NEUTRONIC INVESTIGATION OF FISSILE MATERIAL PRODUCTION FROM MOLTEN SALT FUEL MIXTURE IN A FUSION REACTOR BLANKET

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I certify that in my opinion the thesis submitted by Alper KARAKOÇ titled "NEUTRONIC INVESTIGATION OF FISSILE MATERIAL PRODUCTION FROM MOLTEN SALT FUEL MIXTURE IN A FUSION REACTOR BLANKET" is fully adequate in scope and in quality as a thesis for the degree of PhD.

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““I declare that all the information within this thesis has been gathered and presented in accordance with academic regulations and ethical principles and I have according to the requirements of these regulations and principles cited all those which do not originate in this work as well.”

Alper KARAKOÇ

ABSTRACT

Ph. D. Thesis

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In this thesis, a new approach has been investigated in a Fusion-Fission Hybrid Reactor (FFHR) for the ITER reference geometry, using a thorium and uranium molten salt mixture as a dual-purpose medium for coolant and fissile fuel production. The study highlighted the broader benefits of the thorium ve uranium fuel cycle, its safety features and reduced nuclear waste production. In this study, SS 316 LN-IG was selected as the first wall material of the reactor and LiF-ThF_4 and LiF-UF_4 molten salt fuel mixture was used as the coolant in two different rmodels, considering the eutectic points of the material. For neutronic analyses, the MCNP5 nuclear code was used together with the ENDF/B-VIII and CLAW-IV nuclear data libraries. The evolution of isotopes in the reactor over time was calculated with the MCNPAS interface code. The study results were evaluated in terms of tritium production rate, energy multiplication factor, radiation damage, fissile fuel production and fuel combustion value. A 4-year study history of the total TBR value was calculated and is always above 1.05 and increases over time. In the reactor model using LiF-ThF_4 molten salt and fuel mixture, the thorium mass decreased from 631.3 tons at the beginning to 587.2 tons, while ^{233}U production during this period was 9.1 tons. According to these results, the first wall replacement period was calculated as 3.94 years. In the reactor model using LiF-UF_4 molten salt and fuel mixture, the uranium mass decreased from 720.8 tons at the beginning to 639.6 tons, while ^{239}Pu production during this period was 21.3 tons. According to these results, the first wall replacement period was calculated as 3.92 years.

**Key Word : ** FFHR, ITER, TOKAMAK, Thorium, Uranium, Molten salt mixture
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ÖZET

BİR FÜZYON REAKTÖRÜNÜN MANTO YAPISINDAKİ ERGİMİŞ TUZ YAKIT KARIŞIMINDAN FİSİL MATERYAL ÜRETİMİNİN NÖTRONİK İNCELENMESİ

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Bu çalışmada, ITER referans geometrisi için bir Füzyon-Fisyon Hibrit Reaktöründe (FFHR), soğutucu ve fisil yakıt üretimi için çift amaçlı bir ortam olarak toryum ve uranyum erimiş tuz karışımı kullanılmasıyla yeni bir yaklaşım araştırılmıştır. Çalışma, toryum ve uranyum yakıt döngüsünün daha geniş faydalarını, güvenlik özelliklerini ve azaltılmış nükleer atık üretimini vurgulamıştır. Bu çalışmada, reaktörün ilk duvar malzemesi olarak SS 316 LN-IG seçilmiş ve malzemenin ötektik noktaları dikkate alınarak soğutucu olarak LiF-ThF$_4$ ve LiF-UF$_4$ ergimiş tuz yakıt karışımları ayrı ayrı iki farklı modelde kullanılmıştır. Nötronik analizler için MCNP5 nükleer kodu, ENDF/B-VIII ve CLAW-IV nükleer veri kütüphaneleri ile birlikte kullanılmıştır. Reaktördeki izotopların zaman içindeki evrimi MCNPAS arayüz kodu ile hesaplanmıştır. Çalışma sonuçları, trityum üretim oranı, enerji çoğaltma faktörü, radyasyon hasarı, fisil yakıt üretimi ve yakıt yanma değeri açısından değerlendirilmiştir. Toplam TBR değerinin 4 yıllık çalışma geçmişi hesaplanmış ve her zaman 1,05'in üzerinde olup zamanla artmaktadır. LiF-ThF$_4$ ergimiş tuz ve yakıt

karışımı kullanılan reaktör modelinde toryum kütlesi başlangıçta 631,3ton değerinden 587,2 tona düşerken, bu dönemde ^{233}U üretimi 9,1ton olmuştur. Bu sonuçlara göre ilk duvar değiştirme süresi 3,94 yıl olarak hesaplanmıştır. LiF-UF_4 ergimiş tuz ve yakıt karışımı kullanılan reaktör modelinde uranyum kütlesi başlangıçta 720,8ton değerinden 639,6 tona düşerken, bu dönemde ^{239}Pu üretimi 21,3ton olmuştur. Bu sonuçlara göre ilk duvar değiştirme süresi 3,92 yıl olarak hesaplanmıştır.

Anahtar Sözcükler : FFHR, ITER, TOKAMAK, Toryum, Uranyum, Ergimiş tuz karışımı

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SYMBOLS AND ABBREVIATIONS INDEX

SYMBOLS

n	: neuton
β^-	: beta rays
p	: proton

ABBREVIATIONS

EBR	: Experimental Breeder Reactor
DPA	: Displacement Per Atom
FDS-EM	: Fusion Driver System-Energy Multiplier
BWR	: Boiling Water Reactor
PWR	: Pressurized Water Reactor
CANDU	: Canada Deuterium Uranium
GCR	: Gas Cooled Reactor
IFE	: Inertial Fusion Energy
TBR	: Tritium Breeding Ratio
ITER	: International Thermonuclear Experimental Reactor
ENDF	: Evaluated Nuclear Data File
FFH	: Fission Fusion Hybrid
CLAW	: Activity Cross Section Data Library
TOKAMAK	: Toroidalnaya Kamera and Magnitnaya Katushka

CHAPTER 1

INTRODUCTION

1.1. BACKGROUND

Nowadays, the criterion of sustainable technological development and meeting humanity's needs is energy and energy consumption. Energy consumption, a criterion of the mentioned developments and needs, is constantly increasing due to the increasing population. Scientists are working on existing and alternative energy sources to meet this energy demand. When the existing energy resources are evaluated, it is observed that nuclear energy is the most important energy source that offers sustainable and environmentally friendly energy production. A significant part of the studies conducted in the last century has been focusing on this subject. Ernest Rutherford, a scientist, discovered in 1932 that, in accordance with the mass-energy equivalence principle, a tremendous quantity of energy is produced when protons split lithium atoms in a proton accelerator. Rutherford and fellow nuclear physics pioneers Niels Bohr and Albert Einstein, nevertheless, thought it was improbable that the power of the atom would be used for useful purposes any time soon. James Chadwick, a Ph.D. student at Rutherford's lab, made the neutron discovery in the same year [1]. In 1934, Frédéric and Irène Joliot-Curie discovered induced radioactivity due to neutron bombardment experiments, which led to the discovery of elements similar to radium [2]. In 1934, physicist Enrico Fermi's experiments in Rome showed that many atoms could split by a neutron. Two important results were obtained in the experiments carried out. The first of these discoveries was the discovery of a new element called Hesperium by neutron bombardment of the uranium atom, which is now known in the literature as plutonium. Another important result is that the total mass of waste materials obtained from the bombardment of the uranium atom was slightly lighter than uranium [3]. Niels Bohr and Otto R. Frisch also reached similar results with their experiments. They found that the mass of barium and other elements obtained from neutron bombardment of uranium is not equal to the mass of the initial uranium. Lise Meitner, on the other hand, related Einstein's theory of the lost mass in question and showed that the lost mass is converted into energy [4]. In 1939, in interviews conducted by Bohr, Einstein, and Fermi, the potential of atoms to release huge amounts of energy through sustainable chain reactions was estimated. After these evaluations and conferences, it has come to the fore that humanity's energy needs can be met with chain reactions. According to the findings, if sufficient amounts of uranium are brought together under appropriate conditions, the chain reaction can start. A team of researchers assembled at the University of Chicago in the early months of 1942 to advance their theory. The Chicago Pile-1, the first nuclear reactor ever built, was under construction by November 1942. The reactor also had uranium, graphite, and cadmium control rods. When the rods were inside the reactor core, it was observed that the number and energy of neutrons decreased. When the control rods were gradually withdrawn from the reactor, the nuclear reaction in the core reached a self-sustaining level, and the information obtained in theory was reflected in practice. Fermi and his team had thus successfully transformed their theory into a technological reality. As a result, the nuclear era has begun.

From the time when nuclear energy was first discovered to the present day, all of the power plants built in the process have been powered by fission reactions. If the history of fission reactors is examined, electrical energy was obtained for the first time from the heat in the reactor at the EBR-1 (Experimental Breeder Reactor) on December 20, 1951. On top of that EBR-1 was also the first FBR (Fast Breeder Reactor) type reactor [5]. One of the most important developments in this field is the Obnisk RMBK (Reaktor Bolshoy Moshchnosti Kanalniy/High-Power Kanal-Type Reactor) type nuclear power plant was connected to the power grid for the first time and built in the Soviet Union [6-7]. In 1955, BORAX-III, the first of its kind BWR (Boiling Water Reactor), and by 1958, the USA had completed the construction of the Shippingport Nuclear Power Plant, the first of its kind PWR (Pressurized Water Reactor) [8]. The most common nuclear reactors in operation are of the PWR type. This includes both light-water reactors and CANDU-type nuclear reactors with heavy water cooling [9]. The UK built the first GCR (Gas Cooled Reactor) fission reactor, the Magnox type reactor, in 1956. In addition to developments in this field, British astrophysicist Arthur Eddington came up with a theory that stars obtain their energy by converting hydrogen into helium [10]. The theory in question was later supported with mathematical modeling by Robert D'escourt Atkinson and Fritz Houtermans, who proved that the energy source in stars is a nuclear fusion [11]. In the studies conducted by Atkinson and Houtermans, they also proved that it is not an essential condition for fusion reactions to be carried out at high temperatures as well as in stars, proving that these reactions can also be carried out on Earth. Thanks to the studies carried out during this period, the feasibility of fusion reactors and their development gained momentum. In 1950, Andrei Sakharov and Igor Tamm invented the concept of TOKAMAK (Toroidal Camera and Magnitaya Katushka/Toroidal Chamber and Magnetic Coil), a kind of magnetic confinement fusion device [12]. Following this development, Lyman Spitzer and Richard F. discovered Stellarator and the magnetic mirror concept founded by Post and Gersh Budker in 1951 [13]. Only at the end of the 1960s, the TOKAMAK developed with experimental research by Lev Artsimovich on TOKAMAK systems. Research and development were carried out as the most promising concept of fusion reactors. When fission and fusion reactions and reactor concepts were evaluated in detail, it was found that fission reactors have high power generation capabilities. However, a high amount of radioactive material and weapons-grade plutonium can be obtained. On the other hand, although theoretically environmentally friendly and sustainable energy can be produced in fusion reactors, it has yet to be a viable energy production concept for economic reasons in today's conditions [14]. The concept of a hybrid (fission-fusion) reactor was introduced by Edward Teller, Hans Bethe, and Eugene Wigner to eliminate the negative characteristics of both reactor technologies mentioned before [15]. The main purpose of the hybrid reactor is to use the high-density neutron flux generated by a fusion reactor to maintain a nuclear fission reaction and simultaneously produce fissile fuel from fertile (^{232}Th and ^{238}U) materials. In addition, the waste from fission reactors can also be used as fuel in hybrid reactors. Since hybrid reactors are still at the research and development stage, studies on this reactor concept have yet to go beyond conceptual and numerical. When the hybrid reactor designs and calculations made up to the present day were examined, it was seen that fissile fuel production, radioactive waste disposal, and energy production coincide [16-17-18]. In these studies, neutronic calculations included fuel, blanket structure and content, reactor geometry, fuel production, and waste consumption in hybrid reactors. In the IFE (Inertial Fusion Energy) experiments conducted in the USA in the 1970s, studies were conducted on preserving plasma and creating a fusion reaction with high-power dense lasers. In the concept in question, energy transfer and storage to the fusion fuel have been carried out with lasers. Although the plasma production and fusion reaction are successful, the energy obtained corresponds to only 1% of the energy expended. According to the findings, further research and development studies should be carried out for the IFE concept to become a neutron source for fusion or hybrid reactors [19].

Today, magnetic fusion is the most important application in fusion reactors, and the concepts of TOKAMAK, Stellarator, and Magnetic Mirrors have been developed in the category of magnetic fusion reactors. In the early 1980s, the concepts of TOKAMAK and Magnetic Mirrors were compared in experiments conducted at the Princeton Plasma Physics Laboratory. According to the findings obtained, it was found that the TOKAMAK concept is much more suitable for fusion and hybrid reactor technologies [20].

1.1. AIMS AND OBJECTIVE

In this thesis study, a hybrid reactor design was developed, referencing the ITER geometry and incorporating a D-T fueled plasma neutron source. SS 316 LN-IG was selected as the first wall and thermal shield material, while a eutectic mixture of LiF-ThF_4 and LiF-UF_4 was chosen as the coolant and fuel for two different reactor models respectively, considering the material's eutectic points. The study results were evaluated based on several key parameters;

- Tritium breeding ratio,
- Energy multiplication factor,
- Radiation damage,
- Fissile fuel generation,
- Fuel burn-up,
- Fissile power generation.

The aforementioned parameters will be evaluated within the hybrid reactor model, which has been developed using the Monte Carlo N-Particle Transport Code (MCNP).

An interface code MCNPAS has been developed to address the time-dependent parameters that cannot be directly calculated by MCNP, and the corresponding analyses have been completed. The primary objective of this study is to identify the reactor model that exhibits optimized neutronic performance for the hybrid nuclear reactor. It is anticipated that this study will serve as a valuable resource for future research and applications in this field.

CHAPTER 2

LITERATURE REVIEW

When the studies in the literature about fusion and hybrid reactors are examined, it is seen that these studies focus on the blanket structure and components of the reactor. The components of the blanket are to be examined, there are the first wall, coolant, fuel zone and /or tritium production zone, reflector, insulation and magnets in this structure.

In hybrid reactors, fusion and fission concepts are applied together and the disadvantages of both technologies have been tried to be eliminated. In order to produce fissile fuel, fusion-fission hybrid reactors (FFHRs) can supplement, extend, or even completely replace conventional fast breeders. This function highlights their adaptability and significance in the nuclear energy industry. For this reason, it's crucial to acknowledge FFHRs as creative solutions that have the ability to address pressing energy issues and influence nuclear power in the future, rather than only as transitory technologies [21-22-23]. A paper published by Reed, TOKAMAK hybrid reactor modeling was performed. In this study, D-T fueled plasma is included as a neutron source. Instead of low power generation, as in fusion reactors, the possibility of high power generation of the hybrid reactor concept was seen as a result of calculations [24]. Wu et al., [25] proposed a conceptual hybrid reactor design namely FDS-EM (Fusion Driver System-Energy Multiplier). In the study carried out with reference to the parameters of the EAST (Experimental Advanced Superconducting TOKAMAK) reactor, it was seen that the design was suitable for sustainable energy production. Additionally, a research on the SABER HFFR design concept was conducted at Georgia Tech University, looking at safety studies, fusion R&D needs, development programs, FFH drives, neutron source development windows, SABR reactor design, conversion fuel cycles, and technical obstacles as well as recommendations. [26]

In different studies conducted by Nishikawa, Sawan, and Abdou, the tritium breeding ratio and balance in D-T fuel fusion reactors were examined. Accordingly, it has been calculated that the tritium production rate should be at least 1.05~1.1 [27-28].

In the study conducted by Zheng and Todd, the parameters affecting tritium production in the DEMO fusion reactor were examined. According to the results obtained, it has been emphasized that increasing the first wall thickness and the density of the material used in this layer, reduces tritium production. It has been shown that for the production of tritium required for sustainable fusion reactions, the reactor can be provided by increasing the ^{6}Li isotope density in the tritium breeding zone [29].

Şahin and others have completed the evaluation of the tritium production rate in the hybrid reactor structure according to the changing coolants according to the results of the neutronic analysis carried out using the MCNP5 v1.4 program. They used Flinak, enriched lithium (90%), $Li_{17}Pb_{83}$ and Flibe as coolants in their studies. According to their findings, they found that enriched lithium reaches the highest tritium production value, and this is due to the highest lithium density [30].

In the study conducted by Ishibashi and others, neutronic performance of Li_2TiO_3, lithium, Flibe, and $Li_{17}Pb_{83}$ were investigated. According to the study, the highest tritium production value was obtained in the simulation using He as a coolant, F82H as the first wall material, and Li_2TiO_3 for tritium production [31].

Catalán and others have completed the neutronic analysis of the DEMO fusion reactor, according to the reference design parameters containing the He/LiPb dual coolant system with the MCNPX program. In this study, the effects of the first wall thickness were also examined, and it was observed that the tritium production rate changed inversely proportional to the thickness of first wall [32].

Youssef and others have completed the neutronic analysis of a fusion reactor consisting of combinations of lithium, Flibe, Flinabe, and $Li_{17}Pb_{83}$ coolants with ferritic steel and SiC first wall materials. The effects of lithium enrichment in coolants were also examined. As a result of all the findings, it was found that lithium has the highest tritium production rate [33].

Titarenko et al. conducted experimental and simulation studies to assess ambient dose equivalent rates and reaction rates in thorium, while also estimating the relative deposition of 232U/233U under natural irradiation conditions in a fusion blanket with a specific neutron spectrum [34]. Leshukov et al. performed a neutronic analysis of nuclear fuel production in a hybrid fission-fusion reactor's breeding blanket, investigating various combinations of source materials (metallic uranium, thorium, uranium dioxide, thorium dioxide, uranium nitride, and thorium nitride) and coolant options (lead, sodium-potassium eutectic, carbon dioxide, water, steam-water mixture, and heavy water) [35].

Yalçın et al. conducted a time-dependent neutronic analysis of a deuterium-tritium (D-T) sourced hybrid reactor, utilizing UC, UO_2, and UN fuels, and employing natural lithium, Flibe, Flinabe, and $Li_{20}Sn_{80}$ as coolant materials. They observed that the density of uranium isotopes in the fuels was directly proportional to fissile fuel production. Notably, natural lithium was found to absorb more neutrons for tritium production than other coolants, leading to an increase in tritium production and a reduction in fissile fuel production [36].

Wang et al. used CLAM RAFM as the first wall material, $Li_{17}Pb_{83}$ as tritium production material, and UO_2, UC, UN, and U-10Zr as fuel in their hybrid reactor study. In this study, while sufficient tritium breeding is made in all models, 1.6-3.5 tons of plutonium can be produced annually in the reactor with a fusion power of 500 MW, while 1700-3200 MW thermal power can be produced. According to the results obtained, metallic uranium fuel showed the best neutronic performance [37].
In their investigation of a hybrid reactor structure comprising Flibe molten salt with thorium-uranium, Zhao et al. referenced ITER geometry. Their reactor was capable of producing tritium without the need for lithium enrichment. Their findings suggested that UF_4 and ThF_4 fuels, in varying proportions with Flibe, are suitable for use in hybrid reactor structures [38]. Conversely, Vanderhaegen et al. concluded that Flibe's neutronic performance was suboptimal, and that UF_4 fuel exhibited significantly better

neutronic performance than ThF_4, making it a preferable choice for fuel production [39].

Hancerlioğulları's study involved the addition of UF_4 and ThF_4 heavy metal salts as molten salt in the APEX hybrid reactor in varying amounts for energy reproduction. The study recommended the addition of heavy metal salt up to a quantity that maintains the tritium production rate above 1.05, and found that UF_4 heavy metal salt had a substantially better neutronic performance [40].

Murata and his team modeled solid-fuel hybrid reactors by taking the geometry and components of ITER and JT-60 fusion reactors as references. In the time-dependent simulated hybrid reactor, Li, Li_2O, Li_2ZrO_3, Li_4SiO_4, Li_2TiO_3, and $LiAlO_2$ were used as tritium breeder and a mixture of UO_2 and Pu as fuel composition. As a result of their neutronic analysis shows that sufficient tritium production, energy multiplication factor, and k_{eff} values are obtained in the hybrid reactor with low fusion power and first wall load [41].

Shanliang and Wu compared their studies neutronic performances of tritium production materials. According to the findings, while Li_2O showed the best neutronic performance in solid tritium production materials, LiF reached the highest tritium production value in liquids. According to the study, it is emphasized that oxygen and fluorine's high neutron slowing ability is very effective in tritium production [42].

Zhao and his team used 71% LiF–2% BeF_2–13.5% ThF_4–8.5% UF_4–5% PuF_3 as a fuel coolant mixture in the molten salt hybrid reactor they designed. As a result of the calculations, it is seen that the reactor produces sufficient tritium for sustainable operation, as well as a high energy multiplication factor value [43].

In the fast and thermal hybrid reactor study by Xiao et al., the tritium production value reached its maximum at 40% lithium enrichment. In contrast, the energy multiplication factor decreased in the opposite direction of this trend. In the time-dependent study, 8.5 tons of ^{233}U was produced in 6 years with the existing fuel in the reactor without adding thorium to the fuel inventory [44].

In their study, Zhirkin and his team modeled a fusion-sourced hybrid reactor. Their modeling examined the production of ^{233}U, ^{239}Pu and ^{3}H isotopes. In the reactor models, simulations involving solid blanket structure, fuel dissolved in heavy water, and fuel dissolved in molten salt were completed separately. According to the findings, although the fuel production potential is lower in the hybrid reactor structure in which a mixture of molten salt and ThF_4 is used, it is emphasized that it is necessary to choose fuel mixtures containing molten salt for the safe hybrid TOKAMAK reactor [45].

In their study, Xiao and his team completed the neutronic analysis of thorium and uranium-fueled hybrid reactor. Flinak was used as the molten salt in the simulation of a fusion reaction sourced reactor with 500 MW power and reference to the ITER structure. According to their findings, it was observed that the production of ^{233}U in the reactor increased when helium was used instead of water as a coolant. In the simulation of the reactor using helium, it was observed that the production of ^{233}U increased from 4 tons to 10 tons in 20 years, and the tritium breeding ratio value increased from 1.05 to 1.13. In the simulations, it is seen that TBR, M, burnup value of the fuel, and ^{232}Th (n, γ) reaction rate increase with increasing the thickness of the thorium zone [46].

In the study conducted by Ma et al., the neutronic analysis of the hybrid reactor containing ThN, and ThO_2 fuel was completed. Accordingly, it was observed that the tritium production value increased when the neutron multiplier BeO was added to the fuel mixture. In the study, the production of ^{233}U, a fissile fuel, was also examined, and it was concluded that the ThN and BeO blended fuel had the highest fissile fuel production value [47].

In the studies by Günay and Kasap, the neutronic analysis of the hybrid reactor consisting of $Li_{20}Sn_{80}$ coolant and coolant-fuel mixtures containing Pu, PuF_4, and PuO_2 containing reactor-grade plutonium was completed. The study observed that as the percentage of fuel in the coolant-fuel mixture increases, the tritium production value decreases, and the fissile fuel production and the energy multiplication factor increase [48-49].

In the study conducted by Baxi and Wong, the use of helium as a coolant, its properties, and design parameters was examined. Accordingly, it was determined that the first wall load limit was 10 MW/m^2 in the design, and it was found that the gas in question gave the best results at a load of 5 MW/m^2 [50].

In the simulations done by Karakoç, 1DS-ODS steel, SiC, was used as the first wall material in the D-T fueled magnetic fusion reactor. Flibe, Flina, Flinak, and natural lithium were used as the region's coolant, and the reactor's neutronic analysis was completed with the simulations using MCNP. According to the results, it was seen that the neutron damage in the reactor was less in the models using 1DS-ODS steel, and the tritium production values were higher than in other models. In the study, it was seen that the model with the most extended life reactor and sustainable tritium production was the model formed by the combination of 1DS-ODS steel and Flibe [51].

In the study conducted by Farmer and others, the interactions of ODS steels with molten salts are studied. It is observed that 12YWT, 14YWT, and MA956 steels begin to lose their material properties at high temperatures. It has been observed that Ta-1W and Ta-10W materials containing tantalum have longer material lifetimes in molten salts at high temperatures compared to ODS steels [52].

In the study completed by Şahin, in hybrid reactor simulation using ENDF/B-V, ENDF/B-VI, and CLAW-IV cross-section libraries, ferritic/martensitic steels, ODS ferritic steel, vanadium alloy, and SiC–SiC composites, stainless steels, and high conductivity copper alloy was used as the first wall material. In the study examining the tritium breeding ratio and radiation damage, vanadium alloy (V_4Cr_4Ti) and $Cu_{0.5}Cr_{0.3}Zr$ were found to be the most resistant to radiation damage, with the highest tritium production rate [53].

In the study by Tunç et al., radiation damage parameters of various experimental and commercial steels used in the hybrid reactor structure containing ODS steels were investigated. It has been observed that iron, chromium, and tungsten isotopes in steels extend material lifetime, while magnesium and vanadium isotopes shorten material lifetime. As a result of the neutronic analysis made in the study, it is seen that 1DS-ODS steel performs better than other steels [54].

In the study by Şahin et al., the neutronic analysis of the reactor simulations created with the changing coolants of the first wall materials and thicknesses of the magnetic fusion reactor created with reference to the ITER geometry was completed. In the study, SS 316 LNIG, PM2000 ODS, and CLAM were used as the first wall material, and FLiBe, FLiNabe, and FLiPb were used as coolants. According to the findings obtained in the study, the tritium breeding ratio decreases as the thickness of the first wall material increases. Considering the tritium breeding ratio, it is seen that the reactor model using PM2000 ODS and FLiPb exhibits the best performance. Considering the neutron damage and the life of the first wall material as evaluation criteria, CLAM was the first wall material with the longest material lifetime [55].

The interaction of RAFM steels with liquid PbLi was investigated in the study by Ding and others. The study found that high-temperature PbLi coolant has a high corrosion risk with RAFM steels. Since the mentioned corrosion risk may disrupt the operation of fusion reactors, it has been found that mixing aluminum with coolant can reduce this risk. Although Y, Ti, Al, Si, and Zr increase the durability of RAFM steels, it does not have an essential role in reducing the corrosion effect of PbLi coolant [56].

Wang and others have studied the interactions of CLAM RAFM steel with PbLi coolant. Their findings showed that working with steel and coolant should not be preferred due to its high corrosion effect [57].

Gaganidze and Aktaa tested the mechanical properties of EUROFER97 RAFM steel up to 550 ℃ in their study. Their study found that RAFM steels are highly functional in the operating temperature range of 350-550 ℃ [58].

Lindau and others examined the development of EUROFER steels in their study. It has been seen that RAFM EUROFER 97 steel gives successful results in ITER test modules. With the improvements that ODS EUROFER steels can operate at higher temperatures, it has been determined that using this steel group in reactors may be more appropriate [59].

In the study conducted by Huang and others, they evaluated the development of RAFM steels, and their use in ITER and DEMO projects. Tritium production in the module in question is still necessary for research and development activities, the use of steel, hydrogen, and helium to be more resistant to damage. It was emphasized the need for more experimental studies [60].

In the study by Şahin and Übeyli, austenitic stainless steels, ferritic/martensitic steels, vanadium alloys, refractory metals, and composites were proposed as the first wall materials for nuclear fusion reactors. Although using austenitic steels at a maximum temperature of 700°C is a negative feature compared to other types of steel, it is seen that with the liquid first wall concept and additional precautions, it can be resistant to neutron flux originating from fusion reaction [61].

In their study, Baldev et al. examined the stainless steels used and can be used in fusion reactors. In their study, it was seen that the mechanical properties of steels and their usage areas in the past were also examined. Accordingly, it is suitable for use in 304L, 304LN, 316L, 316LN, and 316 SS fission and fusion reactors [62].

CHAPTER 3

NUCLEAR REACTORS

Today, most of the energy needed by humanity is produced using fossil fuel-based energy production methods. Approximately 10% of the energy needs in question are produced using nuclear energy. Nuclear energy can be produced with three main reactor technologies in the current period. These technologies are as follows;

1- Nuclear Fission Reactors,
2- Nuclear Fusion Reactors,
3- Nuclear Fission-Fusion Hybrid Reactors.

3.1. NUCLEAR FISSION REACTORS

Nuclear fission reactors can be defined as reactors used for energy production by performing controlled nuclear fission reactions of chain reactions. Fission reactors can be classified according to the moderator, coolant, and fuel used in the reactor. In Table 3.1. classification of nuclear fission reactors was shown.

Table 3.1. Classification of nuclear fission reactors [64]

Reactor Type	Coolant	Moderator	Fuel
PWR	Water	Water	Enriched UO_2
BWR	Water	Water	Enriched UO_2
PHWR	Heavy Water	Heavy Water	Natural UO_2
LWGR	Water	Graphite	Enriched UO_2
AGR	CO_2	Graphite	Natural U (metal) and/or Enriched UO_2
FBR	Liquid Sodium	-------	PuO_2 and/or UO_2
HTGR	Helium	Graphite	Enriched UO_2

3.1.1. Main Components of Nuclear Fission Reactors

The main components of a nuclear fission reactor consist of fuel, moderator, coolant, control rods and reactor containment building.

3.1.1.1. Fuel

An essential component of a nuclear reactor is fuel. Currently, nuclear reactors in operation use uranium, thorium, and plutonium as fuel. In reactor designs using fissile fuel, ^{233}U, ^{235}U, and ^{239}Pu are used as the primary fuel component, and ^{232}Th and ^{238}U isotopes are used in reactors where fertile materials are used. In reactors where fertile materials are used, fissile fuel is produced through material interaction with neutrons, and energy is produced using this fuel. The main reactions involving the fissile and fertile materials in question are listed below.

$$\left.\begin{aligned}
&{}^{233}_{92}U + {}^{1}_{0}n \rightarrow {}^{94}_{38}Sr + {}^{137}_{54}Xe + 3{}^{1}_{0}n + 197.9\ MeV \\
&{}^{235}_{92}U + {}^{1}_{0}n \rightarrow {}^{92}_{36}Kr + {}^{141}_{56}Ba + 3{}^{1}_{0}n + 202.5\ MeV \\
&{}^{239}_{94}Pu + {}^{1}_{0}n \rightarrow {}^{103}_{40}Zr + {}^{134}_{54}Xe + 3{}^{1}_{0}n + 207.1\ MeV
\end{aligned}\right\} \boxed{\text{Fission reactions of fissile fuels}}$$

$$\left.\begin{aligned}
&{}^{232}_{90}Th + {}^{1}_{0}n \rightarrow {}^{233}_{90}Th \xrightarrow{\beta^-} {}^{233}_{91}Pa \xrightarrow{\beta^-} {}^{233}_{92}U \\
&{}^{238}_{92}U + {}^{1}_{0}n \rightarrow {}^{239}_{92}U \xrightarrow{\beta^-} {}^{239}_{92}Np \xrightarrow{\beta^-} {}^{239}_{94}Pu
\end{aligned}\right\} \boxed{\text{Fissile fuel generation reactions from fertile fuels}}$$

3.1.1.2. Moderator

In nuclear engineering, moderators are materials used to slow down fast neutrons and reduce energy levels. Reducing the energy levels of neutrons increases the probability of performing a fission reaction with fission fuels.

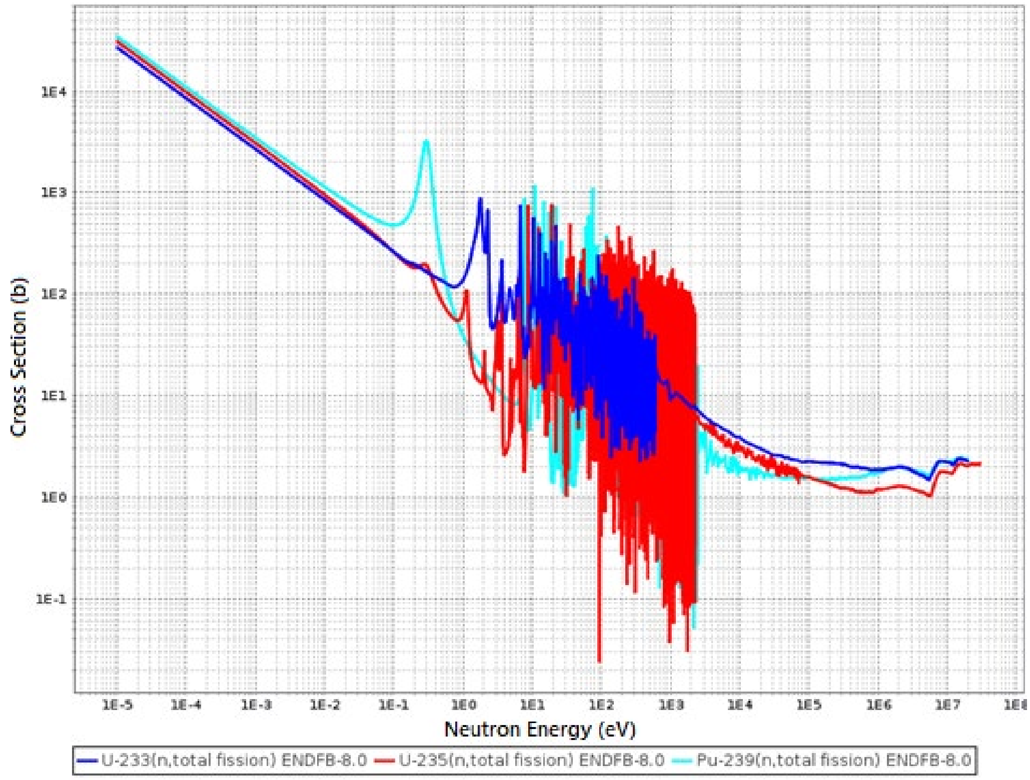

Figure 3.1. Fission reaction cross sections of fissile fuels [65].

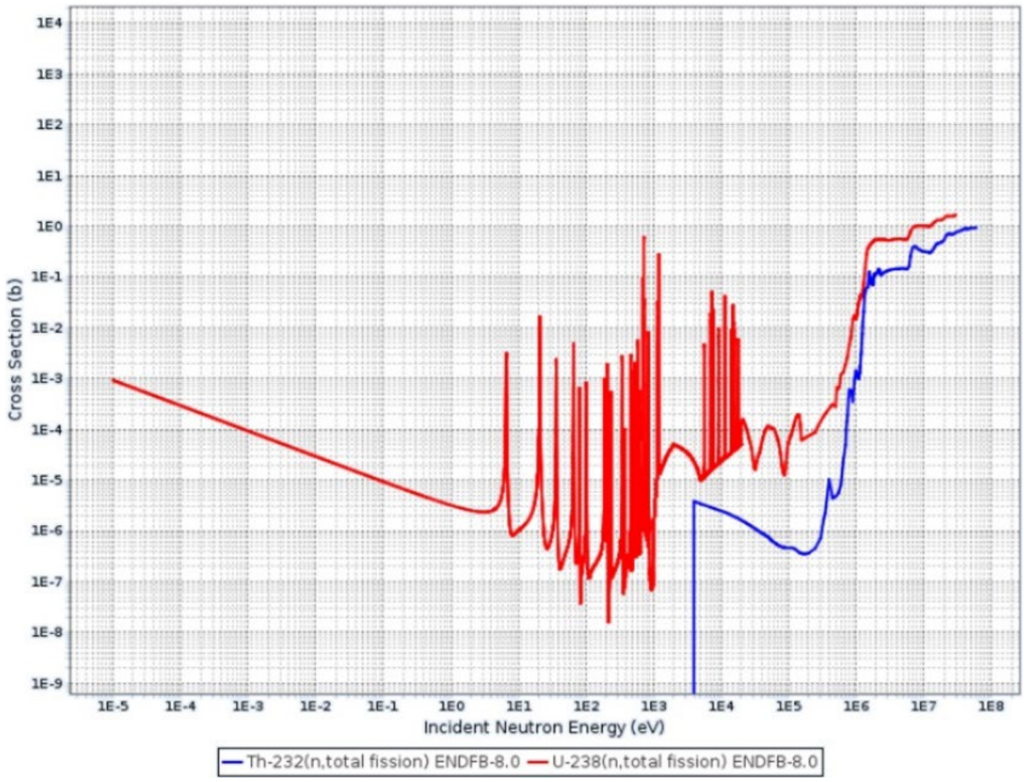

Figure 3.2. Fission reaction cross sections of fertile fuels [65].

As seen in Figure 3.1 and 3.2, the probabilities of fission reactions of fissile fuels vary according to the energy level of neutrons. The range in which these reactions are most likely to occur is the energy range of 0.01 eV ~ 0.30 eV, where neutrons are thermal neutrons. On this basis, moderators should lower their energy levels by slowing down neutrons. A good moderator must have the following properties;

- Low neutron capture cross section,
- High neutron scattering cross section,
- Chemical stability under radiation,
- Low atomic density,
- High boiling temperature,
- Neutron reflection.

3.1.1.3. Coolant

The heat released from the nuclear reactions should be removed from the reactor, and the heat energy obtained should be converted into electrical energy. Coolants are used for the transformation in question, and the coolants should have the properties given as follows;

- High specific heat,
- High heat transfer ability,
- Low viscosity,
- Low melting point for solids,
- High boiling point for liquids,
- Good chemical stability at radiation and high temperatures,
- Low neutron absorption cross section,
- Low corrosion and erosion effects on reactor materials,
- Safe, easily available and economical.

3.1.1.4. Control Rods

Control rods in fission reactors are materials used to control the rate of fission reactions and the reactor's power level and to ensure the reactor's safety. Control rods fulfill the tasks mentioned earlier according to their position and number in the reactor core. While starting the reactor, the number of control rods is increased or decreased to change the power level while having as few control rods as possible in the core. In an emergency, all control rods are dipped into the reactor core to shut down the reactor's power generation completely. To perform these operations, the control rods must resist high temperatures and have a high neutron absorption cross-section to stop fission reactions.

3.1.1.5. Reflector

Reflectors are structures located right next to the core in fission reactors. This structure has the property of reflecting neutrons escaping from the reactor core. Thus, the number of neutrons required to continue the fission reactions is reduced, and the reactor core is smaller. Materials with low neutron absorption cross-section and high neutron scattering cross-section should be selected when choosing reflector material.

3.1.1.6. Reactor Containment Building

The reactor containment building is generally the structure containing the nuclear reactor components. This structure is constructed from reinforced steel, concrete or lead. The reactor containment is designed to prevent the radiation released as a result of fission reactions and the scattering of radioactive materials into the atmosphere in case of an accident.

3.2. FUSION REACTORS

Nuclear fusion reactors can be defined as reactors used to produce electrical energy from the energy released in the nuclear fusion reaction. Nuclear fusion reactors use plasma confined as a neutron source and are classified into two main categories according to the confinement method: inertial confined and magnetic confined fusion reactor.

3.2.1. Inertial Confinement Fusion Reactors

In inertial confinement fusion reactors, the deuterium-tritium fuel is bombarded with lasers containing dense particles and photon beams and compressed until it has a sufficient density (1000 ~ 10000 times denser than the solid state of the fuel) and temperature. Thus, it becomes suitable for performing the fusion reaction. The fuel emits neutrons and radiation in controlled thermonuclear explosions during this process. The released thermonuclear energy is carried by reaction products, including X-rays, neutrons, and charged particles, and converted into usable energy in the blanket structure and electrical energy using thermodynamic cycles. While the size of these explosions is limited according to the energy produced in the reactor, the blanket structure of the reactor is designed accordingly. Figure 3.3 shows the inertial confinement fusion reaction in the reactor's vacuum chamber, and Figure 3.4 shows the representative image of the inertial confinement fusion reactor.

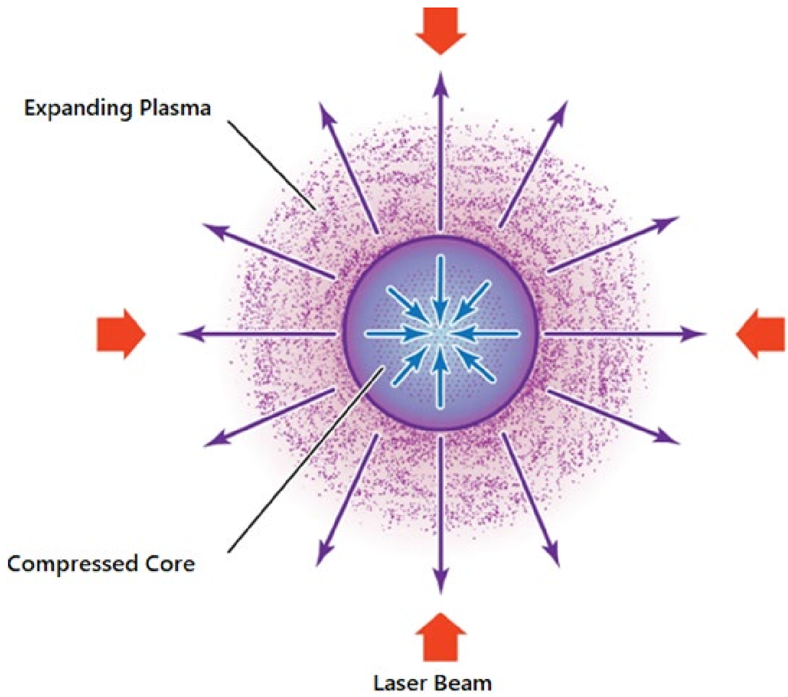

Figure 3.3. Inertial confinement fusion reaction [66].

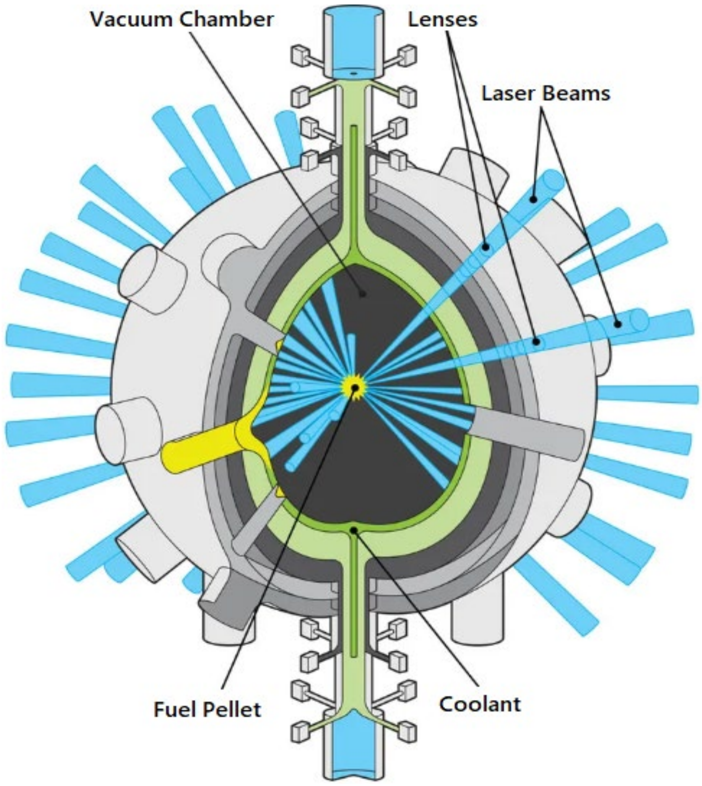

Figure 2.4. Representative image of an inertial confinement fusion reactor [67].

3.2.2. Magnetic Confinement Fusion Reactors

Like inertial confinement fusion reactors, deuterium-tritium is also used as fuel in magnetic fusion reactors, but the plasma density is much lower. In magnetic confined fusion reactors, the D-T plasma is confined by magnetic fields created by superconducting magnets, and a representative image of the reactor is shown in Figure 3.5.

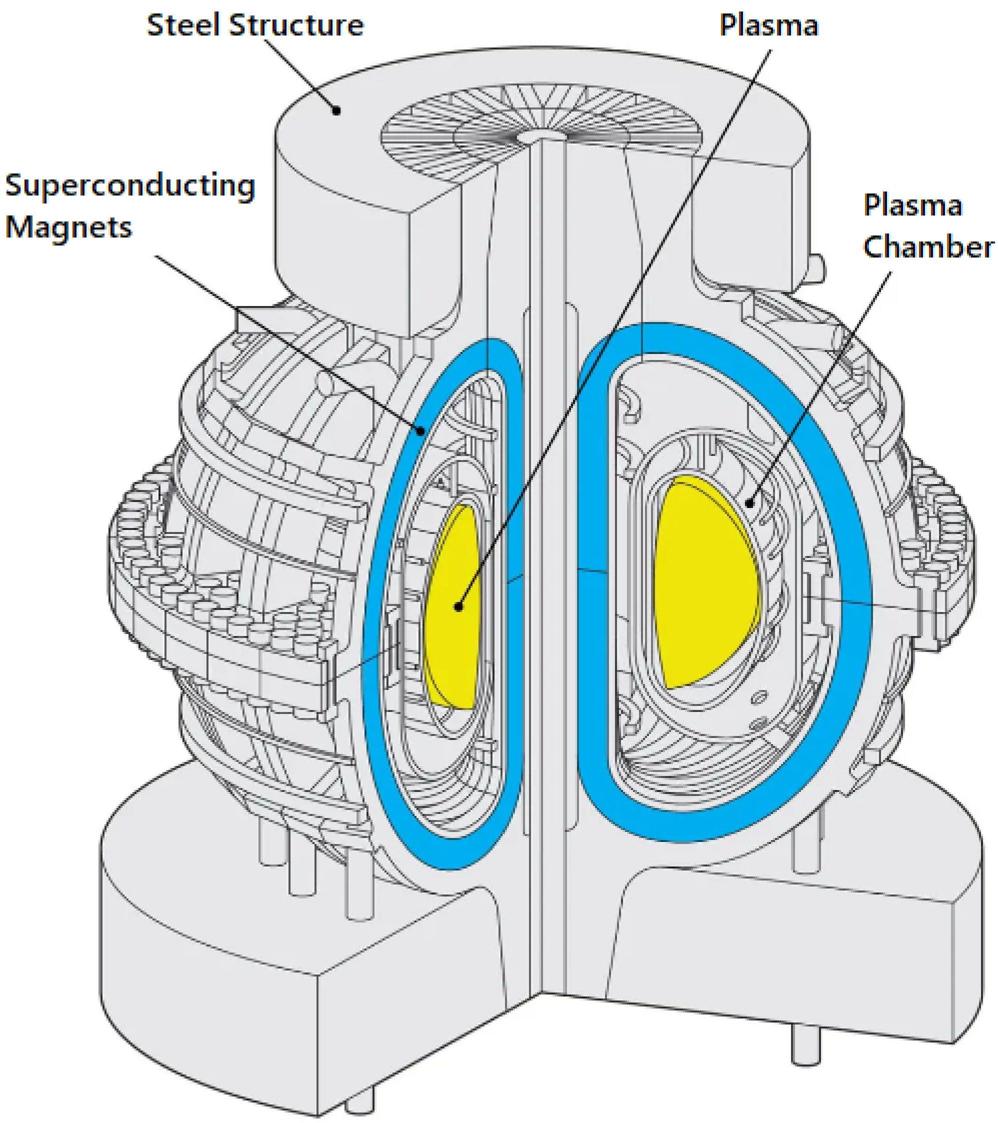

Figure 3.5. Representative image of a magnetic confinement fusion reactor [67].

The plasma contained in the reactor is exposed to two types of magnetic fields. The toroidal field lines run horizontally along with the plasma ions. In contrast, the poloidal field lines run vertically around the plasma, compressing the plasma towards the center of the plasma chamber. A helical magnetic field is created by merging these two magnetic field lines, and the plasma is confined. Figure 3.6 shows the magnetic field lines acting on the plasma.

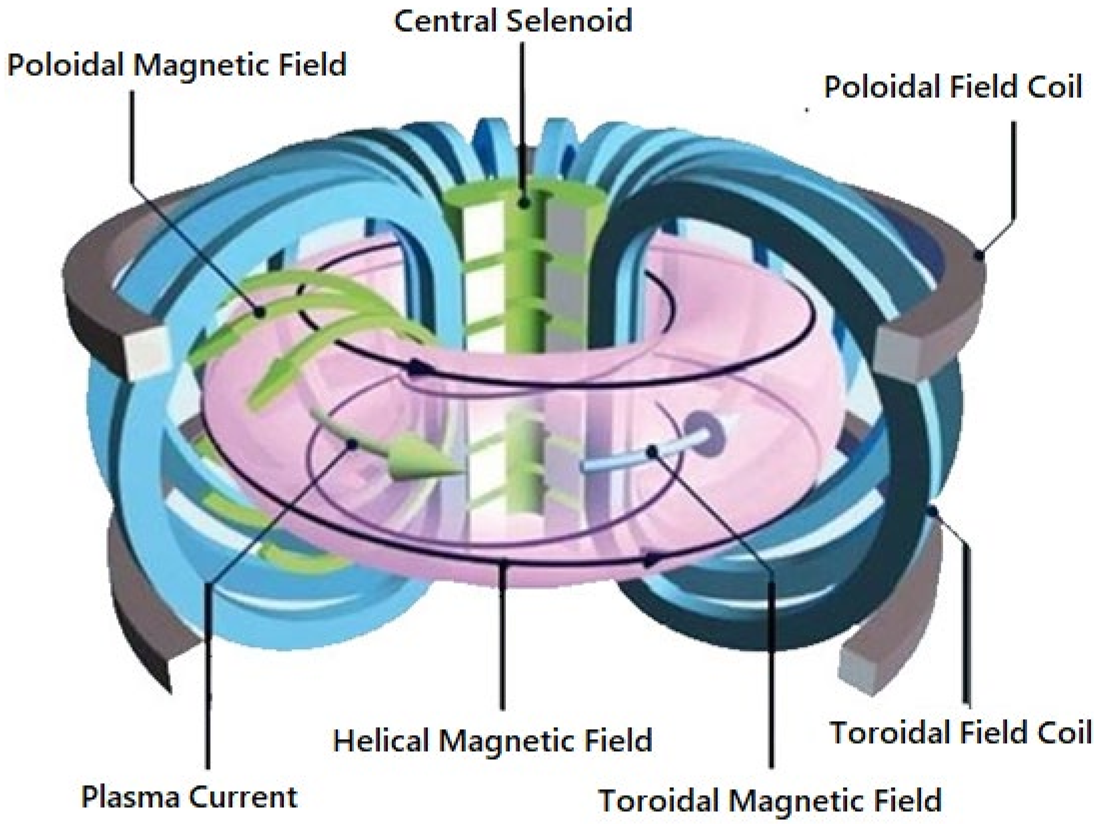

Figure 3.6. Magnetic field lines in a magnetic fusion reactor [69].

3.2.3. Main Components of Fusion Reactor

The main components of a nuclear fusion reactor are a plasma chamber, blanket structure, divertor, magnets, and cryostat. Figure 3.7 shows the main components of a nuclear fusion reactor.

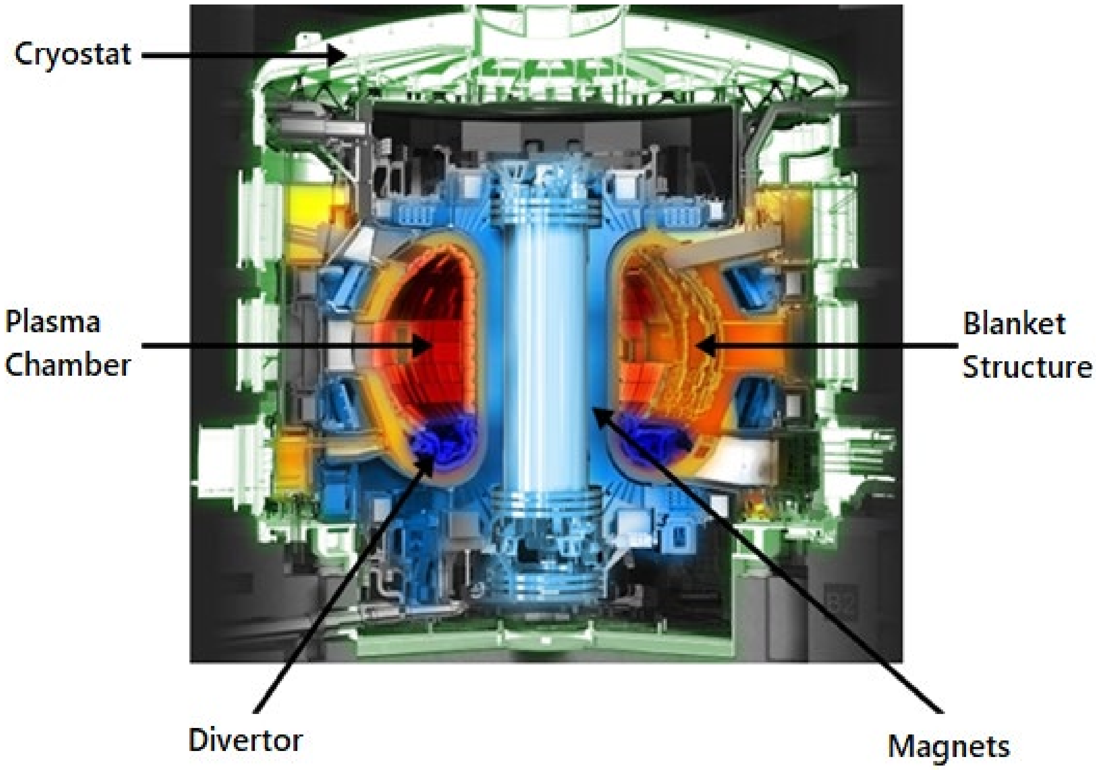

Figure 3.7. Main components of fusion reactor [70].

3.2.3.1. Plasma Chamber

The plasma chamber is the location where plasma, generated from fusion reactions, is confined. Within this chamber, vacuum conditions are established to suit the reactor's concept, limiting the plasma's contact with reactor components. In this study, deuterium-tritium plasma is utilized as a fusion neutron source within the plasma chamber. For the purpose of three-dimensional calculations, a plasma chamber featuring a D-shaped neutron source distribution was employed. Since it didn't hit the first wall of the blanket structure, it was considered to be isotropic. In comparison to a point and spherical source, this provides a more realistic and precise source definition for hybrid nuclear reactors.

3.2.3.2. Blanket Structure

The blanket structure is the layer that surrounds the plasma chamber, and where the energy released due to the fusion reactions taking place there is made available. In this layer, the energy released as a result of fusion reactions is converted into heat energy, and electrical energy is produced with the help of cooling systems. In order to ensure the neutron balance in the reactor and to reduce the losses in energy production as much as possible, fuels that release neutrons as a result of the fusion reaction and that can perform exothermic reactions should be selected. The reactions of fuels that can release neutrons as a result of fusion reactions are listed below.

$D + T \rightarrow n + {}^{4}He + 17.586\ MeV$ $\quad$ $D + {}^{6}Li \rightarrow n + {}^{3}He + {}^{4}He + 1.796\ MeV$

$D + {}^{6}Li \rightarrow n + {}^{7}Be + 3.380\ MeV$ $\quad$ $T + T \rightarrow 2n + {}^{4}He + 11.327\ MeV$

$D + D \rightarrow n + {}^{3}He + 3.267\ MeV$ $\quad$ $T + {}^{3}He \rightarrow n + p + {}^{4}He + 12.092\ MeV$

In the fusion reactions listed, the most likely reaction is the (D, T) reaction. Since tritium involved in the reaction is not a naturally occurring isotope, it must be produced in the blanket structure of the reactor. Tritium production reactions are listed below.

$n\ (thermal) + {}^{6}Li \rightarrow {}^{4}He + T + 4.8\ MeV$

$n\ (fast) + {}^{7}Li \rightarrow {}^{4}He + T + n - 2.5\ MeV$

As seen in the listed reactions, the reactor blanket must contain lithium components for the production of tritium. In these tritium-produced reactions, the exothermic reaction is carried out with low-energy thermal neutrons, while the endothermic reaction is carried out with high-energy fast neutrons. Although the deuterium isotope used as a fuel can be found in nature, tritium is not found in nature, so its production must be carried out in the blanket structure. Tritium is produced in the blanket structure's coolant and tritium breeding layer, allowing the reactor to produce its own fuel. Finally, the reactor's components are protected against the high amount of radiation emitted from the plasma.

3.2.3.3. Magnetic Coils

In fusion reactors, magnetic coils (superconducting magnets) are vital components for the control and confinement of the plasma. Here are the main functions of magnetic coils:

- Plasma confinement
- Plasma shaping and control
- Temperature and density control

In fusion reactors, plasma at extremely high temperatures cannot be directly contained by reactor components. The plasma is maintained stably at the center of the reactor through the use of very strong magnetic fields generated by magnetic coils, thereby preventing its contact with the first wall materials. These magnetic fields, particularly in TOKAMAK-type reactors, are instrumental in forming a D-shaped plasma. Additionally, the temperature and density of the plasma are regulated in accordance with the magnetic field produced by these coils.

3.2.3.4. Divertor

In fusion reactors, divertor is an important component used especially in TOKAMAK type reactors. The divertor is the reactor component that prevents the plasma from coming into contact with the blanket structure of the reactor, keeping the plasma content pure and evacuating the fuel present in the plasma. These functions are critical to ensuring the efficient and safe operation of fusion reactors.

3.2.3.5. Cryostat

The cryostat in fusion reactors is a crucial cooling device responsible for keeping the reactor's vital components at extremely low temperatures. These reactors employ superconducting magnetic coils to control high-temperature plasma. The effectiveness of these coils depends on their maintenance at cryogenic temperatures, specifically at or below -140°C. This need serves as the essential basis for the cryostat's function. In addition to its main purpose of cooling, the cryostat plays an important part in maintaining the stability of reactor components by effectively absorbing and reducing any mechanical stresses that may occur during the operation of the reactor.

3.3. FISSION-FUSION HYBRID NUCLEAR REACTORS

Fission-fusion hybrid nuclear reactors are advanced technological systems that utilize the principles of both fission and fusion reactions to produce energy. In these reactors, the fusion reaction primarily serves as a prolific neutron source. The neutrons produced through fusion initiate fission reactions in the fissile or fertile material embedded in the reactor's blanket structure. This unique configuration allows the reactor to harness energy from both fission and fusion processes, potentially offering a more efficient and sustainable approach to nuclear energy production compared to traditional reactors.

The general structure of hybrid reactors is very similar to fusion reactors, and the main difference is that it has fissile/fertile fuel in the blanket structure. Figure 3.8 illustrates a schematic representation of a hybrid reactor.

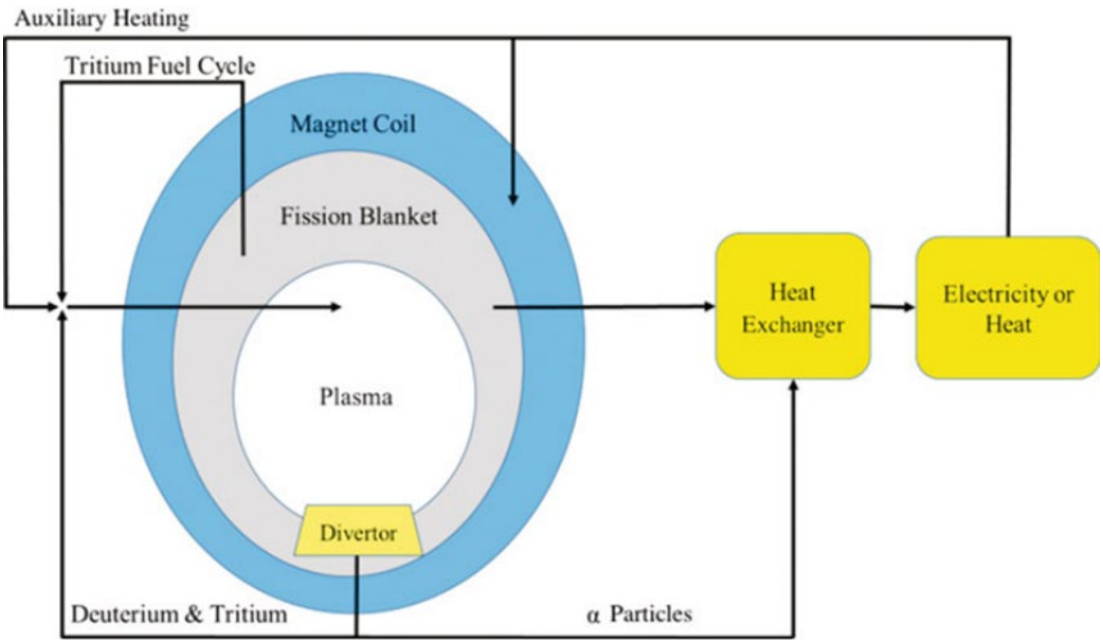

Figure 3.8. Schematic representation of the hybrid reactor [71].

Neutrons released from the fusion reaction in the plasma perform the reactions required for the production of tritium in the blanket structure of the reactor and the production of fissile fuel. The heat released from these reactions produces electrical energy with heat exchangers and a thermodynamic cycle. When constructing a hybrid nuclear reactor for optimal performance, it is important to carefully examine certain crucial variables. Ensuring the long-term profitability of a power plant requires the establishment of a self-sufficient fuel cycle. To meet this requirement, tritium must be produced within the blanket structure of a hybrid reactor. This phenomena arises from the combustion of plasma, which consumes a portion of the fuel, as well as the radioactive decay of tritium and the inherent uncertainties in the calculation process. Multiple studies in the literature showcase a wide range of approaches and material combinations used for tritium production in fusion and hybrid nuclear reactors. The experiments focused on materials that generate tritium and the thicknesses of the layers. The second requirement for a viable hybrid reactor design is to protect reactor components from the high-energy neutrons generated by fusion events occurring in the plasma chamber. Therefore, it is essential to incorporate an intermediary barrier between the plasma and reactor layers. The aforementioned layer is referred to as the initial barrier, and it can be formed by employing two fundamental concepts: solid and liquid. The third requirement for a successful hybrid reactor design is the capacity to maintain a fission reaction while also guaranteeing the essential synthesis of tritium for the plasma. Hybrid reactor design has advantageous features compared to fusion and fission reactor designs. These advantages are respectively;

1) Since the fusion energy gain value is close to one in hybrid reactors, the fusion power required is much lower than for fusion reactors. This difference arises because, in hybrid reactors, fusion reactions serve primarily as a neutron source rather than for power generation, which contrasts with the role of fusion reactions in standard fusion reactors. Owing to the fission reactions within the blanket structure of hybrid reactors, any adverse effects of the low-power source on the reactor's overall power generation capacity can be effectively compensated.

2) While the tritium production rate value is around 1.1 in fusion reactors, this value can exceed 2 in hybrid reactors due to the high neutron flux in the blanket structure [9].

3) Owing to the high-energy neutrons released from fusion reactions, hybrid reactors are not only capable of producing fissile fuel but also offer the

potential to process the waste from fission reactors. The rate of fuel production and the capacity for waste conversion in hybrid reactors surpass those of existing fission reactors.

4) Fuel diversity is much higher in hybrid reactors than in fission reactors. This diversity includes flexibility in the enrichment rate of the fuel and even the use of depleted uranium in hybrid reactors.

Despite all these advantages, hybrid nuclear reactors also have the disadvantages of fission and fusion reactors. Examples of these disadvantages are the limitations of fuel reserves in fission reactors, safety risks and the design of reactor components in fusion reactors, manufacturing difficulties, and sustainable plasma limitations. Many studies have been carried out so far to eliminate these disadvantages. As a result of the research;

1) In order to minimize the effect of the limited fuel reserves on energy production, the options for producing fissile fuel from fertile fuels in hybrid reactors and evaluating the wastes of fission reactors are evaluated.

2) The majority of the security risks are the fission reactions that take place in the blanket structure. In order to eliminate this risk, the fuel is in the blanket together with the coolant, molten salt. When the chain fission reaction gets out of control, the fuel can be quickly evacuated from the reactor with the coolant.

3) To mitigate design and production challenges in fusion reactors, the fusion power gain value of the plasma is deliberately maintained at a low level. Consequently, this approach reduces the radiation and thermal load on the reactor's first wall layer, thereby diminishing the impact of radiation damage on the reactor's other layers.

4) The limitation of plasma in fusion reactors is one of the most critical challenges. It has been seen that the concept of inertial confinement fusion reactor is not suitable for the hybrid reactor concept because the plasma cannot be obtained in sufficient density, and the confinement conditions in accordance

with Lawson's criteria are not met. However, the magnetic fusion concept is still a very realistic approach for fusion and hybrid reactor concepts, which is still under research and development. TOKAMAK concept, a sub-branch of magnetic fusion reactors, is an up-and-coming technology for fusion and hybrid reactors.

CHAPTER 4

METHOD

In the method part of the thesis, the necessary calculations for the input files used in the simulations made with MCNP and these calculation methods are shown.

4.1. CALCULATION OF ISOTOPE DENSITY

One of the most important calculations to be made while preparing the input files in hybrid reactor simulations is to calculate the densities of the isotopes used in the structure of the reactor. When calculating the atomic density of an isotope;

$$N = \frac{\rho \ x \ N_A}{A} \tag{1}$$

The equation is used. In this equation, ρ (atom/cm^3) represents material density, N_A (0,6022 x 10^{24} atom/mol) and A (g/mol) Avogadro number and atomic mass of an isotope respectively. To calculate the atomic densities of mixtures containing more than one isotope;

$$N_i = \frac{\rho_{mixture} \ x \ N_A \ x \ wf_i}{A_i} \tag{2}$$

Equation is used. Here, ρ_{mix} (atom/cm^3) refers density of mixture, wf_i and A_i sırasıyla expresses the mass ratio of the material in the mixture and the atomic density of the material in the mixture and. the subscript *i* is used to show the material number.

4.2. NEUTRON TRANSPORT EQUATION

The neutron transport equation is utilized in nuclear reactors to observe neutron behavior and conduct the necessary numerical analyses based on these observations. This equation facilitates the examination of neutron gains and losses within the reactor.

While it is not possible to predict the behavior of individual neutrons, the average behavior of a large neutron population can be described quite accurately using relevant information on neutron fluxes, cross-sections, and reaction probabilities. The neutron transport equation can be represented by the Boltzmann transport equation, which is an equilibrium statement. This equation, addressing the collision of uncharged particles from one atom to another, delineates the additions and subtractions made to radiation over incremental changes in space, energy, direction, and time.The equation below shows the Boltzmann transport equation.

$$\frac{1}{\upsilon(E)}\frac{\partial\phi(r,E,t)}{\partial t}+\vec{\Omega}.\nabla\phi(r,E,t)-\sum\nolimits_{t}(r,E)\,\phi(r,E,t)=$$

$$\int_{0}^{\infty}dE'\int_{4\pi}d\Omega'\sum\nolimits_{s}(\vec{r},E'\;\longrightarrow E,\Omega'\longrightarrow\Omega)(\vec{r},E',t)+\frac{\chi(\mathrm{E})}{4\pi}\int_{0}^{\infty}dE'\nu\,(E')\sum\nolimits_{f}(\vec{r},E')$$

$$\phi(r,E,t)+S(\vec{r},E,\mathrm{t}) \tag{3}$$

The terms in the equation are respectively;

- $\upsilon(E)$: The speed of the neutron according to its energy level.
- $\phi(r,E,t)$: The neutron density at a given location (r), energy (E) and time (t).
- $\vec{\Omega}$: Unit vector showing the direction of motion of the neutron.
- $\sum_t(r,E)$: Total macroscopic cross-sectional area. It indicates how far a neutron can travel through a material, and this includes the neutron's absorption and scattering probabilities.
- $\sum_s(\vec{r},E'\;\longrightarrow E,\Omega'\longrightarrow\Omega)(\vec{r},E',t)$: Scattering macroscopic cross-sectional area. This term describes the relationship between the energy and the states of a neutron before and after it changes direction.
- $\chi(\mathrm{E})$: Energy dependent spectrum of fission neutrons. It expresses the energy distribution of neutrons resulting from fission.
- $\nu(E)$: Average number of neutrons produced per fission. It shows how many neutrons are produced in each fission reaction.
- $\sum_f(\vec{r},E)$: Fission macroscopic cross-sectional area. This indicates how often fission events occur at a particular energy level.

- $S(\vec{r}, E, \mathrm{t})$: External source term. It refers to external neutron sources, and this can include neutrons coming from outside the reactor or neutrons produced from other sources within the reactor.

In this equation, ϕ angular neutron flux, $\vec{r}$ position of neutrons, E energy of neutrons (eV), Ω direction of neutrons, *t* time (sec), ∑ the macroscopic cross section of material, and S is the neutron sources entering the system.

4.3. MONTE CARLO N-PARTICLE CODE

MCNP, an acronym for Monte Carlo N-Particle, is a comprehensive and extensively employed computer software designed for simulating nuclear processes. MCNP, originally developed by Los Alamos National Laboratory, is a versatile Monte Carlo radiation transport code specifically designed to accurately simulate the motion of various particles, such as neutrons, photons, electrons, or coupled modes, across a wide range of energy levels. Here are the key features of MCNP [16-50]:

- The MCNP software utilizes the Monte Carlo simulation method, which is a statistical technique used to address physical problems. Random sampling is employed in this method to simulate intricate physical interactions, which is particularly advantageous in situations when deterministic solutions are challenging or unattainable.
- The software has the ability to simulate the transportation and interaction of several types of particles, such as neutrons, photons (gamma rays), and electrons, over a wide range of energy levels. The neutron energy ranges for all isotopes in ENDF/B-V and ENDF/B-VI are 10-11 MeV to 20 MeV, and for some isotopes it reaches 150 MeV [72].
- MCNP is employed in diverse domains like nuclear engineering, medical physics, radiation protection, and radiography. The uses of radiation encompass a wide range of fields, including the construction of nuclear reactors and radiation shielding, as well as medical radiation therapies and imaging systems.

- The capability of modeling elaborate geometries and different materials enables users to develop realistic models of nuclear reactors, radiation detectors, and other complex systems.
- MCNP relies on comprehensive nuclear data libraries that store detailed information regarding particle interactions and nuclear processes. These libraries are essential for precise simulations. It utilizes the built-in continuous energy nuclear and atomic data libraries, such as the Evaluated Nuclear Data File (ENDF) system [73].
- MCNP is renowned for its high precision and reliability in simulations, because to its extensive physics models and data libraries. This makes it a highly trusted tool in both research and industry.

In the thesis study, two different codes were used carry out the simulation study of the hybrid nuclear reactor. As the first code, an input file compatible with the work in MCNP was created. In this input file, the required tallies within the scope of reactor geometry, materials and neutronic performance constitute the input file. After the necessary input files are created in MCNP, the code is run and the output file is obtained. The resulting output file is introduced as an input file to the created interface program. The MCNPAS (MCNP Assessment Code) interface program, written in FORTRAN 90 language, is run for time-dependent analyzes and observation of changes in isotope amounts. The actinide isotope transformation schema and the flow chart of the interface code used to process MCNP output via MCNPAS shown in Fig 4.1. and 4.2. respectively.

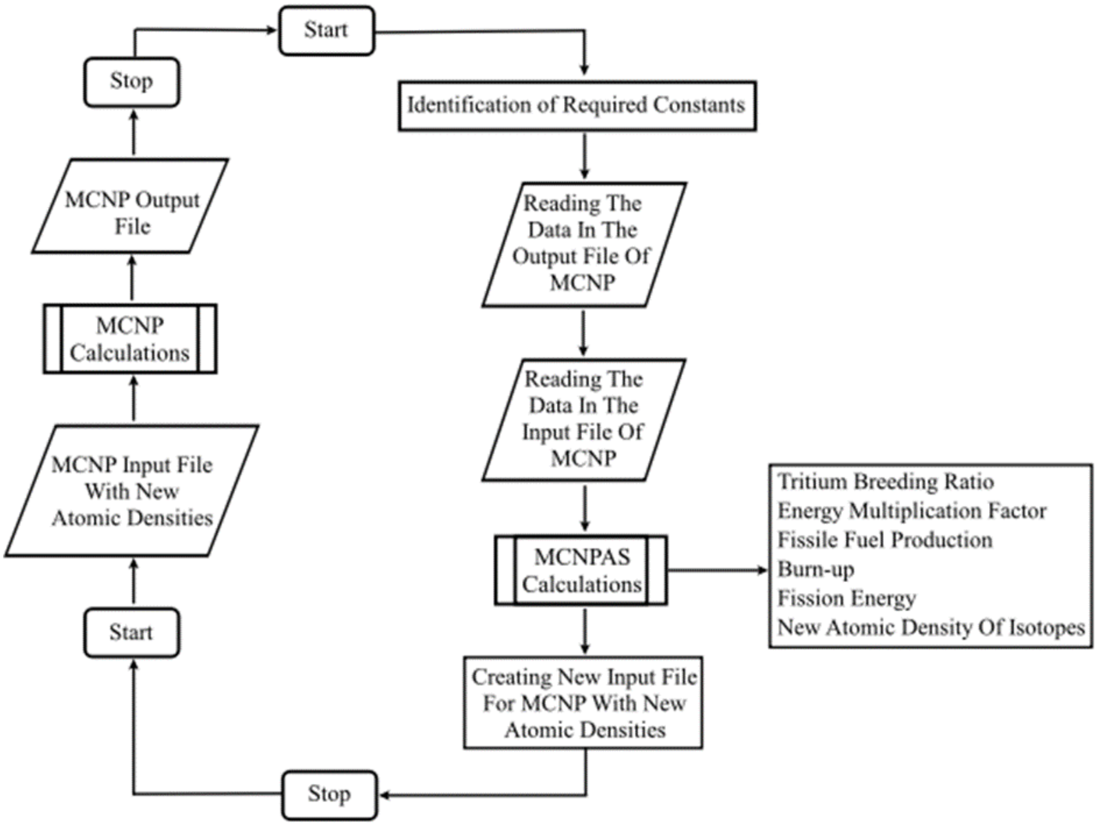

Figure 4.1. Flow chart of the interface code for MCNPAS [16].

The primary function of the interface code is to track the alterations in isotopes that occur as a result of nuclear reactions and radioactive decay. The interface code is executed sequentially using the Windows batch file. The initial stage involves the computation of neutron fluxes and reaction rates within the cells and surfaces using the MCNP code. In the second phase, the interface code retrieves necessary data from the output and input files of the MCNP. Subsequently, the interface code computes neutronic performance parameters and generates a new input file for the MCNP.

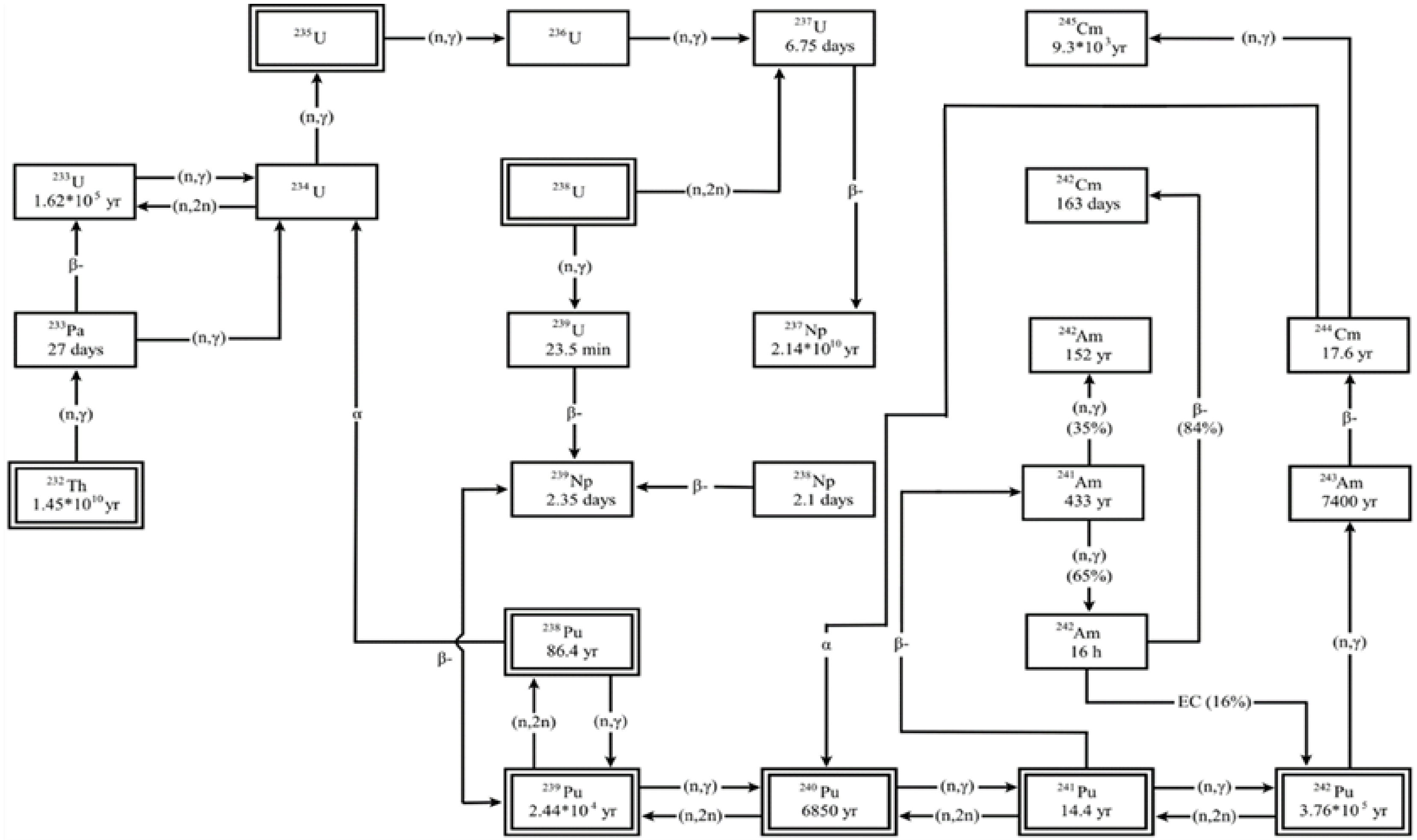

Figure 4.2. Nuclear transformation processes of actinides in nuclear reactors [9].

CHAPTER 5

GEOMETRICAL MODEL OF HYBRID NUCLEAR REACTOR

In this part of the thesis study, the geometry, components, and atomic densities of the selected materials of the hybrid nuclear reactor, the neutronic performance of which is being studied, will be shown. As mentioned in the literature review of the thesis, many designs are related to hybrid reactors. In this study, the geometry of ITER was taken as a reference, and the hybrid reactor blanket structure was used in the fusion reactor geometry in this project. The simulations were completed with MCNP and the generated interface code, and the effect of the changing first wall material, coolant, and fuel mixture ratios in the reactor's blanket structure on the neutronic performance of the reactor were investigated.

5.1. Blanket Structure of Hybrid Reactor

Within the scope of this study, fusion and hybrid reactor geometries, examples of which are available in the literature, were examined. Figure 5.1 shows the blanket structure of the hybrid reactor modeled within the scope of the thesis study. In the modeled reactor, the plasma, the neutron source, is adapted to the D shape geometry, and the blanket structure surrounds the plasma chamber. The reactor layers are listed below;

- First wall,
- Coolant zone,
- Fuel zone,
- Reflector,
- Vacuum chamber,
- Thermal shield,
- Magnetic coil.

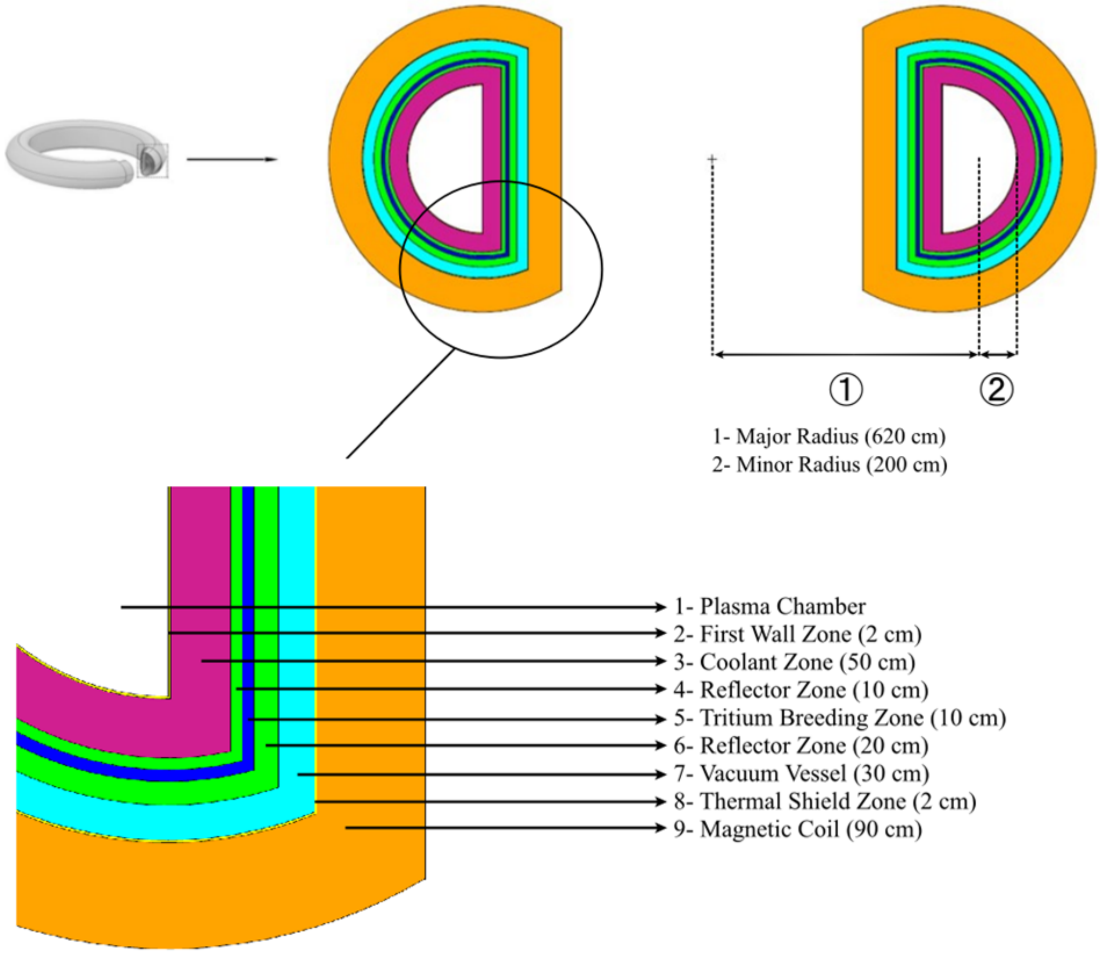

Figure 5.1. Blanket Structure of Hybrid Reactor Blanket.

5.1.1. Plasma Chamber

The plasma chamber is where plasma, formed due to deuterium-tritium (D-T) fusion reactions, is confined. High-energy neutrons resulting from D-T reactions within the plasma reach the first wall. For modeling purposes, these neutrons are represented as a D-shaped neutron source within the plasma chamber. The plasma region in this model assumes a uniform plasma distribution with minor and major radii of 200 cm and 620 cm, respectively. The nominal fusion power for the FFHR Tokamak design concept is 500 MW. The model incorporates the first wall as a reactor component in order to retain the low aspect ratio and produce a 14.1 MeV neutron flux of 2.22×10^{14} $n/cm^2/s$.

5.1.1. First Wall Zone

The selection of the first wall material is a highly important design component in the hybrid reactor blanket construction. The chosen material must possess great resistance to both elevated temperatures and intense radiation. The ideal first wall material should possess excellent hardness and corrosion resistance capabilities, while also having components with a minimal absorption cross-section to ensure that it does not adversely affect the neutron balance of the reactor. The thesis study implemented the solid first wall idea and provided a list of the first wall materials employed, along with their respective atomic densities, in Table 5.2.

5.1.2. Coolant Zone

Hybrid reactors employ a two-cycle coolant system. The coolant in the initial cycle facilitates the transfer of heat produced in the reactor layers to the subsequent coolant cycle, thereby converting it into usable energy. The coolant in the initial cycle serves as the medium for tritium synthesis. When selecting the coolant material for this layer, the following criteria should be taken into account:

- High lithium density, specific heat, and thermal conductivity,
- Low density, viscosity, corrosion and erosion effect,
- High boiling and low melting points,
- Low neutron absorption cross-section,
- Low cost.

5.1.3. Fuel Zone

Fissile and fertile fuels are present in the fuel zone of hybrid reactors. The fissile and fertile fuels in this region utilize the neutrons emitted by fusion reactions as an energy source. The hybrid reactor concept employs fusion processes to initiate the reactor. The initiation of the fission reaction is contingent upon the presence of fissile fuel in the fuel zone, which varies depending on the type of material. If there is a fuel that is capable of supporting the growth of plants, this fuel undergoes a transformation into a fuel that is capable of sustaining a nuclear chain reaction. The study favored the utilization of the molten salt approach, which involves a combination of a liquid fertile material and coolant mixture. This choice guarantees enhanced safety for the reactor. In the research conducted, ThF_4 and UF_4 were utilized as the fuel sources. These fuels were amalgamated with the coolant in proportions corresponding to their respective eutectic points.

5.1.4. Reflector Zone

A reflecting layer in nuclear reactors enhances the likelihood of capturing neutrons that remain unreacted in the coolant and fuel region. The purpose of this layer is to scatter and decelerate fast neutrons, so preventing their escape and enhancing tritium generation, while also redirecting non-reacting neutrons towards the coolant and fuel region. While graphite is commonly utilized in this area, the literature has also documented the utilization of TiC, ZrC, and ZrH_2 materials. Given the expenses associated with these materials, the thesis study utilized graphite in the reflective zone.

5.1.5. Thermal Shield Zone

The protection of the superconducting magnets against excessive nuclear heating, radiation damage, and neutron fluence is the most important shielding need. There are three main reasons for the emergence of this need. In the first instance, the radiation exposure of diagnostic tools and remote maintenance equipment must also fall within established limitations. Any need for manual labor inside the cryostat will result in a further decrease in plasma neutron flux. Secondly, the coil and stabilizer must be protected from fast neutron fluence and displacement damage, as well as the overall radiation dosage to the insulation of the coil winding, with suitable radiation attenuation. It is imperative to safeguard the magnet layer against temperature variations. Hence, the superconducting magnets utilized in TOKAMAK reactors operate at a cryogenic temperature of 4 °K. For every watt of thermal energy that is deposited by neutrons and secondary gamma rays in the magnets, it requires approximately 500 watts of refrigeration energy to eliminate the extra heat [70].

5.2. Atomic Densities of Materials Used in Hybrid Reactor

The atomic densities of the components in the hybrid reactor structure are calculated by equations 1 and 2. With these calculations, according to the information in the literature, the atomic densities of the materials in the MCNP input file and the mass and volumetric percentages of this mixture are calculated. Table 5.1. the atomic densities of the materials used in the hybrid reactor are listed.

Table 5.1. Atomic densities of materials used in hybrid reactor [74].

Reactor Zone	Material	Isotopes	Density (gr/cm^3)	Atomic Density (atom/b.cm)
First Wall & Thermal Shield	SS 316 LN-IG	Fe	8.0	5.65101E-02
		C		4.01099E-05
		Mn		1.57845E-03
		P		3.11074E-05
		S		1.50268E-06
		Si		6.86146E-04
		Ni		1.00948E-02
		Cr		1.61217E-02
		Mo		1.20515E-03
		N		2.06424E-04
		Nb		1.03709E-05
		Cu		2.27438E-05
		Co		2.45240E-05
		B		4.01095E-06
		Ta		2.66242E-06
		Ti		1.00576E-05
Coolant	Coolant & Fuel Mixture Candidate			Seen in Table 5.2
Reflector	Graphite	C	2.267	1.13661E-01
Vacuum Chamber	S30476 (60%)	Fe	7.8	5.24159E-02
		C		7.82143E-05
		Mn		1.39364E-03

		Cr		1.74350E-02
		B		9.12492E-03
		Si		8.86414E-04
		Ni		1.12828E-02
	H_2O (40%)	H	1	6.68566E-02
		O		3.34283E-02
Magnetic Coil	Nb_3Sn (45%)	Nb	8.4	1.71828E-02
		Sn		5.72759E-03
	Al_2O_3 (5%)	Al	3.987	2.35482E-03
		O		3.53223E-03
	Incoloy 908 (50%)	Fe	8.17	1.79278E-02
		Ni		2.05348E-02
		Cr		1.88298E-03
		Nb		7.73161E-04
		Ti		8.93607E-04
		Al		8.47909E-04
		N		3.51351E-06
		Mn		1.83588E-05
		C		2.04811E-05
		Co		4.17419E-05

Table 5.2. Atomic densities of the candidate coolant and fuel mixture materials [74].

Reactor Zone	**Material**	**Isotopes**	**Density (gr/cm^3)**	**Atomic Density (atom/b.cm)**
Coolant	Candidate 1 LiF (78%) - ThF_4 (22%)	Li-6	4.08	1.06751E-02
		Li-7		1.06751E-02
		F		4.59742E-02
		Th		6.17564E-03
Coolant	Candidate 2 LiF (73%) - UF_4 (27%)	Li-6	4.42	3.74365E-03
		Li-7		1.49746E-02
		F		2.76927E-02
		U-235		5.02484E-05
		U-238		6.87293E-03

5.3. Lithium Fluoride

In a hybrid nuclear reactor structure, lithium fluoride serves a dual purpose: as a coolant and a material for producing tritium. It facilitates the generation of tritium, which is crucial for sustained fusion events, by initiating nuclear reactions with Li-6 and Li-7 isotopes. LiF's fluorine content forms fluoride molten salt, notable for its low vapor pressure, excellent heat transfer characteristics, including high thermal conductivity and specific heat capacity, strong resistance against radiation, and non-reactivity with structural materials. These properties underpin LiF's preference as a coolant and tritium-producing material in hybrid reactor designs.

5.4. Thorium Tetrafluoride

Thorium tetrafluoride (ThF_4), a compound resulting from the reaction of thorium with fluorine gas, is synthesized through hydrofluorination of thorium oxide in gas phase using anhydrous HF. As a fertile element, thorium requires neutron absorption during the fuel cycle to transform into the fissile isotope U-233. This transformation occurs through neutron capture followed by beta decay. Thorium reacts with both thermal and fast neutrons for fuel production. Compared to the U-238 isotope, thorium is approximately four times more abundant in nature and has a higher fissile fuel production capability. Thorium fuel cycles do indeed produce less plutonium and other transuranic elements compared to uranium fuel cycles. Thorium possesses an essential advantage in that transuranic elements play a significant role in the persistence of radioactivity in nuclear waste over extended periods of time. The thorium fuel cycle predominantly generates U-233, which exhibits a lower quantity of transuranic byproducts compared to the U-235 or U-238 fuel cycles commonly employed in uranium-based reactors. Thorium demonstrates its versatility in nuclear reactors, as it can exist in either a solid state or as a molten salt, depending on the particular reactor technology utilized. The ability of thorium to change its physical state is of utmost importance in determining its behavior under nuclear circumstances. For example, molten salt fuel, which is a prevalent type of thorium, is considerably less susceptible to neutron or gamma irradiation, indicating a decrease in radiation damage. However, it is not completely impervious to these effects. The durability of molten salt thorium fuels allows for the straightforward change of fuel composition by simply adding thorium-containing molten salt, hence facilitating the maintenance of ideal conditions. Furthermore, the operational effectiveness of thorium-based molten salt reactors is additionally improved by the capability to eliminate inert gases such as xenon and krypton, which do not undergo chemical reactions or dissolve in the molten salt. The gases can be effectively isolated from the reactor system, collected by a filtration system, and securely stored for decomposition. The selection of fuel components in these reactors is additional evidence of thorium's adaptability. Generally, these fuels are created as compounds containing either fluorine or chlorine, with fluorine-based compounds being more common. The preference for employing thorium in various compound forms for nuclear power generation is due to the greater stability and superior neutronic performance of fluorine compounds. This highlights the practical benefits of utilizing thorium in nuclear power production.

5.5. Uranium Tetrafluoride

Uranium tetrafluoride (UF_4) is produced by reacting uranium with fluorine gas. This product can be manufactured by hydrofluorinating uranium dioxide or other uranium compounds using anhydrous hydrogen fluoride (HF). Uranium, specifically in its UF_4 form, is an essential element in several nuclear fuel cycles and has a vital function in maintaining nuclear reactions. Uranium has the ability to capture neutrons and undergo fission, resulting in the release of a substantial amount of energy. This technique is important to the functioning of numerous nuclear reactors. In contrast to thorium, natural uranium consists of a substantial amount of the fissile isotope U-235, but the majority is U-238. Uranium-238 undergoes a process of beta decay after capturing a neutron, resulting in the formation of the fissile isotope plutonium-239. This transformation significantly enhances the fuel's reactivity. The conversion process is a crucial component of the breeding cycle in reactors specifically engineered to generate additional fissile material. Uranium tetrafluoride is noteworthy for its involvement in the uranium-plutonium fuel cycle, a widely utilized process in commercial nuclear power generation. Although the thorium cycle is praised for its reduced production of plutonium and other transuranic elements, the uranium-plutonium cycle is widely recognized and has been thoroughly researched and employed in current nuclear reactor designs. A significant obstacle related to UF_4 and the uranium-plutonium cycle involves effectively handling plutonium and other transuranic elements that are generated as secondary products. The inclusion of these components plays a substantial role in the extended radioactivity of nuclear waste, hence emphasizing the importance of effectively managing and disposing of them in the planning of the nuclear fuel cycle.

5.6. Molten Salt-Fuel Mixture

In the reactors, the mixture of molten salt (lithium, beryllium, and zirconium fluoride) and fuel (thorium, uranium, plutonium, and MA) can be utilized as coolant as well as fuel. Fluorides in the molten salt mixture coolant are also chosen to contribute to a small tritium reproduction rate because the reactor architecture in this study is a hybrid reactor. The fusion-fission hybrid reactor's blanket structure's high-temperature operating characteristics will result in an increase in thermal efficiency if molten salt fuel is employed in it. Importantly, because of its chemical and thermophysical characteristics, the molten salt fuel mixture has a structure that can hold on to fission products in the case of an accident. It can also separate the fuel's fission products from the fuel and has an easier time loading fuel. Fluorine-containing fuel compounds are far more prevalent because of their improved neutronic performance and increased stability. Using a molten salt fuel mixture in the fusion reactor will make it safer and more environmentally beneficial in light of these aspects. When determining the molten salt-fuel mixture, burn-up and thermodynamic features should be taken into account in addition to chemical and neutronic concerns. One of the primary considerations when assessing reactor fuel is the salt's melting point. Salt freezing and issues with corrosion of structural components are less likely when the salt-fuel mixture has a low melting point since this lowers the operating temperature of the reactor [73]. A eutectic mixture, characterized by its lowest melting point, was selected to enhance the cooling efficiency of the hybrid reactor blankets. Considering their thermophysical properties, the LiF-ThF_4 and LiF-UF_4 fuel mixtures were chosen based on their respective ratios at the eutectic point. Furthermore, the Newtonian flow properties of this molten salt-fertile fuel mixture facilitate the straightforward application of thermohydraulic calculations, including the conservation of mass, the

Navier-Stokes equations, and the conservation of energy equations. [76] . In Figures 5.2. and 5.3. the phase diagram of LiF-ThF_4 and LiF-UF_4 were shown.

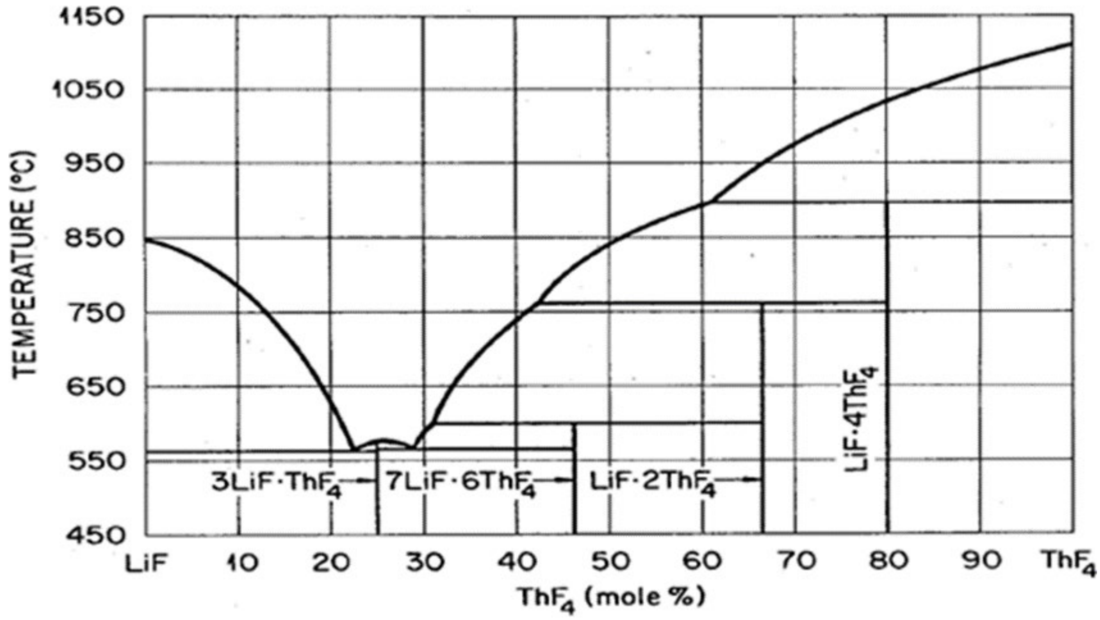

Figure 5.2. Phase diagram of LiF-ThF_4 [77].

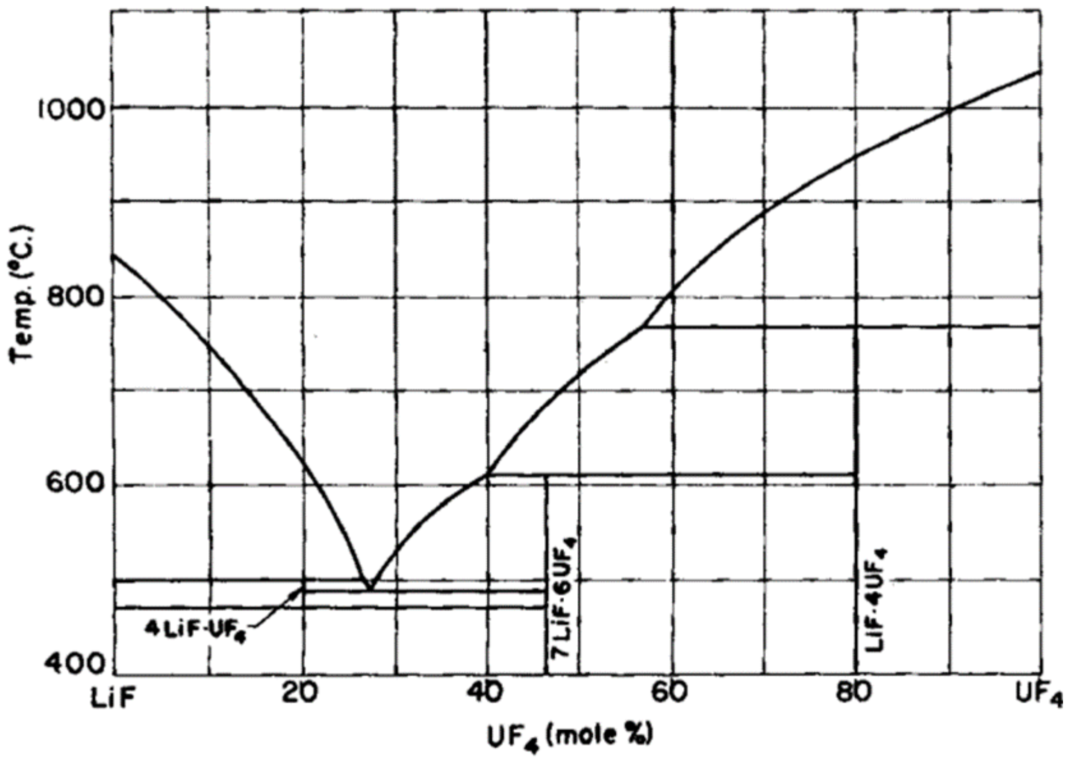

Figure 5.3. Phase diagram of LiF-UF_4 [78].

CHAPTER 6

NUMERICAL RESULTS AND DISCUSSION

In this section, the numerical results that form the basis of the research are presented in detail. This part of the study includes a meticulous analysis of the modeling and simulation processes and a comprehensive evaluation of the data obtained from these analyses. Additionally, in discussion section, these results were compared with existing findings in the literature and the contributions of the study were evaluated from both theoretical and practical perspectives. In interpreting the numerical analyses, emphasis was placed on the meaning and impact of the findings in the broader context of the field. Finally, this chapter will also focus on potential limitations of the study and how the results may guide future research.

6.1. Tritium Breeding Ratio

The tritium production rate is the quotient of the amount of tritium generated in the blanket layer of hybrid nuclear reactors divided by the amount of tritium utilized in the plasma. Given that tritium is not naturally occurring, it is necessary to produce tritium within the blanket structure of the hybrid reactor. The tritium production rate must be at least 1.05, considering the decay time of tritium and other potential losses in the reactor. [28]. In hybrid nuclear reactors, tritium production takes place through the interaction of thermal and fast neutrons with lithium isotopes. These reactions include the exothermic ^{6}Li(n,T) reaction with thermal neutrons and the endothermic ^{7}Li(n,T) reaction with fast neutrons. Consequently, the ^{6}Li isotope plays a more significant role in the TBR, and the impact of lithium enrichment on the TBR value depending on coolant thickness is depicted in Figure 6.2. and 6.3.

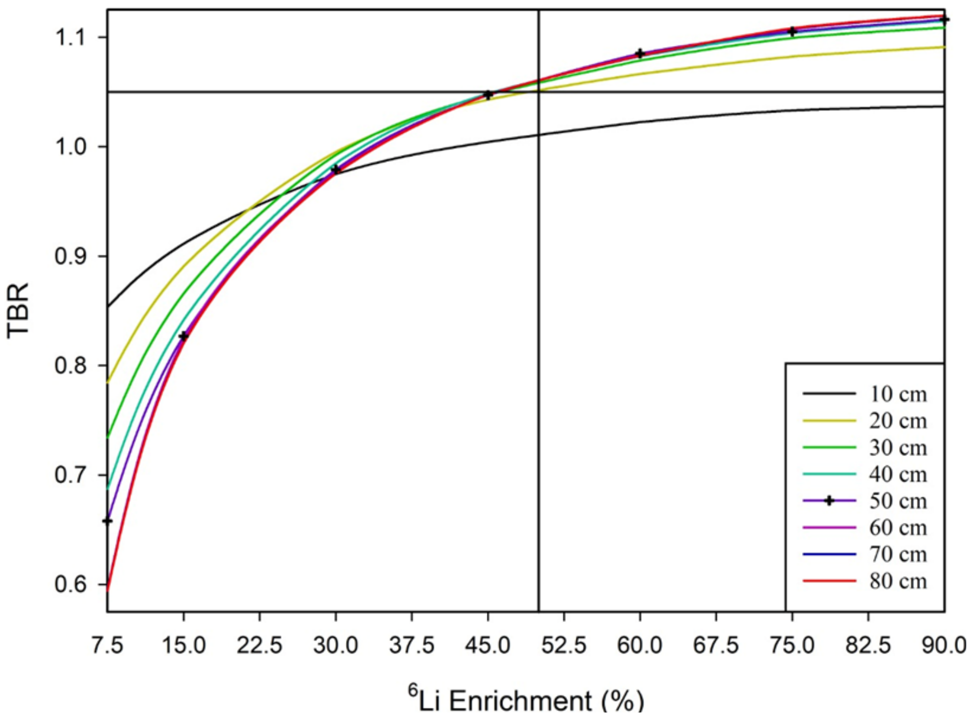

Figure 6.1. Effect of ^{6}Li enrichment on TBR depending on coolant thickness ThF_4 – LiF mixture

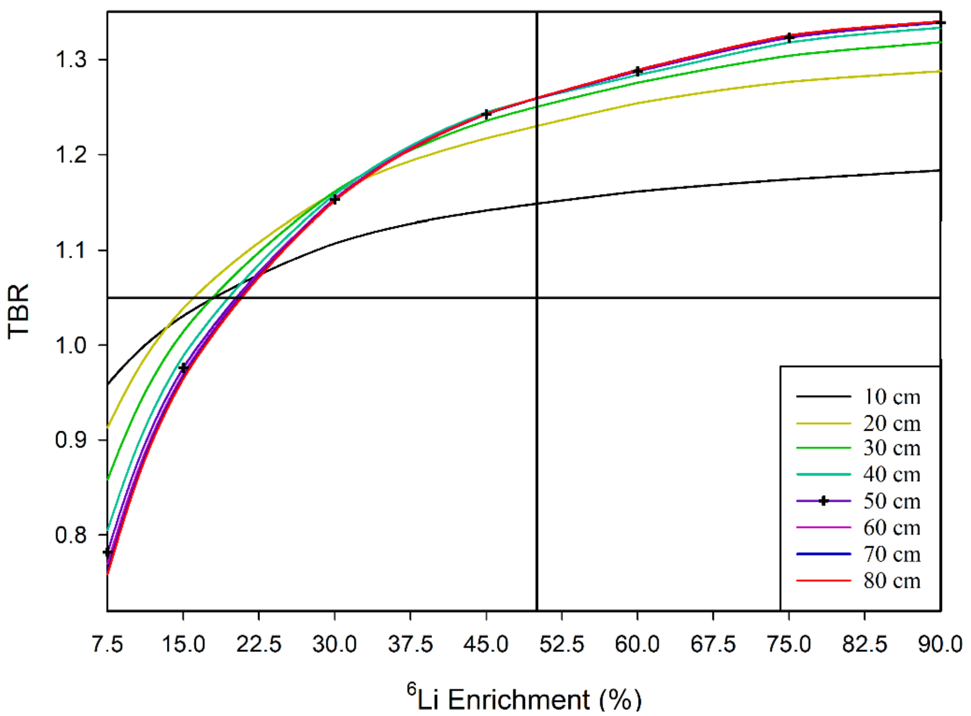

Figure 6.2. Effect of ^{6}Li enrichment on TBR depending on coolant thickness for UF_4 – LiF mixture

6.2. Energy Multiplication Factor

The energy multiplication factor is a measure of the ratio between the energy input and energy output in the reactor. The energy inputs in the reactor can be accounted for in the calculations by considering the energy consumed by the magnets used to control the plasma and the energy consumed by other subcomponents of the reactor. The reactor's energy outputs can be determined by quantifying the electrical energy, waste heat, and radiation emitted as a consequence of fusion and fission reactions. In order to counteract the decrease in reactor power and generate additional power, the energy output of the reactor must surpass the energy input. The minimum requirement for the energy multiplication factor to be reached is 1.5 [79]. Considering the energy released from fission and fusion reactions, energy multiplication factor calculated with following equation;

$$M = \frac{200 * \langle \sum F * \Phi \rangle + 4.786T_6 - 2.467T_7 + 14.1}{14.1} \quad (4)$$

Here, $\sum F$ is total fission reaction rate, T_6 and T_7 tritium breeding reaction contribution from ^{6}Li and ^{7}Li isotopes respectively.

In hybrid reactors, the amount of fission reaction and fission fuel formation are directly proportional, and the same proportion is valid for the energy multiplication factor. The amount of exothermic $^{6}Li(n,\alpha)T$ reaction included in the equation increases the energy multiplication factor and releases 4.786 MeV of energy per reaction. The endothermic $^{7}Li(n,n\alpha)T$ reaction consumes 2.467 MeV energy per reaction and negatively affects the energy multiplication factor value. The energy multiplication factor values obtained as a result of the calculations are shown in figures 6.3. and 6.4.

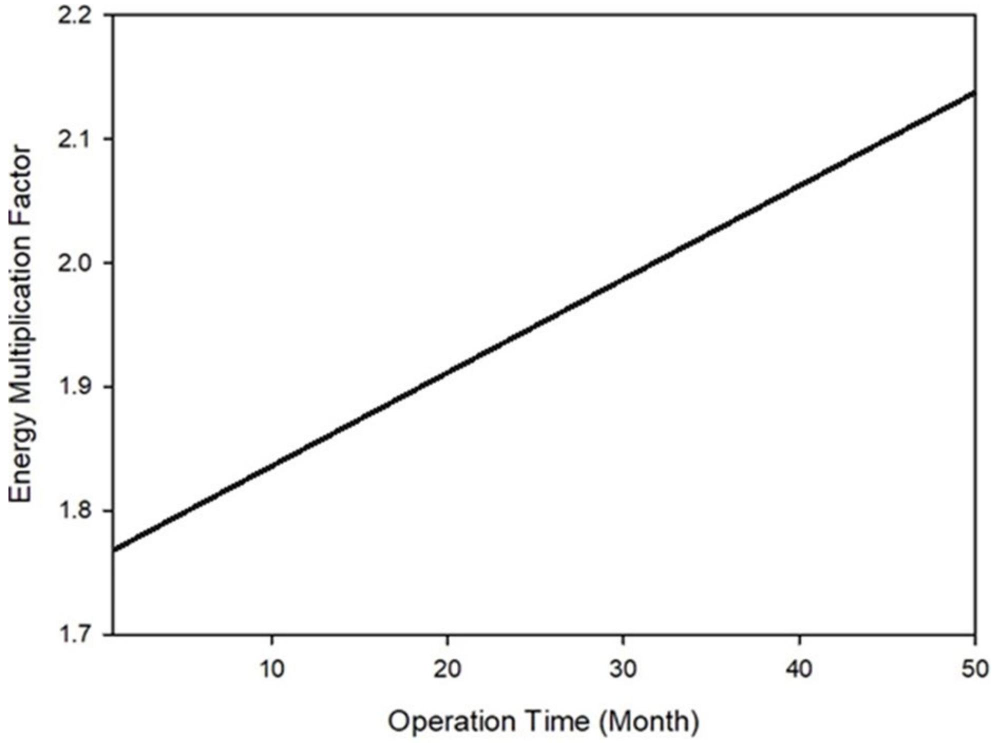

Figure 6.3. Energy multiplication factor for ThF_4 – LiF mixture

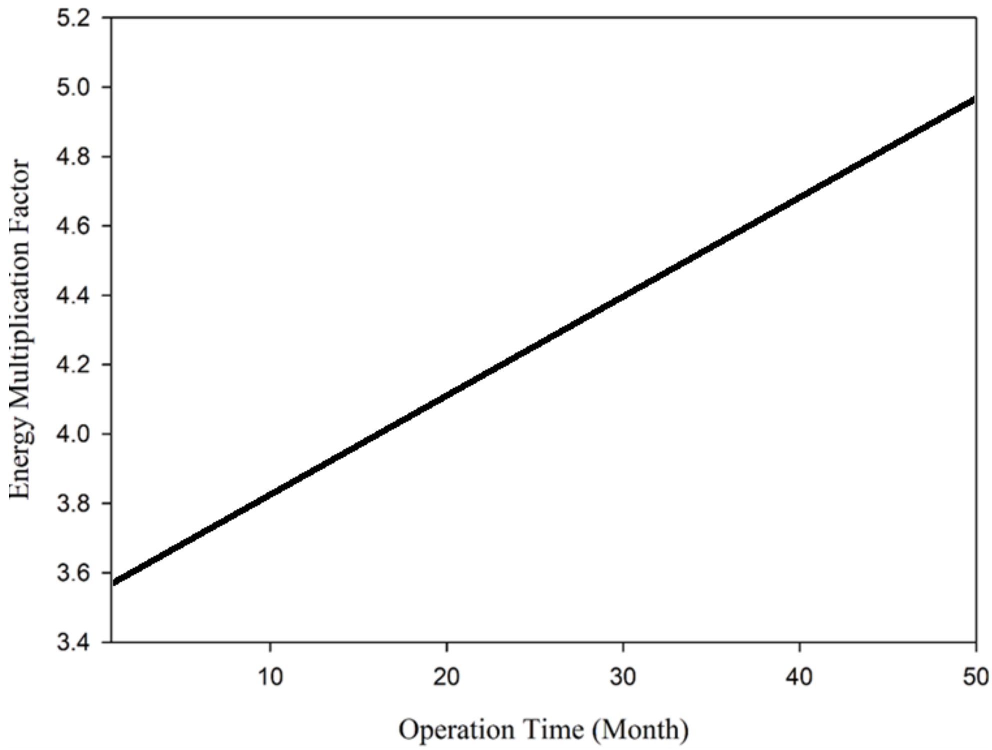

Figure 6.4. Energy multiplication factor for UF_4 – LiF mixture

6.3. Radiation Damage

High energetic neutrons released in fusion reactions react on the first wall of the reactor and cause radiation damage. Helium and hydrogen gas are created in the first wall material as a result of the reactions that have taken place. High working temperatures cause hydrogen gas to escape from the structure of the material, but helium gas stays inside. Helium gas causes the first wall material to lose its qualities over time, which in turn causes the material's lifespan to decline dramatically. Therefore, it is essential to investigate the production of helium gas in the first wall. More specifically, in order to compute the replacement period of the first wall material, the upper production value of helium gas was established as 500 appm, and the calculations were carried out in accordance with one full power year (FPY) [80]. In the first wall material, the term "DPA" refers to the movement of atoms inside the structure of the material as a result of the action of high-energy neutrons and helium isotopes that are produced from the plasma. This parameter, which is used to measure radiation damage on the first wall material of the reactor, should be at as low levels as possible to avoid deterioration and loss of function of the first wall material. Within the scope of the study, the upper limit value of DPA was determined as 100 appm [28-80]. The priority of evaluating damage parameters is determined by the He/DPA ratio. The production of helium gas precedes the displacement of atoms when the ratio between He and DPA (500/100 appm) is greater than 5, the exchange period of the first wall is computed using the production of helium gas. In table 6.1., radiation damage and structural lifetime of first wall values are shown.

Table 6.1. Radiation damage in the first wall

Parameter / Mixture	H/FPY (appm)	He/FPY (appm)	DPA/FPY (appm)	He/DPA	Structural Lifetime (year)
ThF_4 - LiF	331.39	126.98	11.55	10.99	3.94
UF_4 - LiF	335.13	127.51	12.43	10.26	3.92

6.4. Fissile Fuel Production

There are two main methods for fissile fuel production, enrichment and breeding respectively. Breeding is the process of creating fissile isotopes from non-fissile isotopes. This is done by exposing a non-fissile isotope to neutrons in a nuclear reactor. In the thermonuclear fusion reactor's core, neutrons are used as a high-intensity external neutron source for hybrid nuclear reactor. Neutrons released from fusion reactions in hybrid reactors are used to produce fissile fuels from fertile materials in addition to tritium and energy production.

The reserves of fissile fuels used in today's nuclear energy applications are quite limited. On the contrary, the amount of thorium and non-fissile uranium is quite high compared to these reserves. The isotopes in question, upon absorbing a neutron, form ^{233}U and ^{239}Pu isotopes, respectively. The ^{232}Th isotope undergoes a radiative capture process to yield ^{233}U. The half-life of the ^{232}Th isotope is 22 minutes, during which it decays to ^{233}Th upon capturing a neutron. The unstable isotope ^{233}Th decays to produce ^{233}Pa (protactinium). ^{233}Pa, which decays to fissile ^{233}U via beta decay with a half-life of 27 days, can be bred from ^{232}Th using high-energy neutrons [81]. Utilizing thorium as a fuel in a molten salt reactor necessitates the "thorium fuel cycle." In this cycle, ^{232}Th absorbs a neutron, transmuting into fissile ^{233}U. A similar fuel cycle is valid for uranium. The ^{238}U isotope captures neutrons to form the ^{239}U isotope, which turns into the ^{239}Np isotope with beta decay. ^{239}Np, which decays to fissile ^{239}Pu via beta decay with a half-life of 2.35 days. The synergistic use of thorium with fissile isotopes presents a viable strategy for achieving a sustainable and proliferation-resistant nuclear fuel cycle [82].

In this study, fuel composition using molten salt including thorium and uranium halides is used to produce fissile fuel. The results obtained are shown in Fig 6.6. and 6.7.

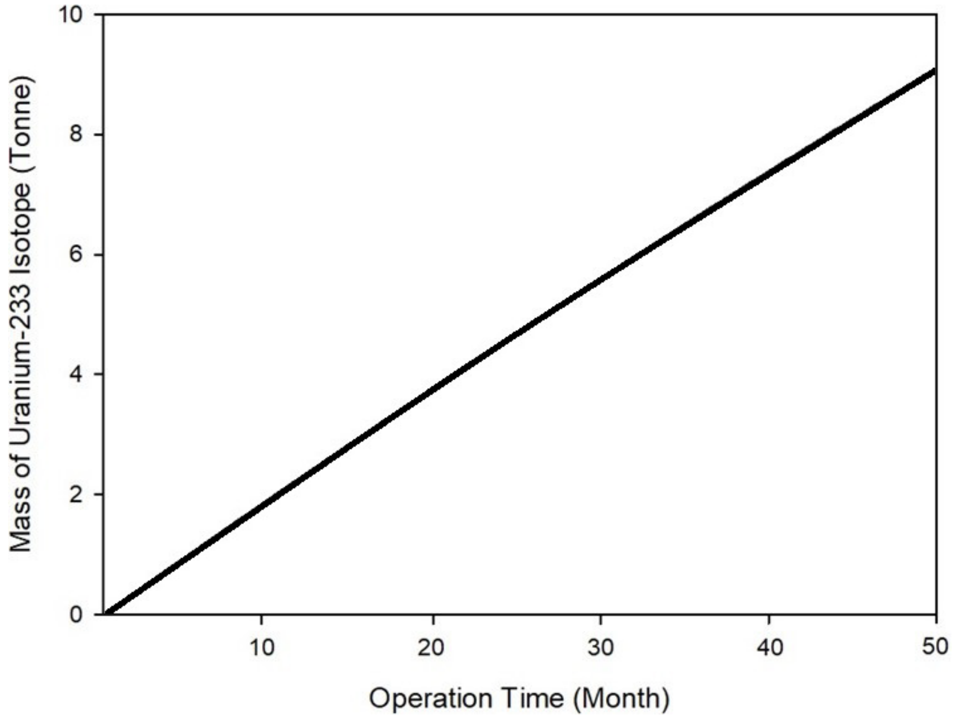

Figure 6.6. Change in the mass of U-233 isotope through nuclear transformations over fission reaction in LiF-ThF_4 molten salt-fuel mixture

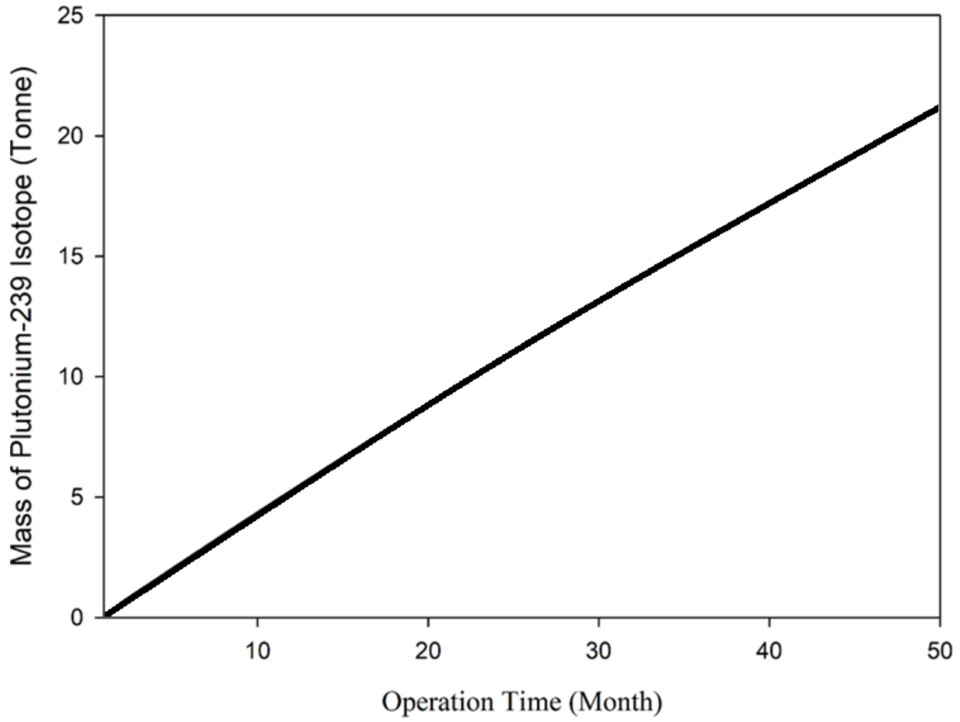

Figure 6.7. Change in the mass of Pu-239 isotope through nuclear transformations over fission reaction in LiF-UF_4 molten salt-fuel mixture

6.5. Fuel Burnup

The amount of energy recovered from nuclear fuel is measured by fuel burnup (also known as fuel utilization), which also serves as a measurement for fuel depletion in nuclear engineering. Burnup is an important factor in determining the efficiency and lifetime of nuclear fuel. How much fuel is consumed affects energy production capacity. High burnup values indicate that the fuel has the capacity to produce more energy, while low burnup values indicate that the fuel is capable of producing less energy. Burnup is expressed in gigawatt-days per ton of fuel (GW.day/Tonne) and the burnup values obtained in the study are shown in figures 6.8 and 6.9.

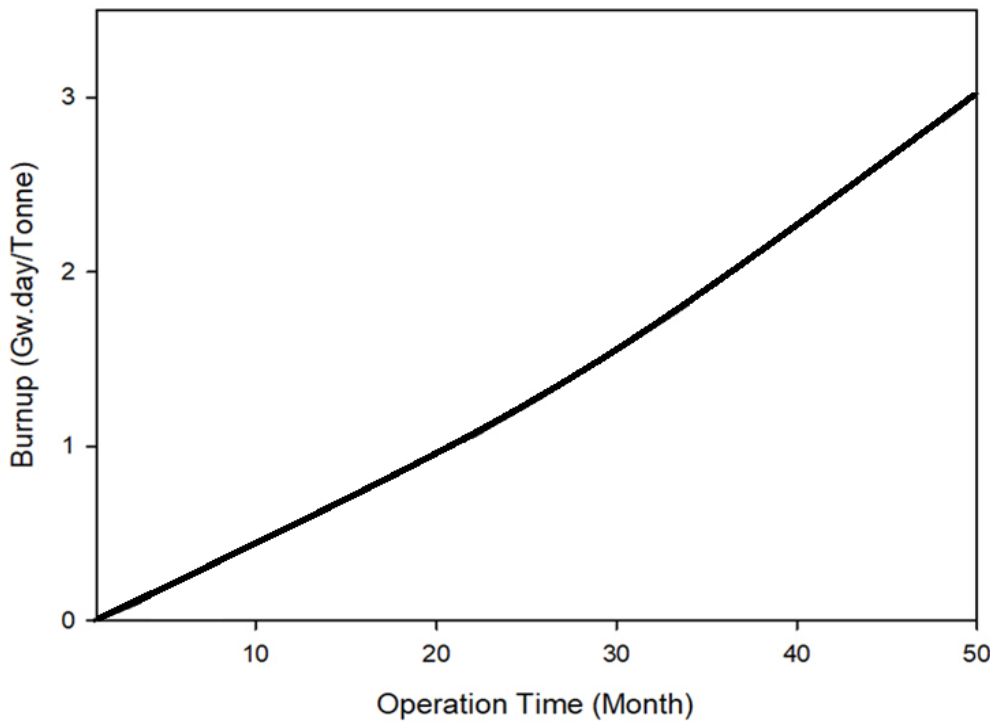

Figure 6.8. Burn-up of the fissile fuel as a function of operating time (LiF-ThF_4 molten salt-fuel mixture)

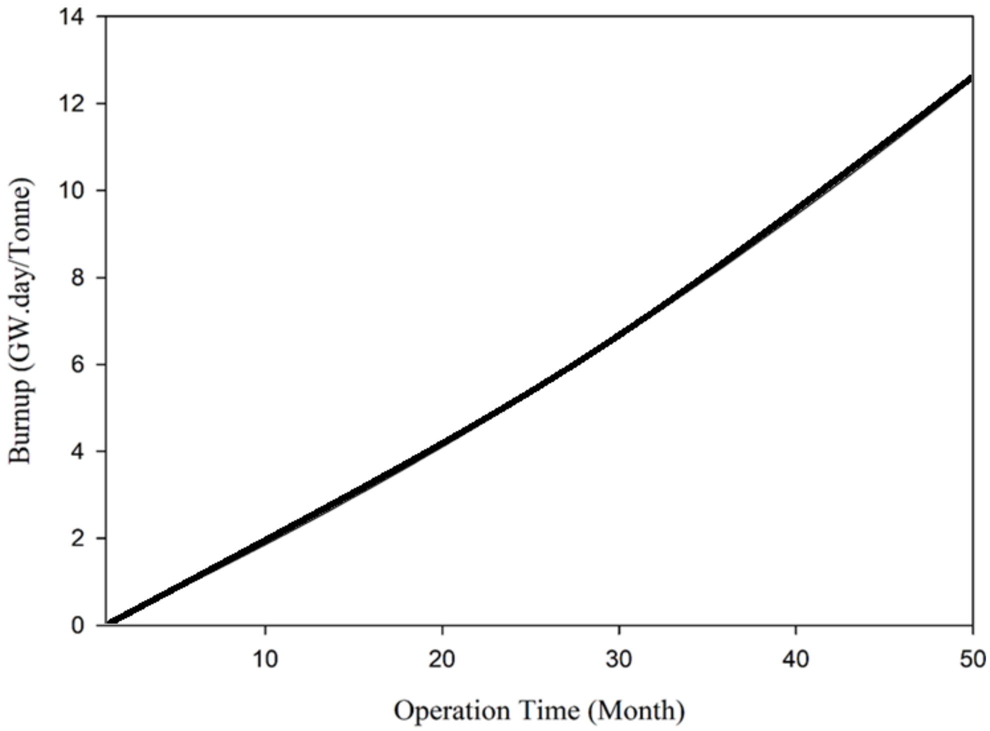

Figure 6.9. Burn-up of the fissile fuel as a function of operating time (LiF-UF_4 molten salt-fuel mixture)

6.6. Fission Power Generation

Fission power generation is the process of using the energy released from nuclear fission to generate electricity. In fission reactors, fissile fuel placed in a core and fission reactions occurs in this place. Contrary to this in hybrid nuclear reactors, fission reaction occurs in a blanket structure of the reactor. The fission process within the blanket contributes substantially to the overall power output of the hybrid reactor. The kinetic energy of fission products and released neutrons is rapidly converted into thermal energy through collisions with surrounding atoms in the blanket material. This heat then drives a thermodynamic cycle, typically involving a coolant, which in turn powers a steam turbine for electricity generation. Beyond direct heat generation, fission reactions in the blanket also release additional neutrons. These neutrons can trigger further fission events in a chain reaction, amplifying the overall power output. The blanket effectively acts as a neutron multiplier, enhancing the reactor's energy production and contributing to the breeding of additional fissile fuel. Fission power generation from reactor blanket calculated with following equation;

$$P_{fission} = PF\ x\ n_t\ x\ V_{FW}\ x\ CF\ x\ \iint \sum\nolimits_f (E)\emptyset(E)\ dEdV \qquad (5)$$

Here, PF is power factor, n_t number neutrons per unit time, V_{FW} volume of first wall, CF capacity factor, E and $\emptyset$, energy and neutron flux (neutron/cm^{2}s) respectively.

In figures 6.10 and 6.11. the fission power produced in the hybrid reactor blanket structure modeled in the study were shown.

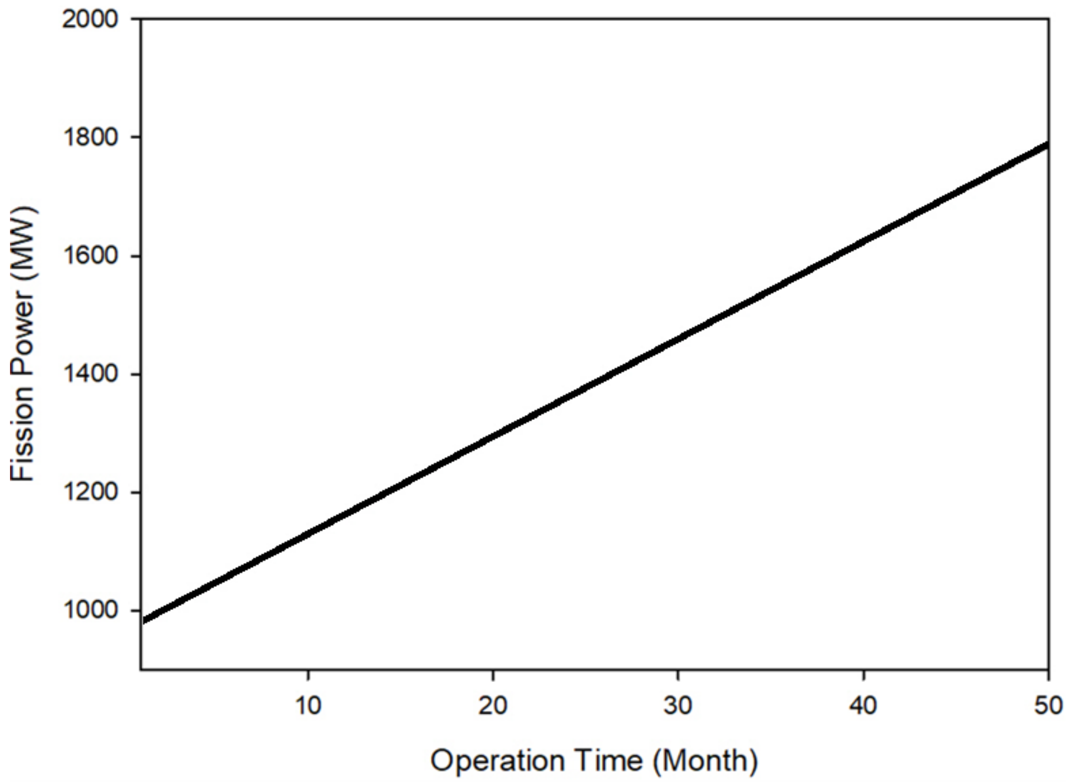

Figure 6.10. Fission power generation as a function of operating time (LiF-ThF_4 molten salt-fuel mixture)

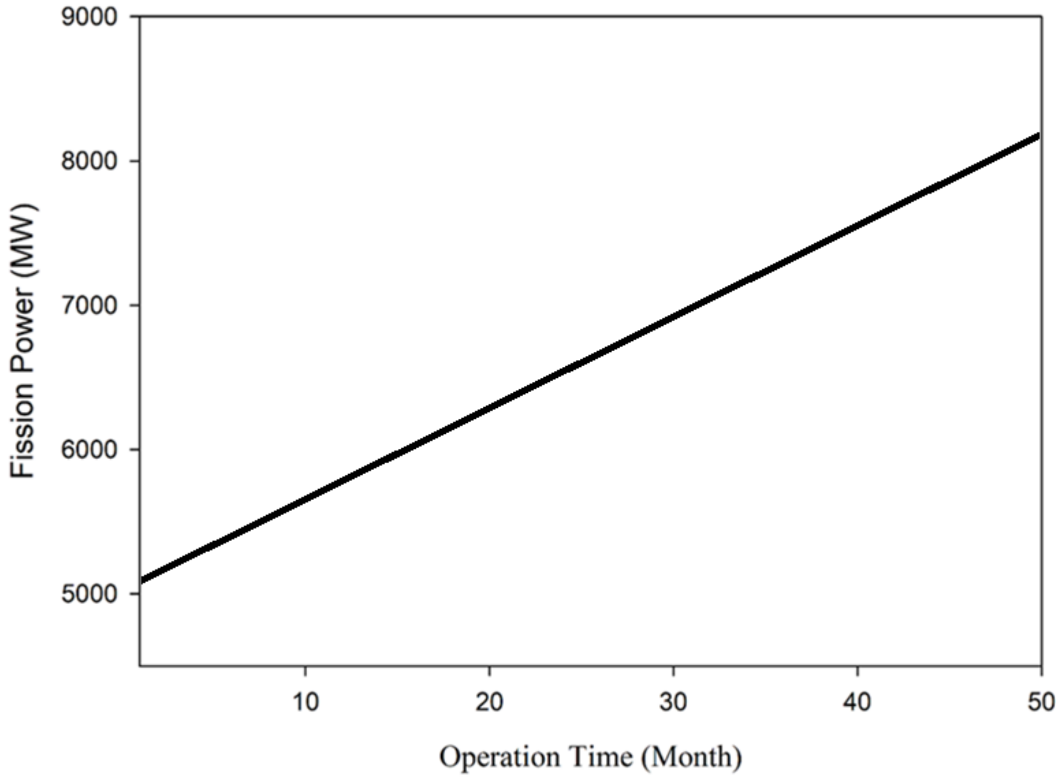

Figure 6.11. Fission power generation as a function of operating time (LiF-UF_4 molten salt-fuel mixture)

CHAPTER 7

CONCLUSIONS

This study explored a novel approach for a FFHR by utilizing a thorium-molten and uranium-molten salt mixture as both a coolant and a fissile fuel breeding medium. The combination of thorium and uranium with liquid fluoride salts, renowned for their exceptional coolant properties, showcased the potential for continuous fuel reprocessing within the FFHR. This enables efficient removal of fission products and sustained fissile material production. The study underscored the broader advantages of thorium and uranium fuel cycling, including enhanced safety features and reduced generation of radioactive minor actinides. The study aimed to demonstrate the feasibility, efficiency, and safety of utilizing thorium as a fuel and molten salt as a coolant in a FFHR. The inherent capability for continuous fuel reprocessing not only facilitated the removal of fission products but also ensured the sustained production of fissile material, thus enhancing the reactor's overall sustainability. This innovative approach, employing a thorium-molten and uranium-molten salt coolant for fissile fuel production, represents a promising avenue towards efficient and sustainable nuclear energy, offering potential benefits in terms of safety and reduced generation of radioactive minor actinides. In this study, a hybrid reactor design was developed, utilizing a D-T fueled plasma neutron source and referencing the ITER geometry. SS 316 LN-IG was chosen as the first wall material, while a eutectic mixture of $LiF\text{-}ThF_4$ and $LiF\text{-}UF_4$ was selected as the coolant, considering the material's eutectic points. Neutron calculations were performed using the MCNP5 nuclear code, incorporating the ENDF/B-VIII and CLAW-IV nuclear data libraries. The time evolution of isotopes within the reactor was calculated using the MCNPAS interface code. The study results were evaluated based on several key parameters, including tritium breeding ratio, energy multiplication factor, radiation damage, fissile fuel production, and fuel burn-up value.

The main results of this study can be summarized as follows for LiF-ThF_4 molten salt-fuel mixture:

- The 4-year operation history of total TBR value is calculated and always above 1.05 and increases with time.
- The changes in ^{6}Li and ^{7}Li mass decreased from 32.22 tonnes to 30.99 tonnes and from 89.21 tonnes to 89.20 tonne respectively.
- The energy multiplication factor (M) values obtained during the reactor operating time and M reached from 1.77 to 2.1 and has always been greater than minimum required value.
- The burnup value obtained in the study is 3 GWday/tonne at the end of 50 months.
- Th initially decreased from 631.3 tonnes to 587.2 tonnes, while ^{233}U production during this period was 9.1 tonnes.
- The first wall replacement period was calculated as 3.94 years.
- The change in fission power generation increased from 987 MW to 1762 MW.

The main results of this study can be summarized as follows for LiF-UF_4 molten salt-fuel mixture:

- The 4-year operation history of total TBR value is calculated and always above 1.05 and increases with time.
- The changes in ^{6}Li and ^{7}Li mass decreased from 18.68 tonnes to 18.25 tonnes and from 102.55 tonnes to 102.46 tonne respectively.
- The energy multiplication factor (M) values obtained during the reactor operating time and M reached from 3.57 to 5.2 and has always been greater than minimum required value.
- The burnup value obtained in the study is almost 13 GWday/tonne at the end of 50 months.

- ^{238}U initially decreased from 720.79 tonnes to 639.61 tonnes, while ^{239}Pu production during this period was 21.31 tonnes.
- The first wall replacement period was calculated as 3.92 years.
- The change in fission power generation increased from 5116 MW to 8131 MW.

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RESUME

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