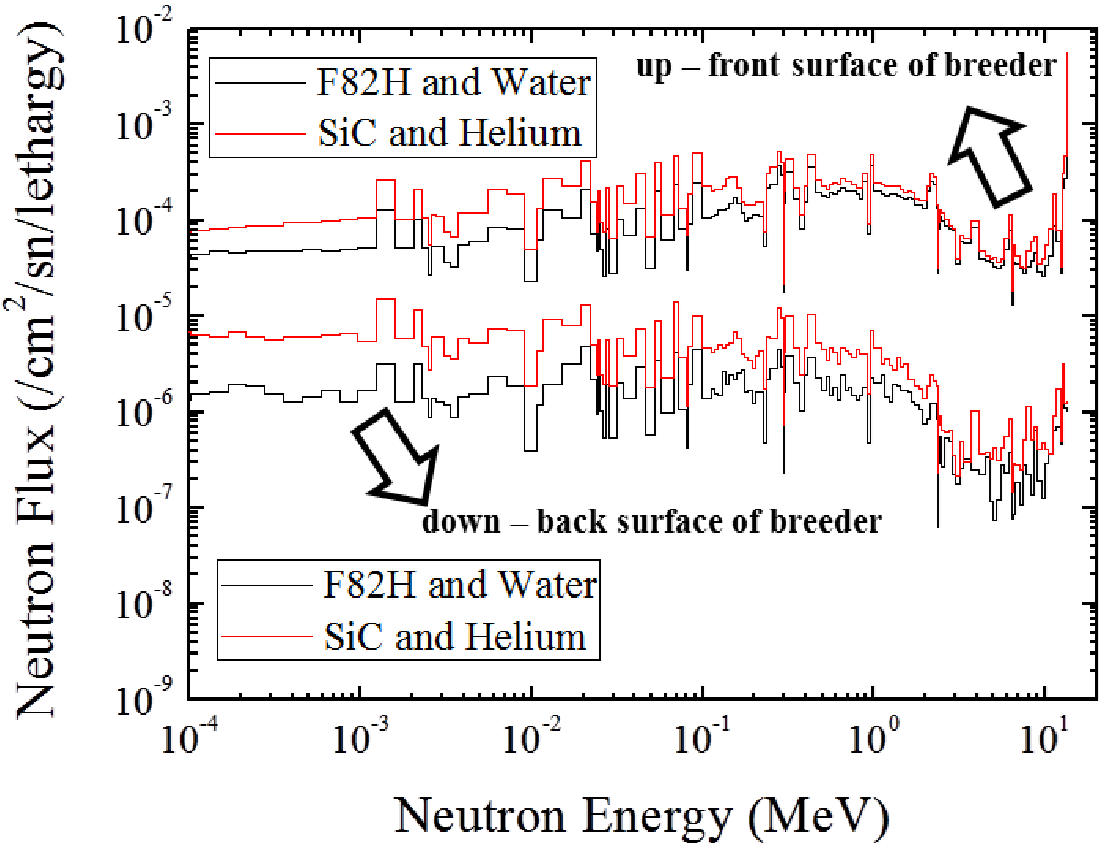

Fig. 2-18. Neutron flux of radial beryllium layer type with different FW/coolant combination

The neutron flux of F82H-water is lower than the case of SiC-helium. The neutron flux is decreased in both of cases. It can be said that SiC-helium has bad neutron shielding performance. Verifying shielding performance of each combination may be helped by checking the neutron flux at the outside of neutron shielding region.

TABLE VI. Calculated TBRs for radial beryllium layer and poloidal beryllium layer

Blanket type	First Wall	Coolant	Breeder Region	TBR_{DT} (TBR_{DD})
Radial	F82H	Helium	Natural LiPb and Be	1.031 (0.686)
		Water		0.747 (0.504)
	SiC	Helium		1.116 (0.823)
		Water		1.066 (0.815)
Poloidal	F82H	Helium	Natural LiPb and Be	1.020 (0.688)
		Water		0.728 (0.499)
	SiC	Helium		1.105 (0.827)
		Water		1.051 (0.813)

2. 6. Chapter Summary

In order to evaluate the feasibility and optimization for the parameters of integral experiment of small test module, TBR, effect of shielding material, nuclear heating in LiPb blanket module were calculated by neutron transport code MCNP-5.14 with nuclear cross-section data from FENDL-2.1 libraries. The measurable tritium amount for each generator was calculated with the results of TBR and actual neutron generation devices.

The numerical simulation showed some interesting features of advanced LiPb blanket design. In order to obtain high TBR, using enriched lithium-6 in the tritium breeder is more effective than use of neutron multiplier such as beryllium. However, natural LiPb with beryllium design was verified to achieve sufficient TBR. Even though, in this chapter, it was not compared that the cost and risks between using natural LiPb with beryllium and using enriched LiPb without beryllium yet. Using natural LiPb with beryllium in tritium breeding region can be worthwhile way to be faced expectable problem of lithium-6 as a natural resource in the future. Combination of SiC-helium where TBR exceed unity, suggests the possibility of natural LiPb with beryllium.

Using a neutron multiplier such as beryllium is unavoidable necessity for achievement of sufficient TBR using natural LiPb. In this chapter, Comparison with different direction partition wall, radial beryllium layer type had higher TBRs in most of the cases. However, an accurate thermo-mechanical analysis and concerns about risks of beryllium should be considered also.

References

[1] Y. Poitevin et al., The European breeding blankets development and the test strategy in ITER, Fusion Eng. Des., 75, 741 (2005)

[2] P. Norajitra et al., Conceptual design of the dual-coolant blanket within the framework of the EU power plant conceptual study (TW2-TRP-PPCS), Final Report, FZKA 6780 (2003)

[3] S. Malang et al., An example pathway to a fusion power plant system based on lead–lithium breeder: Comparison of the dual-coolant lead–lithium (DCLL) blanket with the helium-cooled lead–lithium (HCLL) concept as initial step, *Fusion Eng. Des.*, 84, 2145 (2009)

[4] C. P. C. Wong et al., Overview of liquid metal TBM concepts and programs, *Fusion Eng. Des.*, 83, 850 (2008)

[5] Y. Wu, Conceptual design of the China fusion power plant FDS-II, *Fusion Eng. Des.*, 83, 1683 (2008)

[6] L. El-Guebaly et al., TBR and shielding analyses in support of ST-FNSF study, Fusion Engineering (SOFE), 2013 IEEE 25th Symposium on. IEEE (2013)

[7] T. NAKAMURA et al., "Present Status of the Fusion Neutronics Source (FNS)," *Proc. 4th Symp. on Accelerator Science and Technology*, Riken, Saitama, Japan, November 24–26, p. 155 (1982)

[8] M. MARTONE et al., "The 14 MeV Frascati neutron generator," *J. Nucl. Mater.*, 1661, 212 (1994)

[9] G. H. MILEY et al., "IEF-based neutron generator for security inspection system," *J. Radioanal. Nucl. Chem.*, 263, 159 (2005)

[10] X-5 MONTE CARLO TEAM, *MCNP – A General Monte Carlo N-Particle Transport Code*, Version 5, LA-UR-03-1987, LA-CP-03-0245, LA-CP-03-0284, Los Alamos National Laboratory (2003)

[11] D. L. ALDAMA et al., *FENDL-2.1: Fusion Evaluated Nuclear Data Library for Fusion Applications*, IAEA report INDC-NDS-467 (2004)

[12] K. IBANO et al., "Design studies of innovatively small fusion reactor based on biomass-fusion hybrid concept: GNOME," *Fusion Eng. Des*, 86, 2779 (2011)

[13] E. E. BLOOM, "The challenge of developing structural materials for fusion power systems," *J. Nucl. Mater.*, 7, 258 (1998)

[14] K. IOKI et al., "Design and material selection for ITER first wall:blanket, divertor and vacuum vessel," *J. Nucl. Mater.*, 74, 258 (1998)

[15] D. L. SMITH et al., "Materials integration issues for high performance fusion power systems," *J. Nucl. Mater.*, 65, 258 (1998)

[16] National Nuclear Data Center (NNDC) at Brookhaven National Laboratory (BNL), http://www.nndc.bnl.gov

[17] E. T. CHENG et al., "Progress of the ST-VNS Study," *Fusion Technol.*, 34, 1066 (1998)

[18] R. T. Santoro et al., "Status of the radiation shielding analyses for ITER" *Fusion Engineering*, 2. 995 (1997)

[19] M. A. Abdou et al., US ITER TEST BLANKET MODULE PROGRAM, REPORT NO. UCLA-FNT-216 (2007)

3. Performance Evaluation of D-D Start-up Scenario for Production of Initial Tritium with LiPb Blanket

3. 0. Brief Description of Chapter 3

A scenario for initial tritium to reach steady state D-T operation by D-D commissioning phase of the fusion plant is presented. By the D-D reaction, resulted T and neutron that produces T in the breeding blanket will be accumulated. A Neutronics of LiPb liquid blanket with 90% enriched lithium-6 was evaluated with neutron transport code MCNP-5.14 using FENDL-2.1 data libraries. System dynamics code STELLA-10.0.2 was used for the analysis of tritium system and the scenario of the breeding. As the result, calculated local tritium breeding ratio in the blanket is 1.20 by D-T neutron and 0.84 by D-D neutron. The required extension period of D-D commissioning operation without initial tritium is estimated closely 90 days, depending on the coverage and inventory. These results reveal that launching D-T fusion projects without initial tritium is possible with additional D-D operation period to obtain extra tritium to start up D-T operation.

3. 1. Introduction

Research on nuclear fusion is regarded with smooth progress since the beginning of JET of Europe, JT-60 of Japan, and world collaborative project ITER under construction in France today. Although tritium as D-T nuclear fusion fuel exists about 20 kg in the world, the most of tritium will be burned and disappear after ITER operation. Now, Darlington plant in Canada and Wolsong plant in South Korea are merely available commercial source of tritium that extracts tritium from heavy water reactor. When other D-T fusion project will be started, the quantity of tritium will be insufficient. Therefore, in order to launch D-T fusion project after ITER, the proposal of management plan could be a critical issue for tritium fuel reservation. It is widely believed that initial loading of tritium is inevitable for fusion plants.

In order to reduce or even avoid the initial loading of tritium, an innovative fuel scenario has been proposed already. Tritium can be produced by D-D reaction, and in breeding blanket by D-D neutron which occurs mainly between neutral beam and plasma, and the resulted tritium will be burned in D-T reaction to breed according to the breeding ratio exponentially to fill the fuel loop. Eventually bred tritium will reach the D-T steady state operation. It is called D-D start-up scenario [ref 1, 2]. The conceptual diagram of the scenario in this research is shown in figure 3-1. Tritium will be produced by three ways in this scenario. D-D plasma will very likely to be used in the commissioning phase of the D-T fusion plant. In the first stage operation, D-D reaction, probably dominated by the ions from high energy beam makes small amount of tritium. And after that tritium is produced from lithium in bleeding blanket D-D neutrons generated by a half of D-D reaction (termed T1 in below diagram). Also, tritium is produced by another half of D-D reaction (T2 in below diagram). T1 and T2 will be an initial source for D-T reaction. After those D-D reactions, there are deuterium and small amount of tritium in the plasma. Tritium is consumed for burning, and D-T neutrons generated by D-T reaction (T3 in below diagram). D-T reaction rate is significantly larger than D-D reaction around 1 keV [ref 2]. Therefore, if there are deuterium and tritium at the same time, D-T reaction will dominate the burning soon. All the produced tritium by D-D reaction is immediately recycled as fuel. And tritium in the fuel loop is bred by the effective overall Tritium Breeding Ratio (TBR) that should be greater than unity. In this chapter, TBR in LiPb blanket is calculated using Monte Carlo neutron transport code. Using the TBR values (TBR_{DT} for DT neutron and TBR_{DD} for D-D neutron), the D-D start-up scenario by system dynamics (SD) simulation is evaluated. The approach of this chapter will be showed the establishing possibility of the scenario in fusion reactor system as a solution plan of restriction of fusion reactor construction by doubling time. Also, the time variation of tritium stock and flow is examined with considerable tritium stock status from zero to small amount of initial loading tritium.

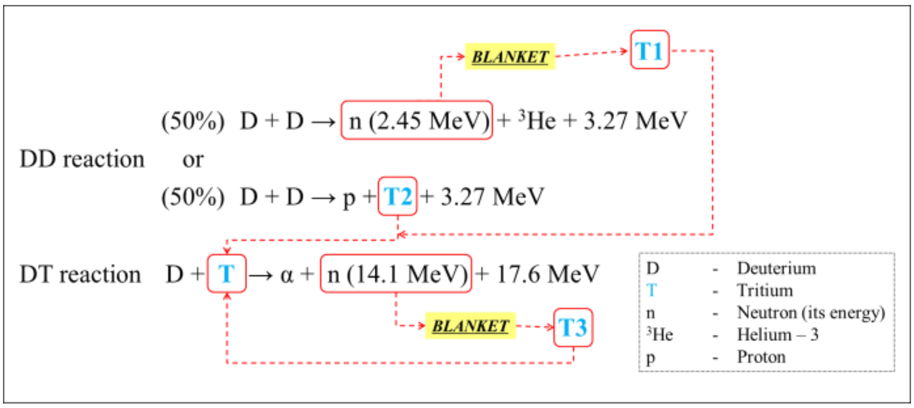

Fig. 3-1. Conceptual diagram of D-D start-up scenario

3. 2. Methodology

3. 2. 1. Calculation of TBR as an important factor for scenario evaluation

Tritium production in the blanket is considered to be one the most important factor for scenario evaluation. In this chapter, TBR is calculated using Monte Carlo neutron transport code, MCNP-5.14 [ref 3] and fusion evaluated data libraries, FENDL-2.1 [ref 4]. The TBRs are derived with changes of tritium production performance after lithium-6 burned in existing blanket design of chapter 2. A uniform plasma as neutron source was assumed in this calculation. All of the TBR was calculated for 90% enriched lithium-6 in 17Li83Pb as a breeder.

Calculated local TBR by MCNP is converted to net TBR in light of a blanket coverage. Although it will be strongly dependent on with design of fusion reactors, the blanket coverage is roughly a ratio of the blanket surface area to total plasma surrounding surface area. Divertor, various ports, gap space between other blanket modules, and side wall of the blanket occupy 10% ~ 30% of total surface area. Therefore, the blanket coverage has value of 0.7 ~ 0.9. This calculation is considered the gap space between other blanket modules and side wall of the blanket could be smaller due to low pressure in this calculation model, and blanket coverage can be accounted 0.75 ~ 0.95 for minimal openings for diagnosis for realistic plant. This may be optimistic compared with the other TBR evaluation models and the local TBR must be improved. But the product of coverage and TBR should be around 1 in the design of the entire plant to evaluate sensitivity and to achieve self-sufficiency for any reactors. Used parameters to calculate TBR in this paper is shown in Table I. The z-x sectional view of the calculated model is shown in figure 3-2. The applied materials in this model are; F82H IEA heat as first wall material, silicon carbide (SiC) as bin material for LiPb and He gas as a coolant. LiPb includes enriched lithium-6 to 90%. Neutron shield material is set to a mixture of SUS316 and water (ratio 8:2).

TABLE III. Details of calculation model

Part name	Calculation model details	
	Material	Thickness (cm) of radial direction
First wall	F82H IEA heat (F82H)	1.5
Armor	Tungsten (W)	0.2
Coolant	Helium gas (He)	0.5
Breeder	17Li82Pb, lithium-6 90% enriched	60.0
Blanket bin	Silicon Carbide (SiC)	1.0
Gap	Void	1.0
Shield	Stainless steel (SUS316), H_2O	100.0

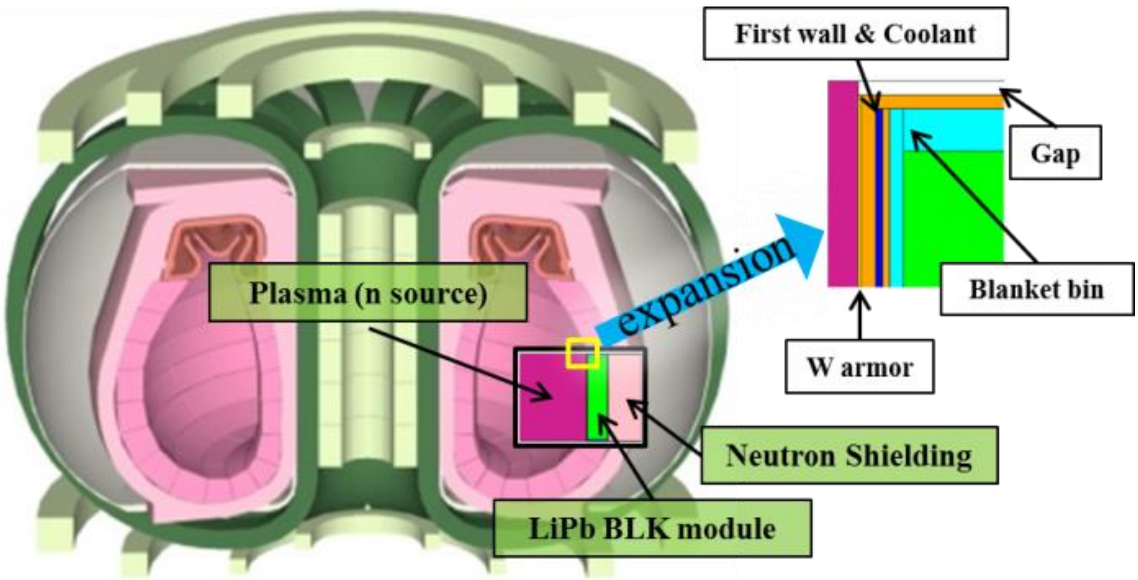

Fig. 3-2. A vertical cross section view of blanket module for MCNP modeling

3. 2. 2. D-D start-up Scenario Evaluation using System Dynamics Code, STELLA™

In this study, the D-D start-up scenario is analyzed by SD simulation software STELLA™ v10.0.2 (isee systems, Inc). A simplified stock and flow diagram of tritium in the fusion reactor plant model is shown in figure 3-3. Also, parameters used for the model are shown in Table II. The stocks in the main flow loop are the active inventory N_i where component i consists of plasma (p), blanket (blk), fuel clean up (fcu), isotope separation system (iss) and fueling and storage (fs). Tritium flow from upstream N_i to downstream N_j is defined with the mean residential time τ_i as following equation:

$$dN_j / dt = N_i / \tau_i \,. \qquad (1)$$

In the steady state, i.e. tritium concentration in plasma $N_T/(N_D+N_T)$ reach 0.5, tritium flow from N_{fs} to N_p, that is tritium fueling, is suppressed to keep the steady value.

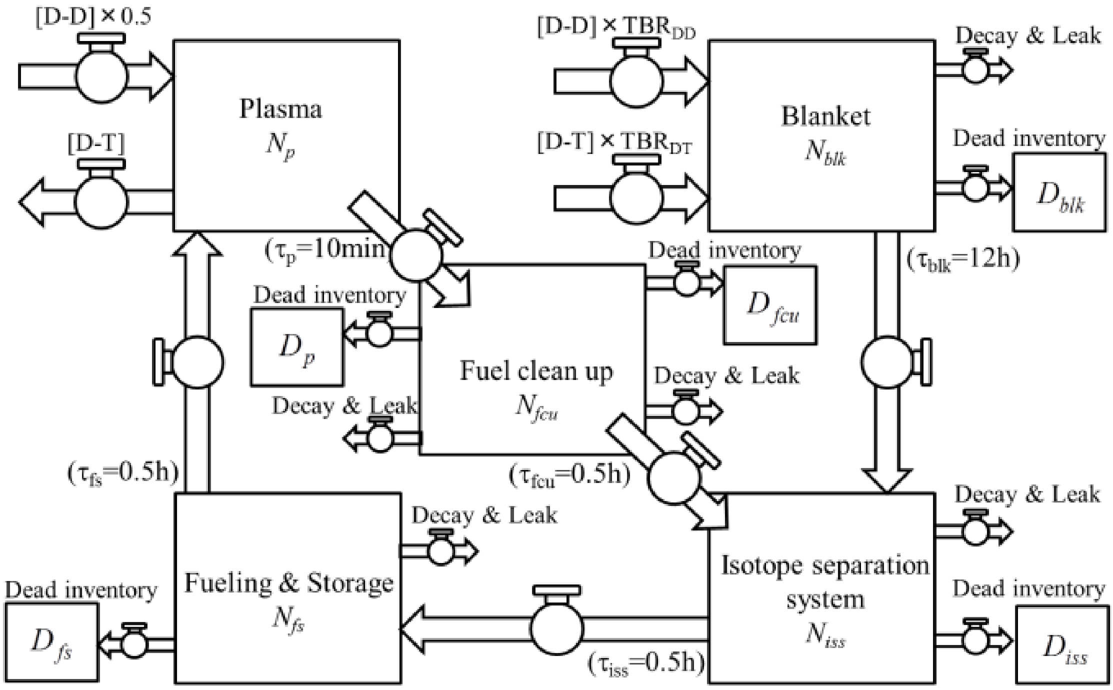

Fig. 3-3. Tritium stocks & flow diagram of the fusion reactor plant model in this study (simplified)

TABLE II. Estimation parameters for STELLA

Stocks, I	Mean time of inventory, τ_i [min]	Saturated dead inventory, D_i^{sat} [gram]	Exchange rate [s-1]	Leak rate [s-1]
Plasma	10	250	1.16×10^{-6}	1.79×10^{-9}
Fuel clean up	30	100	1.16×10^{-6}	1.79×10^{-9}
Isotope Separation System	30	50	1.16×10^{-6}	1.79×10^{-9}
Fueling & Storage	30	50	1.16×10^{-6}	1.79×10^{-9}
Blanket	720	50	1.16×10^{-6}	1.79×10^{-9}

Each stock connects to each sub-stock showing the dead inventory that is tritium on/in the facing material of the components that is not circulated in the flow and used in the reaction. The tritium flow rate into the dead inventory is the exchange rate between tritium and deuterium/hydrogen and short-term retention. In this model, the dead inventory has a saturation D_i^{sat} as shown in Table II. Radioactive decay of tritium with the decay constant $\lambda = 1.79 \times 10^{-9}$ s^{-1} are considered in the model. Leakage rate from the components in the main flow loop is assumed to be same as the decay constant λ. It should be noted that this loss will retain in the system and some will be eventually recovered with farther long time constant. For instance, leaked tritium from primary loop will be captured in the secondary containment, and returned to the fuel loop through secondary tritium removal system such as Water Detritiation, and then Isotope Separation in order to prevent significant environmental emission or release within the plant. If the estimated tritium inventory in the loop or component such as vacuum vessel will continue to increase, it will likely be required to reduce it by regulation. In other word, this "loss term" including decay represents all these longer term dead inventory compared with the time constant of start-up scenario analysis.

3. 3. Results and Discussion

3. 3. 1. TBR_{DT} and TBR_{DD} in 17Li83Pb blanket system

Table III summarizes the TBR calculation results obtained by MCNP. The estimated TBR_{DD} were 30% smaller than TBR_{DT} for the same blanket because D-D fusion reaction produce only 2.45 MeV neutrons and this energy is smaller than the threshold for multiplication reaction with lead. In order to compare, neutron flux and spectra in the LiPb breeder zone by D-T and D-D neutrons, are shown in figure 3-4. The "sn" in y-axis title means source neutron used by MCNP calculation. Also, 175 groups for neutron energies were used for the energy spectrum.

In the extended operation period, lithium-6 in the breeder will be burned and then TBR may change also. In order to simulate that situation, TBR was calculated with amount of burned and lost lithium-6. There are small decreases by quantity of residual lithium-6 after years of burn. Even if lithium-6 would be reduces by a half, only 12.5% of TBR decreases. Hence, lithium burn up in long period operation does not affect the TBR significantly when initial lithium-6 load is large.

As the net-TBR values with the coverage of 0.75 are less than unity, the following SD simulation uses the net-TBR values with coverage 0.95 (BOLD numbers at Net TBR column in Table III). These blanket coverage in this examination is the modified coverage which mentioned paragraph 3.1 already. Moreover, the maximum blanket coverage 0.95 was adopted in order to reduce the loss of tritium production performance. Evaluation of fusion reactor which has 0.95 of blanket coverage also proposed as a past fusion research [ref 5]. However, this blanket coverage may vary when high local TBR is improved with other blanket configuration.

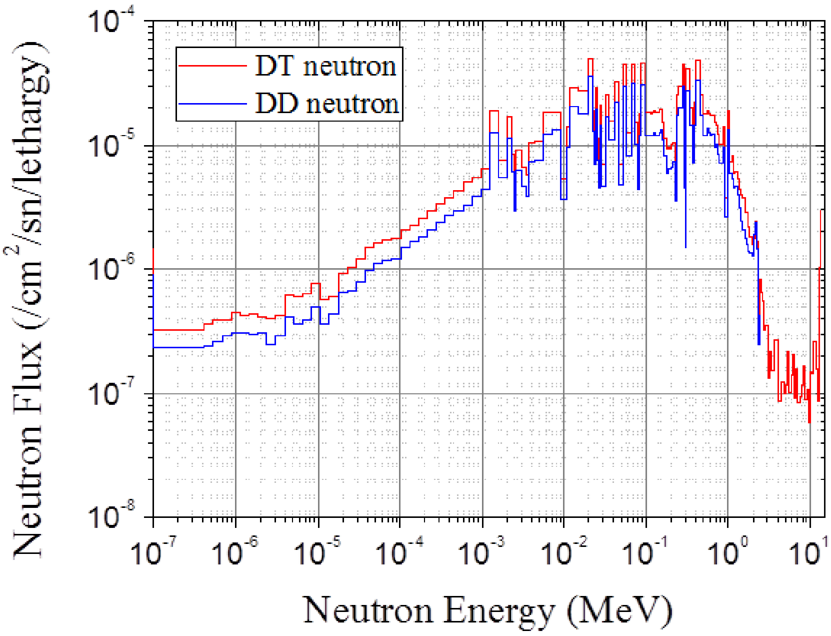

Fig. 3-4. Neutron spectrum by D-T and D-D neutrons at the backside surface of LiPb breeder zone

TABLE III. TBR with blanket coverage

Neutron (energy)	**Burned ^{6}Li**	**TBR (Tritium Breeding Ratio)**		
		Local	***Net (coverage 0.75)***	***Net (coverage 0.95)***
D-T neutrons (14.1 MeV)	100%	1.20	0.90	**1.14**
	80%	1.16	0.87	1.10
	50%	1.05	0.79	1.00
D-D neutrons (2.45 MeV)	100%	0.84	0.63	**0.80**
	80%	0.81	0.61	0.77
	50%	0.73	0.55	0.70

3. 3. 2. SD simulation of D-D start-up scenario with/without initial tritium

In order to evaluate the D-D start-up scenario, the required period was estimated for steady state operation (N_T / (N_D+N_T) = 0.5) with and without initial tritium using SD simulation software STELLA-10.0.2. Time evolution of the tritium concentration in the cases of initial tritium loading of 0, 60, 90, 120, 300 g were shown in figure 3-5. The required periods to reach steady state operation are estimated to be (a) 124 days without initial tritium, (b) 83 days with 60 g of tritium, (c) 15 days with 120 g of tritium, and (d) 10 hours with 300 g of tritium. These results show that launching D-T fusion projects without or small amount of initial tritium is possible. In addition, the tendency of tritium concentration decreases since tritium is consumed from few D-T reactions in D-D reactions by initial operation stage. The tendency reverses by increase of tritium production from reactions between D-T neutron in plasma and lithium in the blanket. D-D reaction becomes dominant reaction gradually, and D-T reaction becomes dominant reaction in plasma along with the operation period passing. After all, the tritium concentration in plasma increases exponentially.

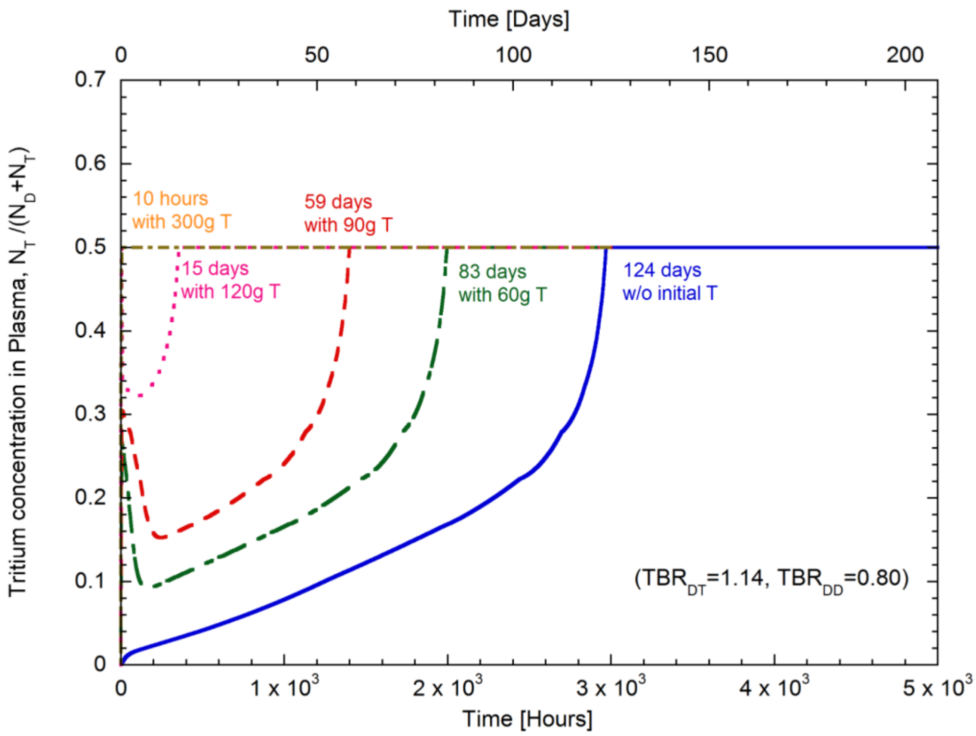

Fig. 3-5. Time-evolution of tritium concentration in plasma with and without initial loading of tritium

It should be noticed that the time arriving to the steady state operation depends on the TBR values as shown in figure 3-6. According to this result, D-D start-up scenario is attained D-T normal operation in the case of fusion reactor equipped with tight tritium production performance (round 1.0 of TBR) even though D-D operation period becomes long at the circumstance. It may be implemented with present local TBR blanket in fusion reactor which has 84% of realistic blanket coverage not idealized large blanket coverage (more than 0.90).

The extension of the period to reach steady state also strongly affected by the dead inventory. If it is excessive, required period will be much longer and may take years. However, it should be noted that extremely large tritium inventory in a single fusion plant will not be acceptable from safety reason.

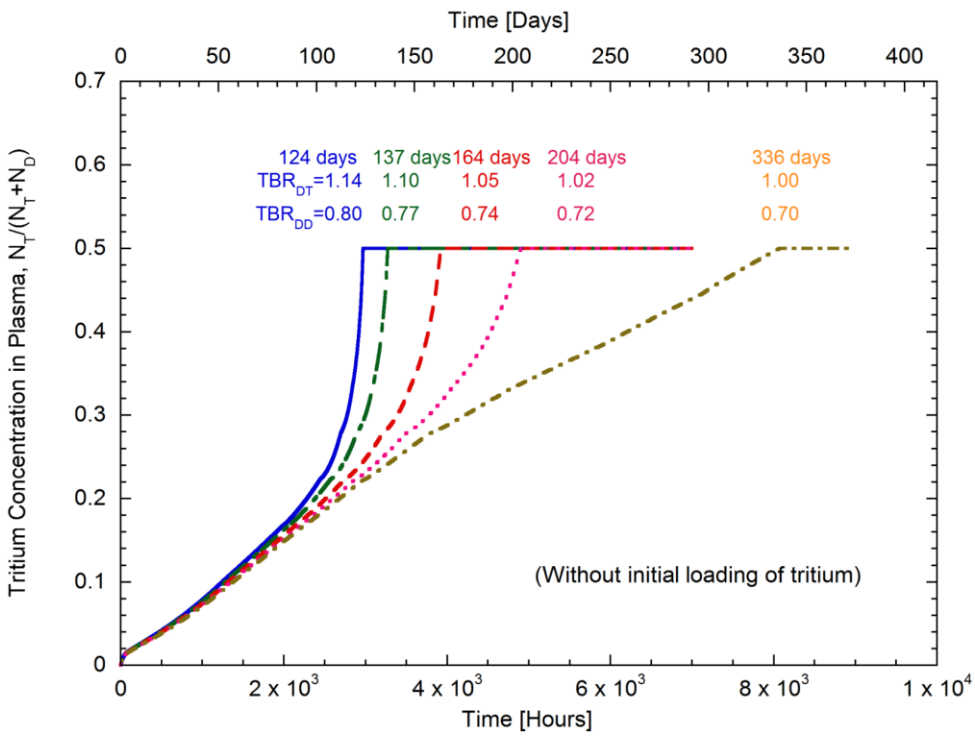

Fig. 3-6. Effect of TBR on the time-evolution of tritium concentration in plasma without initial tritium

Although it was expected that this scenario will be worked, there are some tasks to prepare a countermeasure: the superfluous tritium will be produced when operation mode reach steady state with TBR over unity as shown in figure 3-7. To store the tritium as an initial D-T fuel for next generation of D-T fusion plant can be considered as an alternative plan. However, if there is no construction plan to build another D-T fusion facility before the stored tritium would be excessive, it is necessary to control TBR in the blanket system and use up most of tritium.

There are two considerable options to control TBR: to use control rods in the blanket, and to control lithium-6 content in the liquid blanket, for instance by adding lead in replacement of lithium. In any case steady progress about requisite neutronics calculation for applying advanced scenario study is needed.

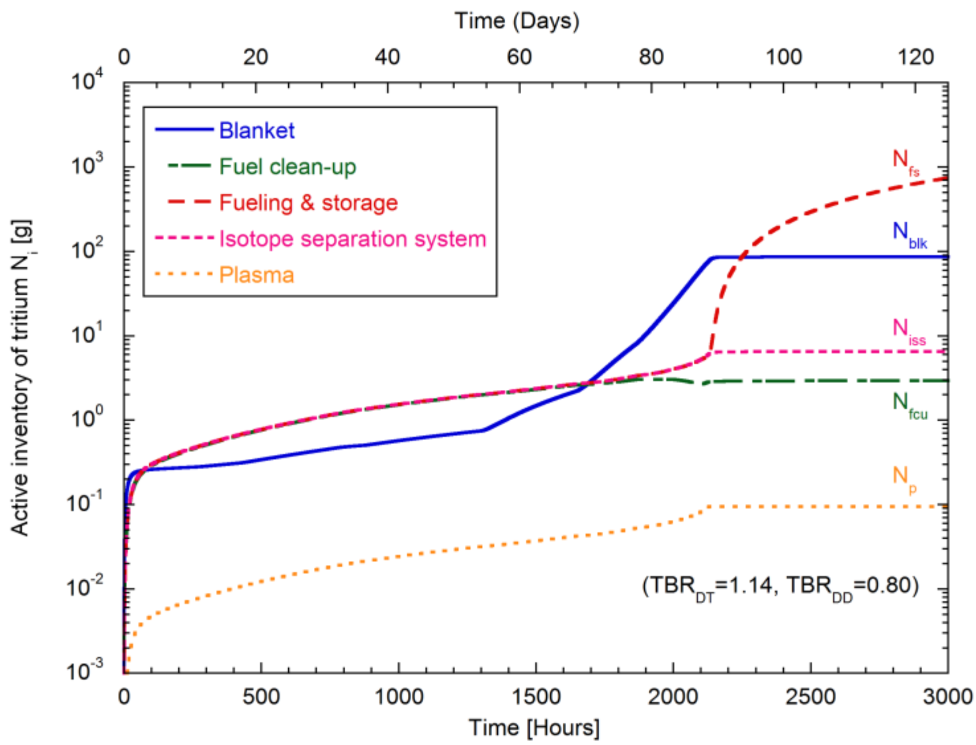

Fig. 3-7. Time-evolution of the active inventory in the components without initial tritium at $TBR_{DT} = 1.14$ and $TBR_{DD} = 0.80$

3. 4. Chapter Summary

Preparing an alternative way for procurement of tritium as fuel beyond ITER operation, improved fuel scenario was suggested including obtaining tritium by D-D reaction. Tritium behavior, in particular TBR in the LiPb blanket was evaluated with neutron transport code MCNP-5.14 using FENDL-2.1 data libraries. STELLA-10.0.2 was used for the tritium flow in a system of the scenario by time variation. Estimation of required period from D-D operation to D-T operation with and without initial tritium were obtained. The blanket with local TBR is 1.20 by D-T neutron, provides 0.84 by D-D neutron. Hence, the required extension period of D-D burning operation is estimated to (a) 124 days without initial tritium (b) 10 hours with 300 g of initial tritium. These results should be strongly dependent on the total effective TBR and dead inventory.

This scenario evaluation assumed large blanket coverage and low tritium loss rate for high performance of the scenario result. Although the conditions of required TBR were clarified for materialization of the D-D start-up scenario, the elucidation of tritium loss rate and further examination will be required. The tritium loss rate which was assumed in this study is very small amount. The followings did not regard as the tritium loss: leaked from the fuel system (plasma to blanket), absorbed into materials of outer vessel, cooling pipes and structure. Since these are not released out of the reactor site, leaked tritium are collected and recovered after isotope separation process. Time constant variations of tritium were considered that are shorter than reactor operation period.

In any fusion plant, however, effective TBR should be greater than 1, and the total tritium inventory has a certain limit from safety reason. Therefore, it can be concluded that all the viable D-T reactors can be started up without initial tritium within a reasonable period of time. This result revealed that launching D-T fusion projects beyond ITER without initial tritium is possible with additional D-D operation, and it is likely to be needed for commissioning purpose. Also, the extension period is extremely shortened with small amount of initial tritium.

References

[1] S. Konishi et al., "Possible Scenrio to Start Up DT Fusion Plant without Initial Loading of Tritium," *J. Plasma Fusion Res.*, 76, 1309 (2000)

[2] Y. Asaoka et al., "Commissioning of a DT fusion reactor without external supply of tritium," 18th IAEA Fusion Energy Conf. PDP-08 (2000)

[3] X-5 MONTE CARLO TEAM, MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, LA-CP-03-0245, LA-CP-03-0284, Los Alamos National Laboratory (2003)

[4] D. L. Aldama et al., FENDL-2.1: Fusion Evaluated Nuclear Data Library for Fusion Applications, IAEA report INDC-NDS-467 (2004)

[5] Ralph W. Moir, "The Fusion Breeder," *J. Fusion Energy*, 2, 351 (1982)

4. Integral Experiment Design by D-D Neutrons

4. 0. Brief Description of Chapter 4

In order to verify the D-D start-up scenario for production initial tritium, neutronics analysis for integral experiment as an exploratory analysis method was conducted with MCNP-5.14 [ref 1] and FENDL-2.1 nuclear data libraries [ref 2]. The assumed verification experiment is a deuterium ion beam irradiation onto a deuteride target with LiPb blanket modules. Neutronics analyses are planned to be benchmarked with D-D reaction occurring at the surface of module coated with titanium deuteride (TiD_2) surface irradiated with D ion beam. The D ion beam experiment was estimated to be able to generate sufficient neutron to benchmark the calculation of tritium breeding performance. Hence, usability of the beam apparatus as a neutron generator is clarified by performance comparison with other fusion neutron sources. Furthermore, the validity and the necessity of D-D neutronics and its verification experiment reveal by the preliminary examination results obtained in neutron flux, reaction rate, and anticipated tritium production in experiment design of D-D neutron integral experiment.

4. 1. Introduction

Since there are very few D-T fusion neutron sources (14.1 MeV) and no fusion reactor yet, the use of D-D neutron source (2.45 MeV) grows increasingly important. In addition, the D-D neutrons are generable easily with beam apparatus. A D-D neutron generator and D-D plasma experiment equipment such as JT-60SA (Neutron production rate, 6.7×10^{16} n/s) and KSTAR (Neutron production rate, 4.7×10^{13} n/s) [ref 3, 4] can be used for benchmark of neutronics analyses of tritium breeding blanket module in the future. In particular, the D-D neutron integral experiment is indispensable to verify the D-D start up scenario described in previous chapter. Chapter 3 was revolved around the scenario and the scenario can be used for improved operation model of prospective fusion reactor [ref 5]. Therefore, the verifiable D-T or D-D neutron integral experiment on a laboratory level is indicated for fusion research to proceed with expedition. However, the D-D neutron benchmark experiment is required the evaluation of effect by energy spectrum difference between D-T and D-D neutrons. Specifically, the evaluation of contribution of neutron multiplier reaction, (n, 2n) reaction is important.

The purposes of this chapter are to clarify the feasibility study of D-D neutron integral experiment for verification the operation model and to benchmark the neutronics calculation with the existing or small scale facilities for integrated blanket module testing. Therefore, examination of capability as a neutron source which has neutron generation performance toward stability is needed. The performance means measurable tritium production after neutron irradiation into breeding blanket candidate materials including lithium, LiPb in this study. The evaluation of D-D neutrons can be used as a method of preparatory examination whether D-T operation will be possible or not for concern about a harmful damage by high energy neutron on devices such as a neutron detector.

The potential of the integral experiment using D-D neutron source can be described below. Some results of tritium production performance from chapter 2, SiC first wall with He coolant, and contents for discussion by proposing D-D neutron integral experiment are shown in figure 4-1.

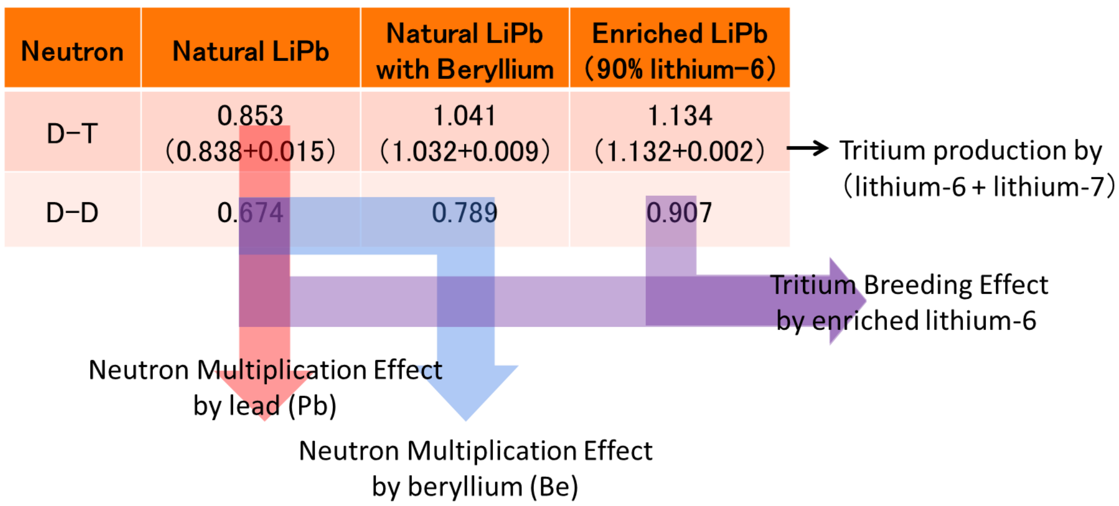

Fig. 4-1. Result of tritium production performance (SiC-He) with possible discussion contents by D-D neutron integral experiment

The relevant scientific issues to tritium production by neutrons and neutron behaviors are described as follow: (a) Tritium production by D-T neutrons involves reactions between the neutron and both of lithium isotopes, lithium-6 and lithium-7. (b) Neutron multiplication reaction by lead does not occur by D-D neutrons (high threshold energy of lead, around 7

MeV). (c) Neutron multiplication reaction by beryllium may occur by D-D neutrons sufficiently (around 1.75 MeV). (d) There is minuscule tritium production from lithium-7 in highly enriched LiPb. Figure 4-2 shows threshold reaction cross sections of lead (^{208}Pb, isotopes of lead which is largest natural abundance, 52.4 %) and beryllium (^{9}Be) [ref 6].

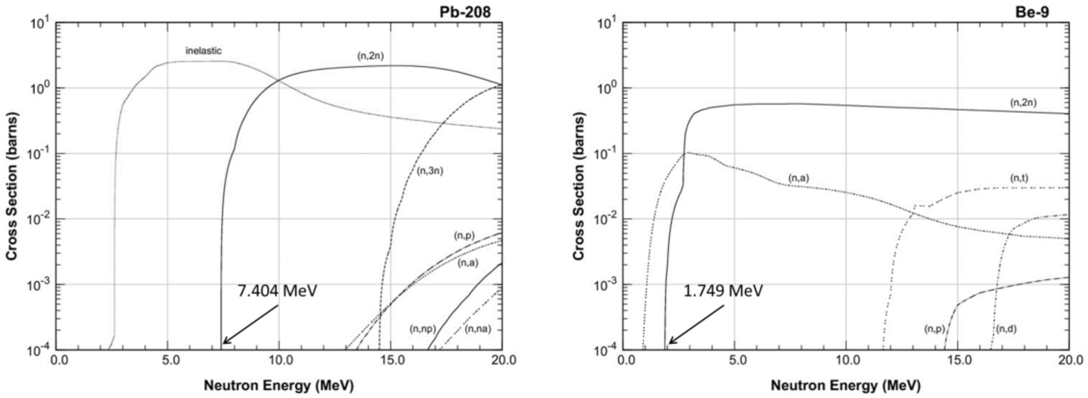

Fig. 4-2. Threshold reaction cross sections of lead (left) and beryllium (right), JENDL-4.0

The D-D neutron integral experiment can be surveyed neutron multiplication by lead, by beryllium and tritium breeding effect by enriched lithium-6, based on the issues. Using D-D benchmark experiment can be regarded as a method to evaluate reliability of D-T neutronics in accordance with the verifications in this chapter.

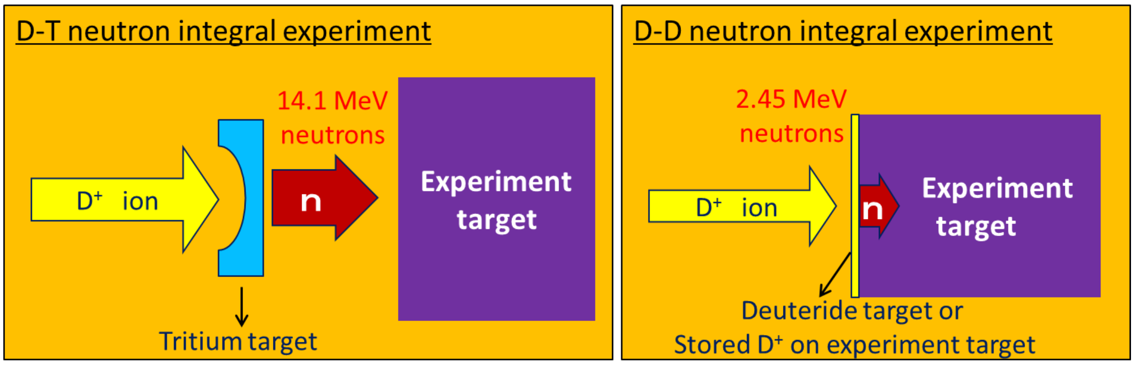

Fig. 4-3. Experimental aspect comparison of using D-T and D-D neutron source

Another potential by comparison with D-T neutron integral experiment is shown figure 4-3. The integral experiment using D-D neutrons has few burdens of neutron activation and tritium handling for existing experiment using D-T neutrons. In other words, there is no tritium in experiment room at the time of the experiment start, even though tritium is produced after the experiment. Tritium target manufacture has to be required from another tritium handling facility in the experiment using D-T neutrons also.

This chapter presents the experiment design of integral experiment using D-D neutron source and preliminary calculation results prior to the start the experiment. The experiment plan is expected to be applied for D-D neutron source development as well as integral experiment using by D-D neutrons for further study.

4. 2. D-D neutron benchmark experiment

4. 2. 1 Experiment design

The ion beam apparatus [ref 7] and its vacuum chamber are shown in figure 4-4. This experiment plan is currently underway in two parts as follows: Part A is ion beam generation apparatus using deuterium (D) ion, Part B is integral experiment using generated neutrons from ion beam part. At part A, existing hydrogen (H/D) ion beam apparatus is renovated. With 25 kV, 6 A steady D ion beam irradiating blanket module target with D containing surface can generate D-D neutron. At part B, generated D-D neutrons react with behind of metal target including tritium breeder material after that measurable tritium will be produced.

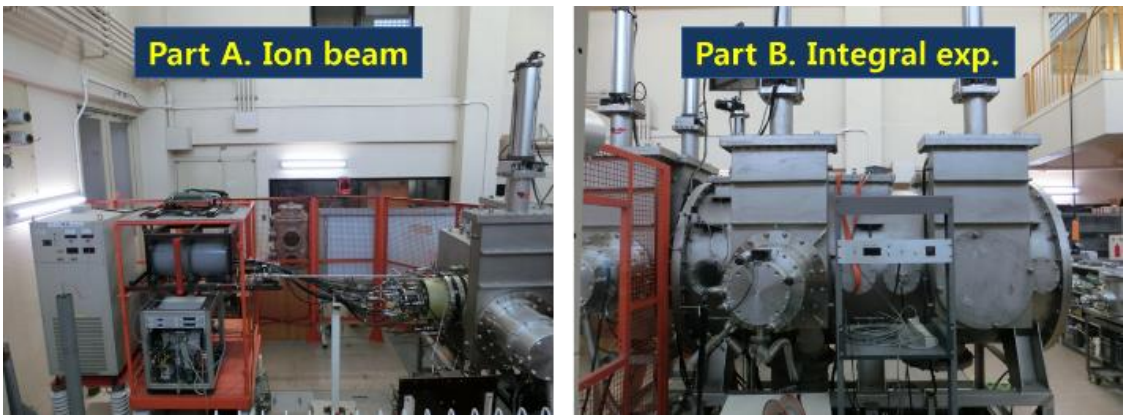

Fig. 4-4. Actual photo of ion beam apparatus

4. 2. 2 Estimation of expected neutron yield

A part of example of experimental result at the FNS facility as a procedure to obtain the neutron yield for this experiment design is shown below. It is a mock-up experiment and clarified the influences on tritium production by SS316 and beryllium as a reflector in blanket module. Figure 4-5 shows (a) a schematic view of the experimental configuration, (b) calculated and measured TPR distribution in the Li_2TiO_3 layer and irradiation time [ref 8]. Neutron yield in this proposing experiment with small LiPb block ($10\times10\times10$ cm^3) was estimated by 1.8×10^{-7} /cc of tritium production rate (TPR) on lithium-6 (TPR_6, /lithium-6 atom/source neutron) obtained in neutronics analysis and possible detection limit by Liquid Scintillation Counter (LSC, measurable tritium limit>1 Bq/cc) as the preliminary calculation. The performance goal of neutron yield for measurable is required more than 10^{10} n/s (full operation for 2 days, 14 h/day).

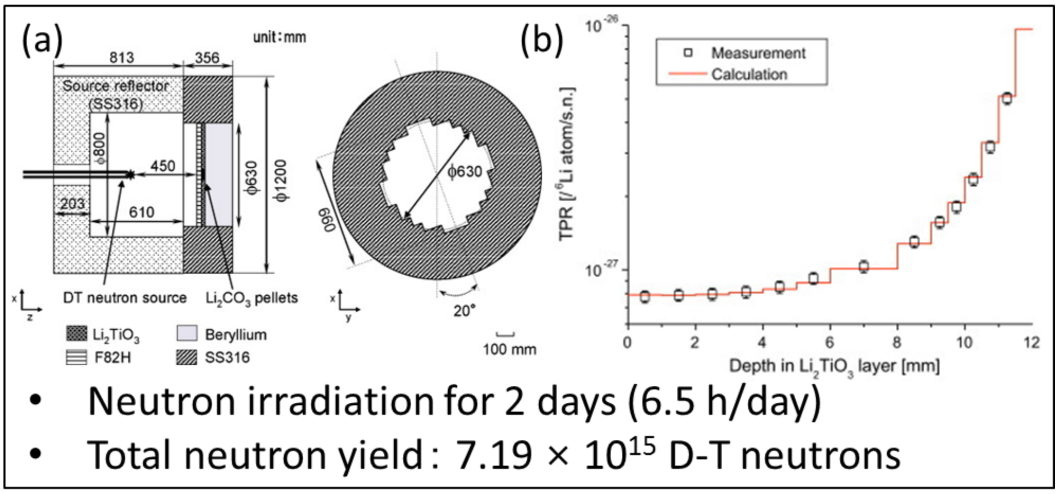

Fig. 4-5. Example of integral experiment using D-T neutrons [ref 7]

The neutron production yield expected with a simple assumption is estimated before forming the experiment scheme. The used assumption of scheme is shown in figure 4-6.

There is TiD_2 thin film as a target in the vacuum vessel. Its gram density is 4.0 g/cm^3 and atomic weight is 51.9 g/mol. Also, assumed beam current of deuterium ion beam is 4.24

A and the voltage is 25 kV. The beam operation can be kept about 20 s. The cross section of D (d, n), 0.605 mb is used for this estimation. The thickness of the target is designed 0.3 μm by the SRIM (formerly TRIM) simulation [ref 9]. Estimated neutron yield is 4.455×10^{10} n/s. According to this neutron yield, sufficient neutron can be generated for being measurable tritium production through this experiment.

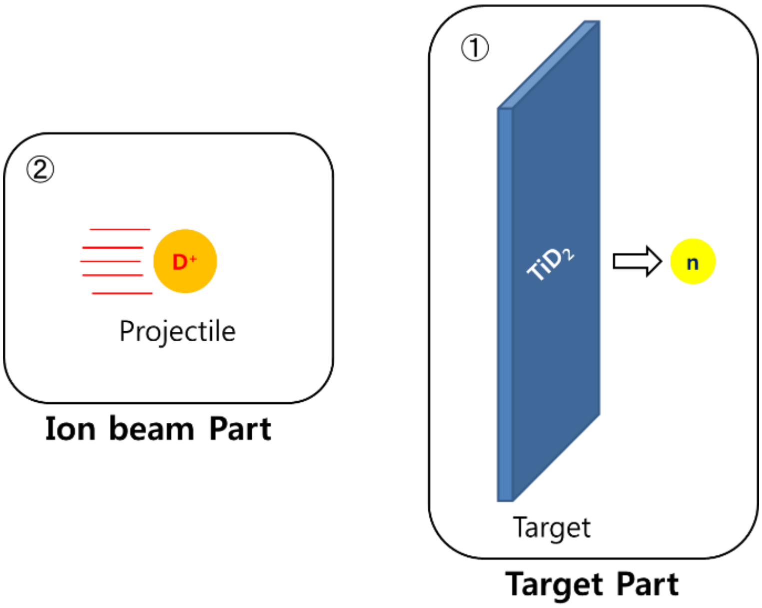

Fig. 4-6. Scheme of deuterium ion and TiD_2 target for estimating neutron yield

4. 2. 3 Reaction rates and neutron flux

In designed integral experiment, neutron spectra and reaction rates of the ^{197}Au(n, g)^{198c}Au and ^{93}Nb(n, 2n)^{92m}Nb were calculated by MCNP-5.14. These results can be used to compare with experimental data for calibration. (a), (b) in the figure 4-7 are different cross sectional view of the vacuum vessel prepared by MCNP. Details of assumed LiPb assembly with SiC and F82H panel are shown (c) in figure 4-7.

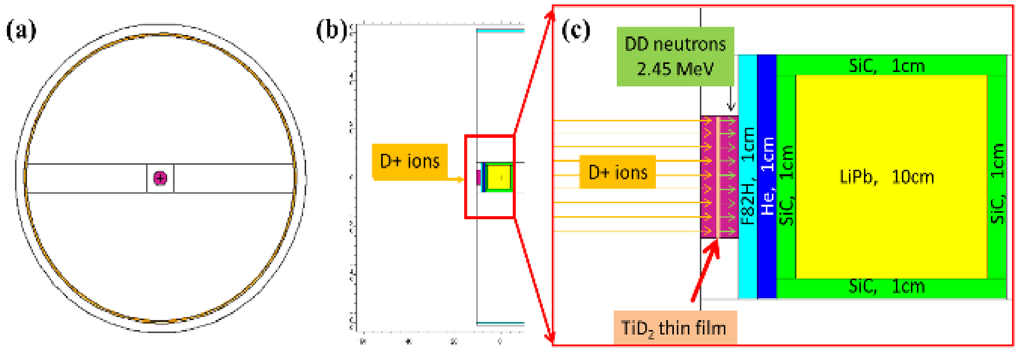

Fig. 4-7. MCNP model of LiPb assembly in vacuum vessel

The reaction rate and neutron flux spectrum were calculated with D-T and D-D neutrons respectively. Two different dosimetry reaction rates, ^{93}Nb(n, 2n)^{92m}Nb and ^{197}Au(n, g)^{198c}Au, are used to measure the neutron flux, from low energy neutron to fusion energy neutron. Most fusion benchmark experiments with neutron generator adopted those reactions as the activation foil method [ref 10-13]. The reaction rates of ^{93}Nb(n, 2n)^{92m}Nb and ^{197}Au(n, g)^{198c}Au decrease with the depth as shown in figure 4-8. Although, Nb(n, 2n) reaction has a high threshold energy and cannot be used with a D-D neutron experiment, this is for a further research plan that use the apparatus as a D-T neutron source with some modifications in the future study.

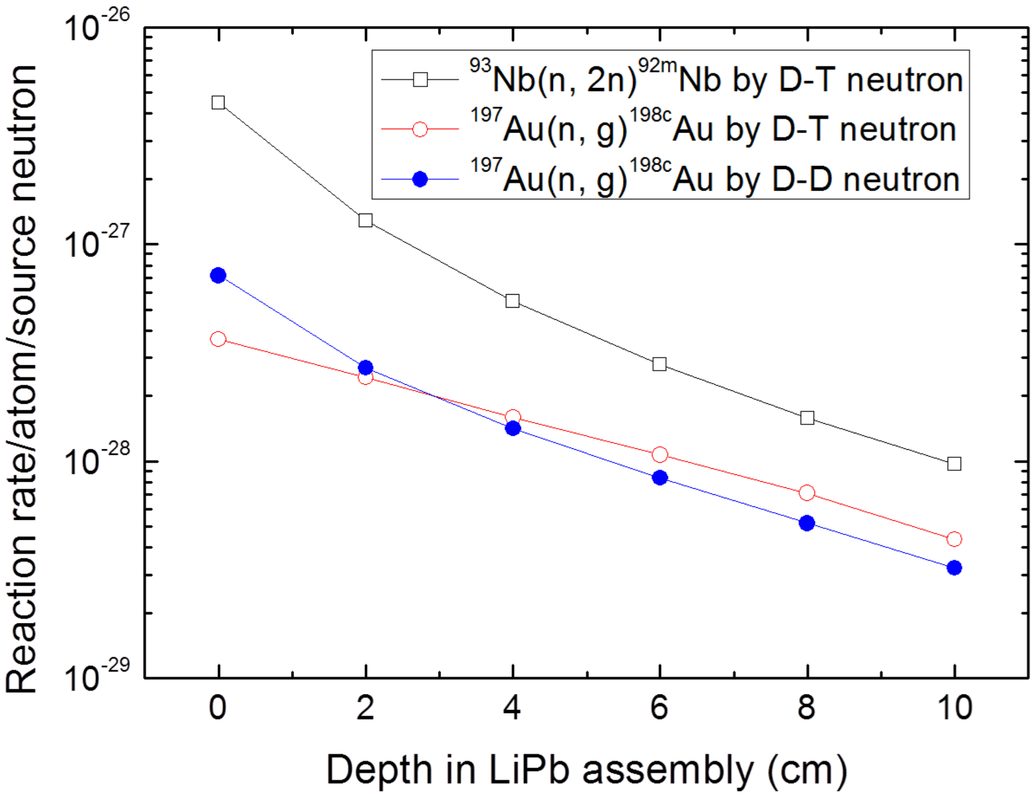

Fig. 4-8. Estimated Nb and Au reaction rates by MCNP in the LiPb assembly

Also, calculated neutron flux spectrum at 4 cm and 6 cm depth in the LiPb assembly between 0.01 MeV and 20 MeV of energy range with FENDL-2.1 is plotted figure 4-9. According to figure 4-9, it turns out that the evaluation of tritium production is possible to perform sufficiently by this D-D neutron integral experiment, since both of neutrons have similar neutron flux from 10 keV to 1 MeV energy region. These numerical results not only show the behavior of fusion neutron in the blanket, but also to be used for the preparation and planning of experiment, and comparison with the experiments.

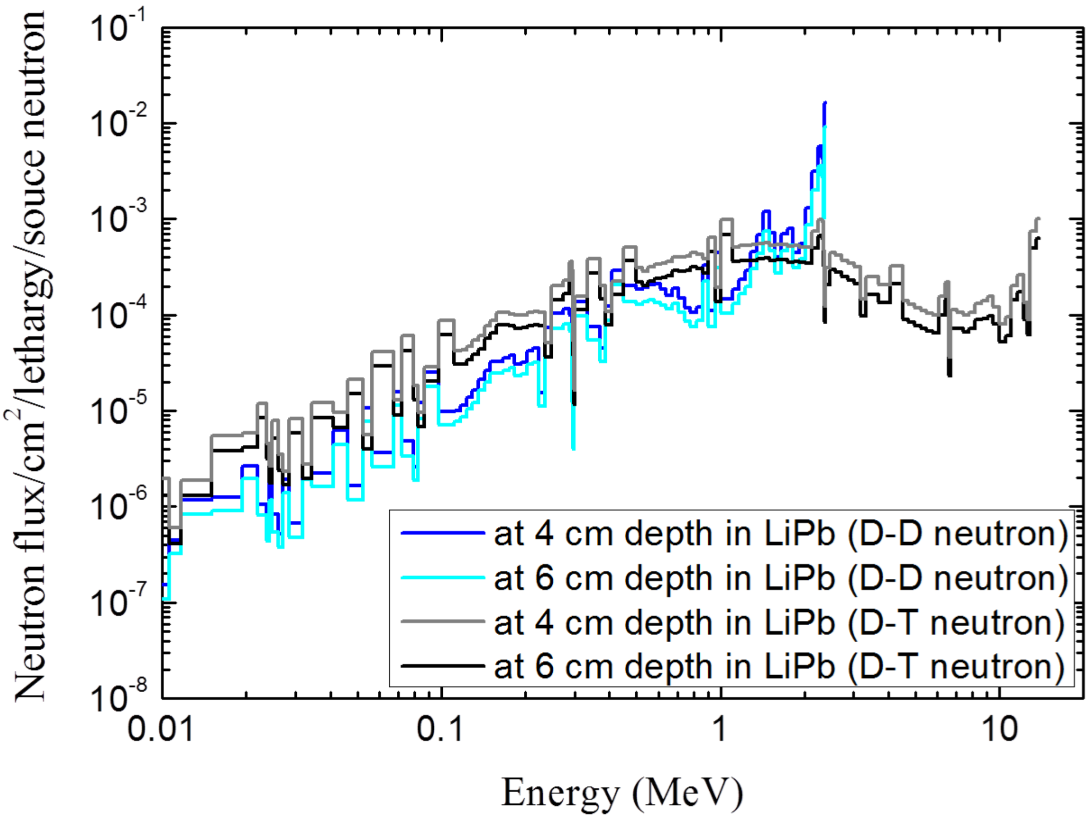

Fig. 4-9. Calculated neutron flux spectrum

4. 2. 4. Expected Measurable Tritium by D-D Neutrons

As the result of produced tritium by neutronics analysis, 1.8×10^{-7} tritium per a neutron per cm^3 were produced in LiPb assembly. More than 1 Bq/cc of tritium can be obtained with about 19 hour-operation (1.5 days, 6.5 h per day) of the proposed D-D neutron source. Since the tritium production is average value per total LiPb volume ($10\times10\times10$ cm^3), the duration of this experiment can reduce by location selection of measurement for high concentration tritium expected. In other words, even if there is no D-T fusion neutron source such as FNS and FNG, D-D neutron integral experiment with the proposed device has indicated the potential to be utilized for fusion blanket research. The designed experiment is adoptable to other preliminary integral experiments such as benchmark of other breeding materials. And then, a volume neutron source is needed for large blanket module test. The significance of other D-D plasma experiment equipment such as JT-60SA and KSTAR is suggested also before ITER-TBM test.

4. 2. 5. Considerable Limitation of Target Size by Heating

The TiD_2 thin film target as shown in (c) of figure 4-7 at section 4. 2. 3 may overheat by ion beam irradiation, and the target receives damage. In addition, neutron energy will be lost using water coolant because of its property as a neutron moderator in particular except using helium coolant. In such the water cooled case, the experiment using exact D-D neutrons cannot be carried out obviously. Hence, the size condition of titanium target without cooling was estimated by simple thermal calculation of heat quantity. Figure 4-10 shows considerable internal configuration with cooling heat.

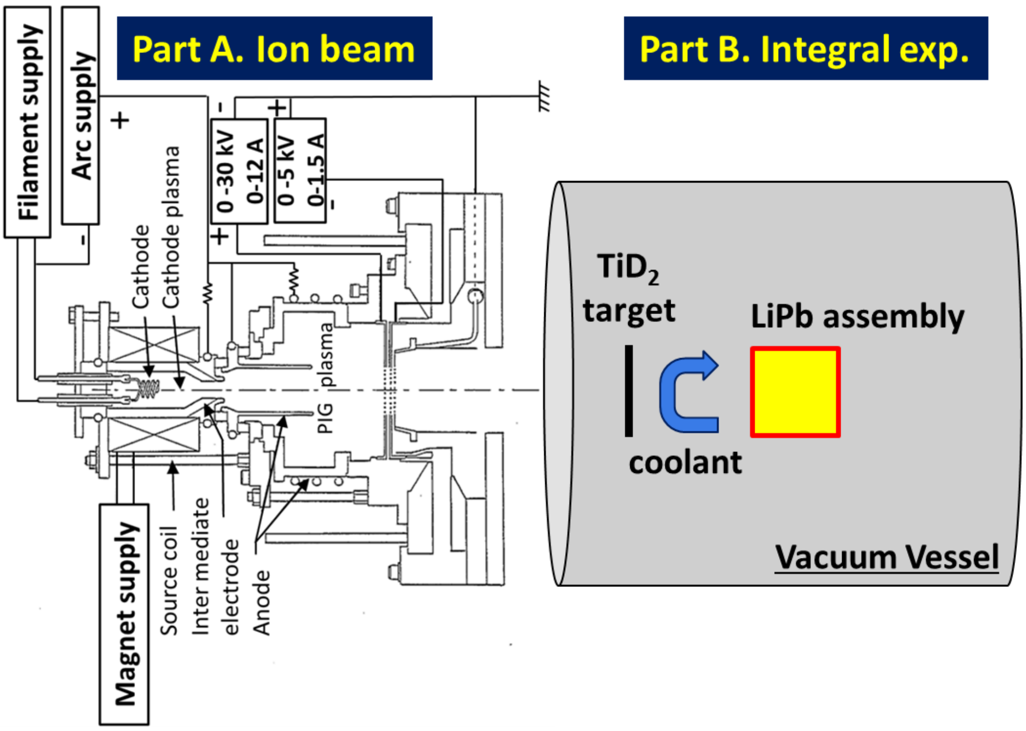

Fig. 4-10. Considerable internal configuration with cooling heat

It is necessary to design the titanium target which is larger than 18 cm^3 of total volume (80 g weight) with pulsed beam, it was estimated by melting point of titanium (1660 °C), specific heat(0.54 kJ/kg·K), beam current and beam energy. However, neutron behavior will be affected by target thickness because of neutron absorption into target material. When thin deuteride target is assumed, it is reasonable way to cool with materials of small cross section of neutron reaction.

4. 3. Chapter Summary

This chapter proposed the design of D-D benchmark experiment for the integral experiment using deuterium ion beam equipment. About 10^{10} n/s of neutrons can be generated on current device performance by numerical estimation result. As the result with calculated TBR of the LiPb assembly, tritium can be measured sufficiently with realistic experiment time (device operation time). The D-D neutron integral experiment can be carried out by the proposed experiment design. The preliminary calculation prior to the experiment suggested the feasibility of small scale integral experiment or mock-up experiment for fusion blanket, and the results such as neutron spectra will be utilized as comparative calculation data with experimental data obtained in the designed experiment. D-D neutronics and its experiment will also be useful for the initial commissioning of fusion plant.

References

[1] X-5 MONTE CARLO TEAM, *MCNP – A General Monte Carlo N-Particle Transport Code*, Version 5, LA-UR-03-1987, LA-CP-03-0245, LA-CP-03-0284, Los Alamos National Laboratory (2003)

[2] D. L. Aldama et al., *FENDL-2.1: Fusion Evaluated Nuclear Data Library for Fusion Applications*, IAEA report INDC-NDS-467 (2004)

[3] S. Kwon et al., "Operation Scenario of DT Fusion Plant Without External Initial Tritium," *25th SOFE Proceedings* (2013)

[4] JT-60SA Research Unit, *JT-60SA Research Plan – Research Objectives and Strategy*, Version 3.1 (2013)

[5] M. S. Choen et al., "Diagnostic neutron activation system for KSTAR," *J. Instrumentation*, 7.05, C05009 (2012)

[6] K. Shibata et al., "JENDL-4.0: A New Library for Nuclear Science and Engineering," *J. Nucl. Sci. Technol.* 48(1), 1 (2011)

[7] S. Matsuda et al., "Constitution of the Injector Test Stand (ITS-1) and the Preliminary Results with Model – I Ion Source," JAERI-M 6431 (1976) [in Japanese]

[8] K. Kondo et al., "Investigation on the TPR prediction accuracy in blanket neutronics experiments with reflector at JAEA/FNS," *Fusion Eng. Des.*, 86, 2184 (2011)

[9] J.F. Ziegler et al., *SRIM – The Stopping Range of Ions in Matter, Ion Implantation Press* (2008)

[10] F. Maegawa et al., "Benchmark experiment on silicon carbide with D-T neutrons and validation of nuclear data libraries," *Fusion Eng. Des.*, 58-59, 595 (2001)

[11] Y. Chen et al., "Sensitivity and uncertainty analyses of 14 MeV neutron benchmark experiment on silicon carbide," *Fusion Eng. Des.*, 69, 437 (2003)

[12] I. Kodeli, "Cross-section sensitivity analyses of 14 MeV Neutron Benchmark Experiment on Tungsten," *J. Nucl. Mater.*, 329-333, 717 (2004)

[13] P. Batistoni et al., "Neutronics benchmark experiment on tungsten," *J. Nucl. Mater.*, 329-333, 686 (2004)

5. Conclusion

This study aimed at clarifying to supply the indispensable fusion fuel, tritium in particular, in order to materialize nuclear fusion energy as a future energy resource. The signification of this study is comprehensive fusion blanket research including the method to reduce lithium-6 in breeding blanket, the operation scenario for tritium self-sufficiency and its verification using existing device for clarifying feasibility of DEMO reactor.

In this study, three evaluation methods were used to clarify the fusion fuel limitations in order to achieve DEMO reactor as follows: neutronics analyses including LiPb blanket design for conviction of its tritium self-sufficiency with natural lithium breeder in chapter 2, numerical analyses of fuel supplement scenario to secure initial loading tritium with neutronics results of LiPb blanket performance including tritium bred in chapter 3, and an empirical study including small scale D-D neutron integral experiment design for the supplying scenario in chapter 4.

5. 1. Chapter Summary: Chapter 2. Breeding Blanket Design with High Tritium Production Performance

In chapter 2, tritium production/breeding performance was evaluated with LiPb blanket inner structure to clarify the influence on TBR including combination of FW and coolant material. Natural LiPb blanket design with beryllium arrangement was also examined for lithium-6 resource problem to solve. As the results, even though enrichment of lithium isotopes was not carried out commercially in the world, it is not an essential restriction of fusion energy in practical use. The clarified results are shown below.

→ Backside reflector in LiPb blanket is effective for increase tritium production performance.

→ The combination of SiC-helium is most effective way to raise tritium production performance in LiPb blanket.

→ Natural LiPb blanket including beryllium region has sufficient possibility as an attractive breeder for further conceptual liquid blanket design without lithium-6 enrichment.

→ Most valid ratio of LiPb to beryllium is 1:3.

→ It is effective to arrange beryllium on radial direction like a structure material in the blanket module.

→ Neutron shielding region should be strongly required multi-layer structure including a neutron moderator and absorber which has large neutron capture cross section.

5. 2. Chapter Summary: Chapter 3. Fuel Supply Scenario with less or without Initial Loading Tritium

In chapter 3, TBR in LiPb blanket was calculated using Monte Carlo neutron transport code. Using the TBR values (already analyzed and evaluated in chapter 2, TBR_{DT} by DT neutron and TBR_{DD} by D-D neutron), the D-D start-up scenario by system dynamics (SD) simulation was evaluated. Furthermore, it was shown that the variation of D-D start-up period by assumption of tritium inventory in the fusion plant with the operation process, by shortening effect with small amounts of initial loading tritium and by dependence of TBR. As the results, it is found out that it is not an essential restriction of fusion energy without existing initial loading tritium globally. The clarified results are shown below.

→ The required periods to reach steady state operation were estimated to be (a) 124 days without initial tritium, (b) 83 days with 60 g of tritium, (c) 15 days with 120 g of tritium, and (d) 10 hours with 300 g of tritium.

→ Launching D-T fusion projects without or small amount of initial tritium is possible according to the results.

→ It became clear that the scenario can be used as a solution of initial loading tritium secured problem which may arise beyond ITER operation to achieve DEMO reactor

5. 3. Chapter Summary: Chapter 4. Research Approach with D-D Neutronics and its Merit on Fusion Reactor

In chapter 4, the required D-D neutron integral experiment design was proposed for examination the D-D start-up scenario, was discussed its feasibility. Preliminary calculation results were summarized for prior to the start the experiment. The experiment plan can be applied for D-D neutron generator development as well as integral experiment using by D-D neutrons for further study. Moreover, it is shown that tritium breeding performance of blanket module can be validated by existing and near-future D-D plasma experiment equipment are used as a volume neutron source. Hence, start-up scenario without initial loading tritium by D-D plasma operation was proposed. The clarified results are shown below.

→ About 10^{10} n/s of neutrons can be generated on current device performance by numerical estimation result. As the result with calculated TBR of the LiPb assembly, tritium can be measured sufficiently with realistic device operation time, about 70 sec.

→ Integral experiment at school or laboratory level can be carried out and achieved more quickly by the proposed experiment design.

5. 4. Overall Conclusion

A new energy source such as fusion energy cannot turn into base load supply of electricity in the world simply. However, fusion energy has a possibility as a sustainable and abundant energy source. The international fusion experiment, ITER is currently being constructed and several DEMO designs are being proposed in each country. Moreover, computer analyses and related experiments with neutron sources such as FNG and FNS are processing in various research groups. Even though everything seems to be going smoothly, securement of fusion fuel such as tritium and lithium-6 has to be solved for fusion energy as the new energy source. Experimental fusion reactor, ITER is under construction. However, there is no way to test and verify the tritium breeding performance of a fusion blanket before ITER-TBM test (from first plasma of ITER scheduled in 2027). At the same time, nuclear design and the actual proof experiment of fuelling using a neutron source are required for DEMO reactor. Then, this study is carried out about the fuel breeding blanket and the tritium fuel cycle including the examination on neutronics analyses using Monte Carlo code and on numerical analyses using system dynamics code in order to clarify tritium supplying method. This study evaluated the tritium production capability of a fusion blanket by neutron transport calculation, and considered its technical validity for feasibility of fusion energy, in particular, self-sufficiency of the fuel tritium. The LiPb blanket design with capability of self-sufficient tritium, the operation scenario for securement of initial loading tritium and the experimental methodology for its verification were addressed from the results in each chapter. As the results, it is shown that the commercial challenge of isotope separation technology of lithium-6 and the scarcity of initial loading tritium are not essential restrictions using nuclear fusion energy as a sustainable energy source, and its feasibility was revealed.

List of Publications

Evaluation of Tritium Production in LiPb Blanket System Using Neutrons Analysis

S. Kwon, K. Noborio, R. Kasada, S. Konishi

Zero-Carbon Energy Kyoto 2012, 239-244 (2012)

Neutronics of SiC-LiPb high temperature blanket for tritium production

S. Kwon, S. Sato, R. Kasada, S. Konishi

Fusion Science and Technology, 64, 599-603 (2013).

Operation scenario of DT fusion plant without external initial tritium

S. Kwon, R. Kasada, S. Konishi

IEEE 25th Symposium on Fusion Engineering (SOFE) Proceedings (2013).

List of Presentations

Study of ZnO thin film deposition and dependence to substrate temperature by ALD system

Saerom Kwon, Doohyoung Lee, Seungjeong Noh and Satoshi Konishi

The 3rd International symposium: Kyoto University Global COE program, Suwon, Korea, August 18-19, 2011

Neutronics analysis of SiC-LiPb high temperature blanket for tritium self-sufficiency

Saerom Kwon, Kenzo Ibano, Takahiro Maegawa, Kazuyuki Noborio, Ryuta Kasada and Satoshi Konishi

Asian core program, Beijing, China, January 16-17, 2012

Neutronics analysis of tritium production in a SiC-LiPb blanket concept for evaluation with integral experiment

Saerom Kwon, Kazuyuki Noborio, Ryuta Kasada and Satoshi Konishi

4th International symposium: Kyoto University Global COE program, Bangkok, Thailand, May 22-23, 2012

トリチウム生成のための SiC-LiPb 高温度ブランケットに関するニュートロニクス解析

権セロム、登尾一幸、笠田竜太、小西哲之

第 9 回核融合エネルギー連合講演会、神戸、2012 年 5 月 28 日-29 日

Neutronics calculation of a lithium-lead module for tritium production with evaluation of reflector effect

Saerom Kwon, Satoshi Sato, Ryuta Kasada and Satoshi Konishi

核融合工学部会夏の学校、青森、2012 年 8 月 19 日-22 日

Neutronics of SiC-LiPb high temperature blanket for tritium production

Saerom Kwon, Satoshi Sato, Ryuta Kasada and Satoshi Konishi

ANS 20th Topical Meeting on the Technology of Fusion Energy, Nashville, TN, August 27-31, 2012

LiPb 体系におけるトリチウム生成・増殖の評価

権セロム、佐藤聡、笠田竜太、小西哲之

日本原子力学会「2012 年秋の大会」、広島大学、2012 年 9 月 19 日-21 日

Neutronics of LiPb blanket with D-T and D-D neutrons

Saerom Kwon, Ryuta Kasada and Satoshi Konishi

地球温暖化時代のエネルギー科学拠点」平成 24 年度年次報告会、京都、2013 年 1 月 16 日

リチウム鉛ブランケットにおけるニュートロニクスと D-D 中性子積分実験による評価検討

権セロム、佐藤聡、笠田竜太、小西哲之

日本原子力学会「2013 年春の年会」、近畿大学東大阪キャンパス、2013 年 3 月 26 日-28 日

Operation Scenario of DT Fusion Plant without External Initial Tritium

Saerom Kwon, Ryuta Kasada and Satoshi Konishi

25th Symposium on Fusion Engineering, San Francisco, California, June 10-14, 2013

リブ構造を有するブランケットモジュールの TBR 評価

権セロム、笠田竜太、小西哲之、佐藤聡、谷川尚、落合謙太郎

日本原子力学会「2013 年秋の大会」、八戸工業大学、2013 年 9 月 3 日-5 日

Neutronics of LiPb blanket and design and evaluation of integral experiment with D-D neutrons

Saerom Kwon, Satoshi Sato, Ryuta Kasada and Satoshi Konishi

11th International Symposium on Fusion Nuclear Technology, Barcelona, Spain, September 16-20, 2013

システムダイナミックスによる核融合炉におけるトリチウムフローに関する研究、

笠田竜太、Kwon Saerom、小西哲之

プラズマ・核融合学会　第 30 回年回、東京工業大学　大岡山キャンパス、2013 年 12 月 3 日-6 日

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