International Journal of Current Science Research and Review
ISSN: 2581-8341
Volume 09 Issue 08 August 2026
DOI: 10.47191/ijcsrr/V9-i8-04, Impact Factor: 8.048
IJCSRR @ 2026
www.ijcsrr.org

Neutron Economy and the Thorium Fuel Cycle in Thorium Molten Salt Reactors: A Systematic Review

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ABSTRACT: This study investigates neutron economy and the thorium fuel cycle within Thorium Molten Salt Reactors to evaluate their potential for sustainable nuclear energy production. Using a Systematic Literature Review (SLR) methodology, this study synthesized recent research published between 2020 and 2026, while also using selected foundational sources from 1960 onward for core nuclear-theory parameters. Data were analyzed using a qualitative and comparative approach, focusing on thorium-232 conversion pathways, uranium-233 production mechanisms, and neutronic performance parameters. The findings indicate that U-233 is a superior fissile fuel for the thorium cycle, exhibiting a high neutron reproduction factor of 2.28 for thermal neutrons and 2.50 for fast neutrons, which supports efficient fuel breeding. The thorium cycle offers significant environmental and security benefits, including a substantial reduction in long-lived transuranic actinides and enhanced proliferation resistance due to the presence of uranium-232, which emits high-energy gamma radiation. Furthermore, TMSR technology provides inherent safety through a negative temperature coefficient and the ability for online fuel processing to remove gaseous fission products. Unlike previous review studies, this research integrates neutron economy, U-233 purity, breeding ratio, salt chemistry, and reactor safety into a unified conceptual framework, providing a comprehensive analytical perspective for the evaluation of thorium molten salt reactors. The study concludes that while the thorium-uranium cycle is a highly effective alternative for improving neutron efficiency and long-term sustainability, successful commercial deployment depends on achieving U-233 purity levels of approximately 85%. Additionally, further research is required to address remaining challenges in structural materials, salt redox chemistry control, and fuel reprocessing.

KEYWORDS: Thorium Fuel Cycle, Neutron Economy, Molten Salt Reactors, Uranium-233, Breeding Ratio

1. INTRODUCTION

The global transition toward low-carbon energy systems has renewed interest in advanced nuclear technologies as reliable and sustainable sources of electricity (OECD, 2015). Among these technologies, the thorium (Th-232) fuel cycle has emerged as a promising alternative to the conventional uranium fuel cycle because thorium is approximately three times more abundant than uranium and has the potential to reduce long-lived radioactive waste (OECD, 2015; (Dwijayanto & Harto, 2024). Molten Salt Reactors (MSRs), one of the Generation IV reactor concepts, are particularly well suited for thorium utilization due to their low operating pressure, high operating temperature, and capability for online fuel processing (Ashraf et al., 2020; Zhang et al., 2020).

Thorium is a fertile rather than fissile material and must first be converted into uranium-233 (U-233) through neutron absorption (OECD, 2015). Since U-233 does not occur naturally, it must be produced in a reactor, and its production is inevitably accompanied by isotopic impurities, particularly U-232, which influence reactor neutronics, fuel performance, and safety characteristics (Dwijayanto & Harto, 2024). Recent studies have shown that moderate U-233 impurity levels can improve reactor safety by enhancing the temperature coefficient of reactivity while maintaining a high conversion ratio of approximately 85% (Dwijayanto & Harto, 2024).

The feasibility of thorium molten salt reactors was first demonstrated during the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory in the 1960s. Although development slowed in the 1970s, growing concerns over energy security and environmental sustainability have revived worldwide interest in thorium-based reactors (Heuer et al., 2014; Dwijayanto & Harto, 2024). Current research includes the Solid-Fuel Thorium Molten Salt Reactor (SD-TMSR), the Molten Salt Fast Reactor (MSFR), accelerator-driven systems, and other advanced concepts designed to improve neutron economy, fuel utilization, and operational safety (Ashraf et al., 2020; Yang et al., 2020). Reactor-grade plutonium and transuranic fuels have also been investigated as startup fuels to enable the transition to the thorium fuel cycle (Ashraf et al., 2020). In addition, the presence of U-232 and its high-energy gamma-emitting decay products provides an inherent advantage in nuclear non-proliferation despite increasing fuel-handling requirements (OECD, 2015).

Despite considerable progress, previous studies have generally addressed reactor design, fuel chemistry, neutron economy, breeding characteristics, or safety as separate topics. A comprehensive review integrating neutron production, neutron economy, breeding ratio, U-233 purity, salt chemistry, reactor safety, and fuel sustainability within a unified analytical framework remains limited. Therefore, this review aims to synthesize and critically evaluate these interconnected aspects to provide a comprehensive understanding of thorium-based molten salt reactor technology and its future potential.

2. MATERIALS AND METHODS

2.1 Study Design

This study was conducted using a Systematic Literature Review (SLR) approach to investigate neutron economy, thorium fuel cycle in Thorium Molten Salt Reactors (TMSRs). The review was conducted following the PRISMA 2020 guidelines to ensure a transparent, systematic, and reproducible study selection process .The systematic review methodology was selected because the objective of this study was to collect, evaluate, and synthesize existing scientific knowledge related to thorium-based molten salt reactor technology, with particular emphasis on neutron balance, fissile material production, and neutronic performance parameters. The review focused on analyzing previously published research articles and scientific books related to thorium fuel cycles, molten salt reactor concepts, neutron production mechanisms, and nuclear fuel characteristics.

2.2 Literature Search Strategy

A comprehensive literature search was conducted using reliable scientific databases and technical sources, including Google Scholar, ScienceDirect, International Atomic Energy Agency (IAEA) publications, National Nuclear Laboratory (NNL) reports, and other recognized nuclear science resources. The review primarily focused on studies published between 2020 and 2026 to capture recent developments in thorium molten salt reactor technology, while selected foundational publications published from 1960 onward were included to provide established reactor physics principles, nuclear constants, and historical background. The literature search was performed using combinations of keywords connected by Boolean operators (AND and OR) to maximize the retrieval of relevant studies. The principal search string included: ("Thorium Fuel Cycle" OR "Thorium-U-233 Fuel Cycle") AND ("Thorium Molten Salt Reactor" OR TMSR OR MSR) AND ("Neutron Economy" OR "Breeding Ratio" OR "Neutron Production" OR "U-233" OR "Nuclear Fission" OR "Salt Chemistry" OR "Reactor Safety") Additional keyword combinations were also employed to identify studies related to thorium breeding, neutron balance, delayed neutrons, fuel salt composition, molten salt chemistry, online fuel processing, and reactor safety characteristics. Duplicate records were removed before screening, and the remaining studies were evaluated according to the predefined inclusion and exclusion criteria. The study selection process was documented using the PRISMA flow diagram to ensure a transparent and systematic review procedure.

2.3 Selection Criteria

The selection of scientific sources was conducted based on predefined inclusion and exclusion criteria to ensure the relevance and quality of the reviewed literature. Studies were included if they discussed thorium fuel cycle mechanisms and U-233 production, reported neutron economy, neutron production, or breeding characteristics, presented nuclear properties of U-233, U-235, and Pu-239, investigated molten salt reactor design, operation, safety, or fuel salt composition, and were published in peer-reviewed journals, scientific books, conference proceedings, or recognized institutional reports. Studies were excluded if they were unrelated to thorium-based nuclear systems, lacked sufficient scientific or technical information, were duplicate publications, or originated from non-scientific sources without reliable references.

2.4 Data Extraction

Relevant information was extracted from the selected studies using a structured data collection approach to ensure consistency and reliability in the review process. The extracted parameters included thorium fuel cycle characteristics and Th-232 conversion pathways, U-233 production mechanisms through neutron absorption and beta decay, and neutron economy parameters such as neutron balance, neutron yield, and breeding ratio. In addition, nuclear properties of fissile isotopes were collected, including thermal

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neutron fission cross-section, neutron reproduction factor (η), delayed neutron fraction (β), average number of neutrons released per fission, critical mass, and fissile material purity. Molten salt reactor parameters were also extracted, including fuel salt composition, redox chemistry characteristics, fuel processing and fission product removal mechanisms, as well as safety characteristics and temperature coefficients. The collected data were systematically organized into comparative tables to evaluate the differences between thorium-based fuel cycles and conventional uranium/plutonium fuel cycles. The collected information was analyzed using a qualitative and comparative analytical approach.

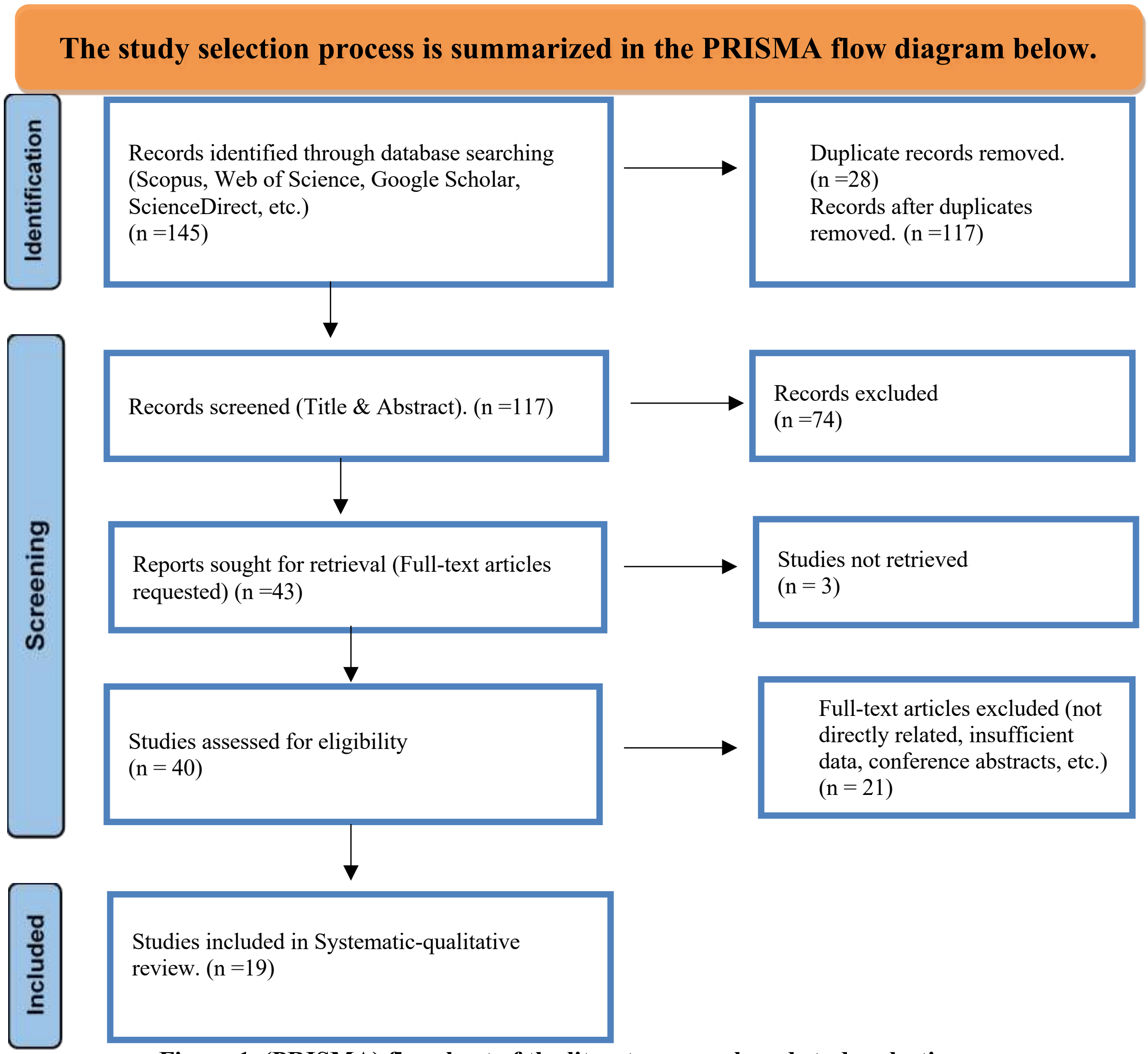

Figure 1. (PRISMA) flowchart of the literature search and study selection process.

3. RESULTS AND DISCUSSION

3.1 Thorium Fuel Cycle

Uranium-233 is produced through the neutron irradiation of thorium-232 nuclei, in which thorium-232 absorbs neutrons and undergoes a series of nuclear transformations leading to the formation of fissile uranium-233. Thorium is several times more abundant in the Earth's crust than uranium, making it a promising alternative nuclear fuel resource. Compared with other nuclear fuel cycles, the thorium fuel cycle generates significantly lower quantities of long-lived transuranic actinides, such as plutonium, americium, and curium. This reduction in actinide production contributes to simpler and more efficient nuclear waste management strategies.

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$$^{232}_{90}Th(n,\gamma)^{233}_{92}U \rightarrow {}^{233}_{91}Pa + \beta^-(T_{1/2} = 22\text{ min}) \rightarrow {}^{233}_{92}U + \beta^-(T_{1/2} = 27.4d)$$

In addition, the thorium fuel cycle is considered to be more resistant to nuclear weapons proliferation compared with the uranium–plutonium fuel cycle. This is because uranium-233 can be rendered unsuitable for weapons applications when it is mixed with uranium-238. Furthermore, uranium-232 is also produced during the thorium fuel cycle, which emits intense gamma radiation through its decay products. The presence of these high-energy gamma emissions increases the difficulty of handling, processing, and utilizing uranium-233 for nuclear weapons production, thereby enhancing the proliferation resistance of the thorium fuel cycle (Jeong & Park, 2006).

Thorium is approximately four times more abundant in the Earth's crust than uranium, and the conversion of thorium-232 into uranium-233 can occur through interactions with both thermal and fast neutrons. Unlike uranium fuel, thorium does not require the complex uranium enrichment process, which is a costly and security-sensitive operation; therefore, thorium can be utilized in reactors without prior enrichment. When thorium-232 absorbs a neutron, it undergoes a series of nuclear transformations involving two successive beta decays, resulting in the production of uranium-233, a fissile isotope with favorable neutron absorption and fission characteristics. This conversion process improves neutron economy and contributes to more efficient utilization of nuclear fuel while simplifying nuclear waste management (Kaygisiz, 2023).

Thorium Molten Salt Non-Moderator Reactors (TMSR-NM) operate based on a fuel cycle in which the fuel material is continuously processed during reactor operation. This online processing enables the removal of noble metals, gaseous fission products, and lanthanides from the fuel salt, ensuring stable and efficient reactor performance. The chemical composition of the molten salt and its purification processes are considered fundamental aspects of the reactor's operation, as they directly influence fuel utilization, neutron economy, and long-term reactor sustainability (Doligez et al., 2008).

Supporters of thorium reactors claim that the radioactive waste from the thorium fuel cycle remains hazardous for only about 300 to 500 years, whereas the waste generated by current nuclear reactors requires management and isolation for tens of thousands of years (DiLisi et al., 2018).

The thorium fuel cycle can operate as a "breeder" cycle, in which more fissile fuel is produced than is consumed within the reactor. The physical process involves the absorption of a neutron by Th-232, converting it into Th-233, which subsequently undergoes beta decay to form protactinium-233 and finally fissile U-233. During fission, U-233 releases energy along with additional neutrons, allowing the fuel cycle to be sustained. Furthermore, the amount of U-232 impurity present in uranium-233 depends on the reactor type and the fuel burnup level. U-232 gradually produces thallium-208 (Tl-208) through its decay chain, which emits highly energetic gamma radiation. These gamma emissions create significant radiological hazards for individuals attempting to use this material for nuclear weapons production and also facilitate the detection of such materials, thereby enhancing proliferation resistance (Kang & Von Hippel, 2001).

The results indicate that the thorium fuel cycle has strong potential for sustainable nuclear energy because thorium is approximately four times more abundant than uranium and is converted into fissile U-233 through neutron capture, enabling a self-sustaining breeder cycle (Kaygisiz, 2023). Compared with conventional uranium fuel cycles, it produces fewer long-lived transuranic actinides, reducing the hazardous lifetime of radioactive waste to approximately 300–500 years, compared with tens of thousands of years for conventional spent fuel (DiLisi et al., 2018). Moreover, the production of U-232 and its decay product Tl-208, which emits high-energy gamma radiation (~2.6 MeV), significantly enhances proliferation resistance, although it also increases fuel handling complexity (Kang & Von Hippel, 2001;Jeong & Park, 2006). Furthermore, continuous online fuel processing in molten salt reactors improves neutron economy by removing neutron-absorbing fission products, thereby enhancing fuel utilization and long-term reactor performance (Doligez et al., 2008). Therefore, the effectiveness of the thorium fuel cycle depends on integrating its favorable nuclear characteristics with advanced fuel processing and reactor technologies.

3.2 Neutron Economy and Breeding Ratio

Molten salt reactors can function as thermal breeder reactors in the thorium fuel cycle only when they achieve a favorable neutron economy. In this context, the term conversion ratio is used to evaluate the breeding process, representing the capability of a reactor to produce new fissile materials relative to the amount of fissile material consumed. Research indicates that impurities (contaminants) present in the fuel, such as U-234, can influence neutron economy by absorbing neutrons. However, under certain conditions, this neutron absorption may partially compensate for the lower neutron absorption characteristics of thorium and slightly

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improve the conversion ratio. Nevertheless, if the purity of U-233 is significantly reduced, the breeding performance decreases substantially. For effective breeding operation, a U-233 purity level of approximately 85% is recommended (Dwijayanto & Harto, 2024).

Neutron balance in a reactor shows the ratio between the neutrons produced and the neutrons that are either absorbed or lost from the system through leakage. Under steady-state conditions, the produced neutrons must be equal to the consumed neutrons. If this balance is disturbed, the neutron population in the reactor changes. In a stable reactor, there must be a balance between production, absorption, and leakage. In a simple form, this relationship can be expressed as: Production = Absorption + Leakage(Puthiyavinayagam, 2009).

Neutrons are produced as a result of nuclear fission. When a heavy nucleus (such as uranium-233) absorbs a neutron, nuclear fission occurs. As a result, in addition to energy, new neutrons are also produced. On average, approximately 2 to 3 neutrons are generated in each nuclear fission event. The neutron production rate in a reactor, represented by the fission rate $R_f = \nu\Phi\Sigma_f$, can be expressed by the following relationship (Tsoulfanidis & Landsberger, 2015).The neutron diffusion equation $D\nabla^2\Phi - \Sigma_a\Phi + S = 0$ shows the neutron balance or equilibrium in the medium. Based on diffusion theory, the amount of leakage per unit volume is given in the following equation:

$$Leakage = -D\nabla^2\Phi$$

where D is the diffusion coefficient and $\nabla^2\Phi$ is the flux Laplacian. The neutron flux Laplacian $\nabla^2\Phi$ determines the measure of the spatial curvature of the flux. It determines whether neutrons at a point leave that region or enter that region due to diffusion/spreading. Therefore, this quantity forms an essential part of the reactor diffusion equation and plays an important role in the analysis of the spatial distribution of neutrons. (Reuss, 2020)

Uranium-233 possesses important nuclear characteristics that make it one of the most suitable fissile materials for the thorium fuel cycle. It is produced through neutron absorption by thorium-232 and has a half-life of 1.6×10^5 years. Thorium-232 is more abundant in the Earth's crust (9.6 ppm) than uranium-238 (2.7 ppm), and its thermal neutron absorption cross-section (7.4 barn) is higher than that of uranium-238 (2.7 barn), which contributes to the effectiveness of the thorium fuel cycle. U-233 has a lower critical mass compared with U-235. With a beryllium reflector, its critical mass is 8.4 kg, compared with 21 kg for U-235 and 7.5 kg for Pu-239. This indicates that U-233 can sustain a chain reaction with a relatively smaller amount of fissile material. It also has favorable neutronic properties, producing an average of 2.28 neutrons per absorbed thermal neutron and 2.50 neutrons per fast neutron, which provides advantages in neutron economy and fuel breeding capability. The spontaneous fission rate of U-233 is very low (0.5 $s^{-1}kg^{-1}$), which improves reactor startup and operational safety. Its decay heat is also lower (0.3 W/kg) than Pu-239 (2.4 W/kg), reducing the challenges associated with residual heat management after reactor shutdown. The delayed neutron fraction of U-233 is 0.00266, lower than U-235 but slightly higher than Pu-239. Although reduced compared with U-235, it remains sufficient for reactor control. In molten salt reactors, protactinium-233 can be chemically separated from the fuel salt to produce relatively pure U-233 and reduce U-232 contamination. However, small amounts of U-232 generate strong gamma radiation through the decay of Tl-208 (about 2.6 MeV), making fuel handling more difficult while also providing a natural barrier against nuclear proliferation. Therefore, U-233 is sometimes denatured with U-238 to increase its critical mass and prevent weapon use (Kang & Von Hippel, 2001).

Table 1: Comparison of Nuclear Properties of U-233, U-235, and Pu-239(Kang & Von Hippel, 2001).

Parameter	Uranium-233	Uranium-235	Plutonium-239
Half-life (years)	1.6×10^5	7.0×10^8	2.4×10^4
Fertile isotope	Th-232	Obtained from natural uranium	U-238
Abundance in the Earth's crust	9.6 ppm	–	2.7 ppm
Thermal neutron absorption cross-section	7.4 barns	–	2.7 barns
Critical mass (kg)	8.4	21	7.5
Fissile material purity	98.3% U-233	93.5% U-235	4.9% Pu-240
Type of reflector	Beryllium (Be)	Beryllium (Be)	Beryllium (Be)

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Reflector thickness (cm)	3.7 cm	5.1 cm	4.2 cm
Neutrons released per absorbed neutron for 1 MeV neutrons	2.50	2.30	2.90
Neutrons released per absorbed neutron for thermal neutrons (0.025 eV)	2.28	2.28	2.11
Delayed neutron fraction (β)	0.00266	0.0065	0.00212

The following are the reasons explained in Chapter 8 of John R. Lamarsh's *Introduction to Nuclear Engineering* theory book. The low-energy cross-section values are usually tabulated for an energy of 0.0253 eV. Neutrons that are in thermal equilibrium with their surroundings at room temperature have approximately this energy. The cross-section corresponding to this thermal energy range is called the thermal cross-section. The following table presents some parameters of the isotopes uranium-233, uranium-235, natural uranium, plutonium-239, and plutonium-241 (Lamarsh, 1966).

Table 2: Parameters of Uranium-233, Uranium-235, Natural Uranium, Plutonium-239, and Plutonium-241 at an Energy of 0.025 eV. (Lamarsh, 1966).

Nucleus / Isotope	σ_a (barn)	σ_f (barn)	α	η	ν
U-233	573	525	0.093	2.29	2.50
U-235	678	577	0.175	2.08	2.44
Natural Uranium	7.59	4.16	0.910	1.31	2.50
Pu-239	1015	741	0.370	2.12	2.90
Pu-241	1375	950	0.357	2.21	3.00

The results indicate that the success of the thorium fuel cycle is primarily governed by neutron economy and breeding performance. Uranium-233 exhibits favorable neutronic characteristics, producing 2.28 neutrons per absorbed thermal neutron and 2.50 neutrons per fast neutron, which provide sufficient excess neutrons to sustain the breeding cycle while maintaining reactor criticality (Kang & Von Hippel, 2001). Furthermore, the relatively low critical mass of 8.4 kg for U-233, compared with 21 kg for U-235, demonstrates its superior capability for sustaining chain reactions with a smaller quantity of fissile material (Kang & Von Hippel, 2001). However, reactor performance strongly depends on fuel purity, as reducing the U-233 purity below the recommended level significantly decreases the conversion ratio; therefore, an isotopic purity of approximately 85% is considered optimal for efficient breeding operation (Dwijayanto & Harto, 2024). Although the delayed neutron fraction of U-233 (β = 0.00266) is lower than that of U-235 (β = 0.0065), it remains adequate for reactor control, particularly in molten salt reactors where continuous fuel processing improves neutron utilization and reduces neutron losses (Kang & Von Hippel, 2001; Tsoulfanidis & Landsberger, 2010). Consequently, maintaining neutron balance by minimizing absorption and leakage is essential for achieving stable reactor operation and long-term fuel sustainability (Puthiyavinayagam, 2009; (Reuss, 2020).

3.3 Prompt and Delayed Neutrons

One of the important products of nuclear fission is the neutrons released during this process. The majority of these neutrons are produced almost instantaneously, within approximately $10^{-17} sec$ after the fission event, and are called prompt neutrons. A very small fraction of neutrons is released much later after the fission process, and these are known as delayed neutrons. For uranium-233, the relative fraction of delayed neutrons is very small, with β=0.0026, indicating that delayed neutrons constitute only about 0.26% of the total neutron yield from fission. Additionally, approximately 0.0066 delayed neutrons are produced on average per fission event. According to the source, the decay constant of the first group is 0.0126 s^{-1}, while that of the sixth group is 2.50 s^{-1}. Based on these data, the delayed neutron fraction of uranium-233 is lower than that of uranium-235, for which this fraction is 0.0065. Nevertheless, delayed neutrons play a crucial role in reactor control, stability, and safe operation (Lamarsh, 1966).

The results indicate that reactor safety and controllability strongly depend on delayed neutrons, despite their small contribution to the total neutron population. For U-233, the delayed neutron fraction is only β = 0.0026 (0.26%), with an average of 0.0066 delayed neutrons per fission, which is considerably lower than that of U-235 (β = 0.0065) (Lamarsh, 1966). Although prompt neutrons are emitted within approximately 10^{-17} s after fission, the delayed neutrons released over longer time intervals provide sufficient time for reactor control systems to respond, thereby maintaining stable and safe reactor operation. Therefore, the relatively low delayed neutron fraction of U-233 requires more precise reactivity control in thorium molten salt reactors while still providing adequate operational stability for safe reactor performance (Lamarsh, 1966).

3.4 Nuclear Fission

Thorium-232 is a fertile material that is converted into fissile uranium-233 through neutron absorption. Since thorium does not naturally contain any fissile isotopes, the fuel cycle must be initiated with uranium-235 or plutonium-239. The thorium fuel cycle can produce useful energy compared with the uranium fuel cycle; however, its recycling processes and waste management are still not completely understood (National Nuclear Laboratory, 2010).

Nuclear fission has practical significance mainly for very heavy nuclei. The values of the critical energy for several heavy nuclei are given in the following table, and it should be noted that the specific nucleus in which the nuclear fission process occurs is shown in this table. Nuclear fission can also occur without increasing the critical energy through the process of overcoming the Coulomb barrier by quantum mechanical tunneling, similar to the way that α-particles pass through the barrier during α-decay. However, this type of spontaneous fission occurs with a very low probability. For example, in U^{238}, spontaneous fission occurs only once per 100 seconds per gram, which is equivalent to a half-life of approximately 5.5×10^{15} years. It is expected that the spontaneous fission rates of U^{233}, U^{235}, and Pu^{239} would be higher than that of U^{238} because, according to the table, these nuclei have lower critical energy values. However, in practice, the spontaneous fission rate of U^{238} is several orders of magnitude higher than those of U^{233}, U^{235}, and Pu^{239}. This is an unexpected result and has not yet been fully explained. Although spontaneous fission events occur very slowly, they still play an important role in the analysis of nuclear reactors (Lamarsh, 1966).

Table 3: Critical Energy for Nuclear Fission and Binding Energy of the Last Neutron. (Lamarsh, 1966)

Fissionable Nucleus (Z, A)	Critical Energy (MeV)	Binding Energy of the Last Neutron (MeV)
Th^{232}	5.9	*
Th^{233}	6.5	5.1
U^{233}	5.5	–
U^{234}	4.6	6.6
U^{235}	5.7	–
U^{236}	5.3	6.4
U^{238}	5.8	–
U^{239}	5.5	4.9
Pu^{239}	5.5	–
Pu^{240}	4.0	6.4

*This symbol indicates that the binding energy of the last neutron is not provided for Th^{232} and similar nuclei because these nuclei are not formed through neutron absorption; rather, after neutron absorption, they can transform into another nucleus.

Nuclear fission can also occur through gamma radiation; this process is called photofission. This process takes place when the energy of gamma radiation is higher than the critical energy of the nucleus. In most nuclear reactors, gamma rays with such high energy are relatively rare; therefore, the effect of photofission is generally neglected in reactor design and analysis. However, photofission is a useful and important experimental method for determining the critical energy of nuclei, and most of the (E_{crit}) values presented in the above table have been obtained using this method (Lamarsh, 1966).

The results indicate that the thorium fuel cycle relies on the efficient conversion of fertile Th-232 into fissile U-233, requiring an initial fissile inventory such as U-235 or Pu-239 to sustain the chain reaction (National Nuclear Laboratory, 2010). The fission characteristics of U-233 are favorable because its critical fission energy is 5.5 MeV, which is comparable to Pu-239 (5.5 MeV) and slightly lower than U-235 (5.7 MeV), facilitating neutron-induced fission (Lamarsh, 1966). Although U-238 undergoes spontaneous fission only about once per 100 s per gram, corresponding to a half-life of approximately 5.5×10^{15} years, spontaneous fission has a negligible contribution to reactor power and is mainly important for reactor physics analyses (Lamarsh, 1966). Furthermore, photofission occurs only when gamma-ray energy exceeds the critical fission energy; however, because such high-energy gamma rays are rarely present in operating reactors, its contribution to reactor performance is generally insignificant (Lamarsh, 1966).

3.5 Neutron Production

In the nuclear fission process, the number of prompt neutrons produced varies among different nuclear interactions. Therefore, in some nuclear reactions, no neutrons are produced, while in other cases, up to five neutrons may be released. However, in reactor calculations, only the average number of neutrons emitted as a result of each nuclear fission event is considered, and this quantity is represented by the symbol (ν). According to the conventional definition, (ν) refers to the total average number of both prompt and delayed neutrons. The value of (ν) can be obtained from the equation $\nu(E)=\nu 0+aE$ and its value depends on the fissile isotope and the energy of the incident neutron (Lamarsh, 1966).

The number of neutrons produced for uranium-233 is 2.28 neutrons per absorbed thermal neutron, while for fast neutrons with an energy of 1 MeV, it produces 2.50 neutrons. This indicates that as the energy of the incident neutrons increases, the number of produced neutrons also increases (Kang & Von Hippel, 2001).

3.6 Neutron Absorption and Nuclear Fission Cross Sections

The probability of nuclear fission by thermal neutrons is not the same for different fuels; rather, it varies according to the nuclear characteristics of each isotope. The values in the table show that plutonium-239 has the largest fission cross section (742 barns), followed by uranium-235 (582 barns), and uranium-233 (523 barns). In these three isotopes, the fission cross section is several times higher than the neutron absorption cross section, indicating their effective fission capability with thermal neutrons. In contrast, uranium-238 has an almost zero thermal neutron fission cross section ($\sigma_f = 0$); meaning that it does not undergo fission at this energy but only absorbs neutrons. Therefore, it is evident that uranium-233, uranium-235, and plutonium-239 are suitable fissile fuels for thermal reactors, while uranium-238 mainly acts as a fertile material and contributes to the production of new fissile isotopes through neutron absorption (Liverhant, 1960).

The results indicate that neutron production and fission cross sections are the primary factors determining neutron economy and reactor performance. Uranium-233 produces an average of 2.28 neutrons per absorbed thermal neutron and 2.50 neutrons for 1 MeV fast neutrons, demonstrating that neutron production increases with incident neutron energy, thereby improving the breeding capability of the thorium fuel cycle (Kang & Von Hippel, 2001). Furthermore, the thermal neutron fission cross sections of Pu-239 (742 barns), U-235 (582 barns), and U-233 (523 barns) confirm their high fission probability, whereas U-238 has an almost zero thermal fission cross section and primarily functions as a fertile material through neutron absorption (Liverhant, 1960). Therefore, although U-233 has a slightly lower thermal fission cross section than U-235 and Pu-239, its favorable neutron production characteristics make it an effective fissile fuel for sustaining the thorium breeder cycle and improving long-term reactor efficiency (Kang & Von Hippel, 2001).

3.7 Effects of Salt Concentration

The salts used in molten salt reactors are generally composed of fluoride salts because fluorine has only one stable isotope, fluorine-19, and a low neutron absorption cross section, which plays an important role in improving the neutron economy and stability of the reactor. The composition of these salts is determined according to the reactor design and fuel cycle, and it contains dissolved fluoride compounds of fissile and fertile materials, such as UF_4, ThF_4, and PuF_3. For example, the ARE used a salt composition of 53% NaF, 41% ZrF_4, and 6% UF_4; the MSRE used 65% LiF, 29% BeF_2, 5% ZrF_4, and 1% UF_4; and the MSFR used a salt composition of 77.5% LiF and 22.5% ThF_4. The latter composition is considered one of the important and suitable compositions for thorium-based molten salt reactors. In molten salt reactors, the uranium oxidation states U(III) and U(IV) form the basis of the salt redox chemistry. The U(IV)/U(III) ratio is an important indicator of the redox potential of the salt and plays a fundamental role in controlling the chemical stability of the reactor, corrosion of structural materials, and uranium solubility in the salt. An increase in U(IV) makes the salt more oxidizing and increases the possibility of corrosion of metallic materials, while an increase in U(III) makes the salt more reducing and increases the possibility of precipitation of some metallic elements. Therefore, maintaining the U(IV)/U(III) ratio within an appropriate range through redox control is essential for safe and long-term reactor operation. Furthermore, the salt composition affects the reactor's neutronic characteristics, breeding ratio, inherent safety, heat transfer capability, and the ability for online purification of fission products. All of these factors form the foundation for the high efficiency and safety of molten salt reactors (Serp et al., 2014).

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ISSN: 2581-8341
Volume 09 Issue 08 August 2026
DOI: 10.47191/ijcsrr/V9-i8-04, Impact Factor: 8.048
IJCSRR @ 2026
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Thorium molten salt reactors have a negative temperature coefficient; when the fuel salt is heated, it expands and the nuclear reaction naturally decreases, allowing the reactor to regulate itself. These reactors are equipped with fuel drain tanks that operate by gravity. If the reactor temperature exceeds the safe limit, the fuel automatically drains into safe tanks where the reaction stops. The use of liquid fuel allows the reactor to operate in a highly flexible state, and therefore, unlike water-cooled reactors, there is no risk of sudden explosive accidents. Additionally, this system passively transfers the generated heat to the environment, preventing reactor overheating and core damage (Holcomb, 2020).

The results indicate that salt composition and redox chemistry are critical factors influencing the performance and safety of thorium molten salt reactors. Fluoride-based salts, such as 77.5% LiF–22.5% ThF_4, provide favorable neutron economy because ^{19}F has a low neutron absorption cross section, while maintaining an appropriate U(IV)/U(III) ratio improves chemical stability, uranium solubility, and minimizes corrosion of structural materials (Serp et al., 2014). In addition, thorium molten salt reactors exhibit an inherent negative temperature coefficient, allowing reactor power to decrease automatically as fuel temperature increases, thereby enhancing passive safety. Furthermore, gravity-driven fuel drain tanks provide an effective passive shutdown mechanism by transferring overheated fuel into subcritical storage tanks, significantly reducing the risk of core damage and severe accidents (Holcomb, 2020). Therefore, optimized salt composition combined with effective redox control and passive safety systems plays a fundamental role in improving the efficiency, reliability, and long-term sustainability of thorium molten salt reactors.

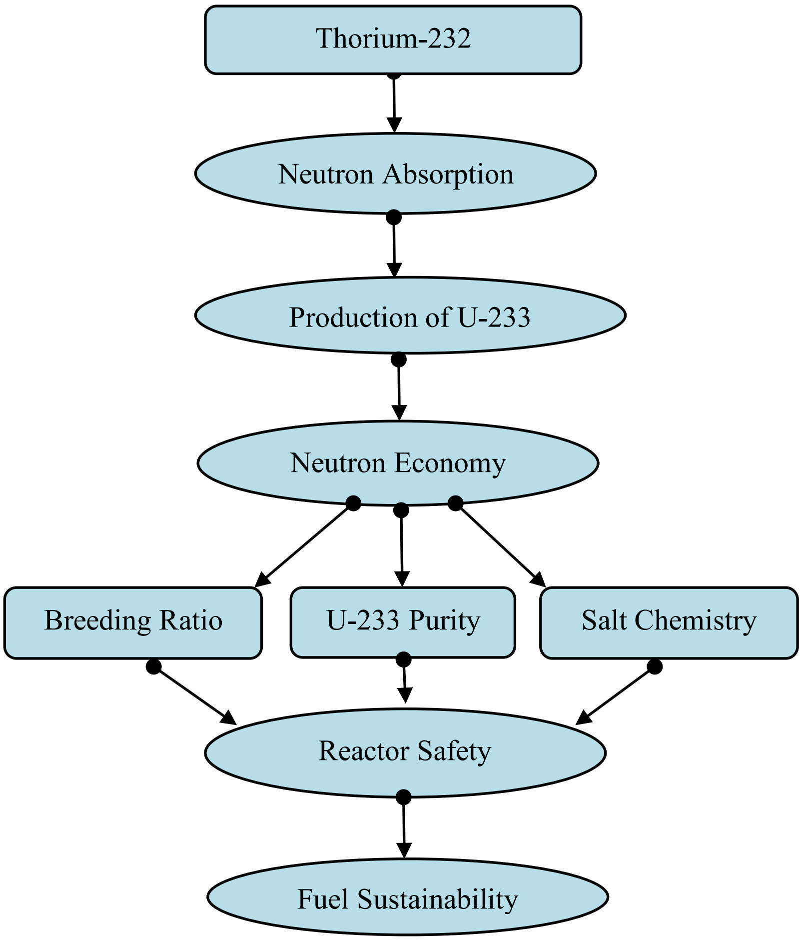

Figure 2. Based on the synthesis of the reviewed literature, this study proposes an integrated conceptual framework demonstrating that neutron economy acts as the central parameter linking thorium conversion, U-233 production, breeding performance, salt chemistry, reactor safety, and long-term fuel sustainability. Unlike previous studies, which generally evaluated these parameters separately, the proposed framework integrates them into a unified analytical model that may assist future reactor design and comparative evaluation of thorium molten salt reactor concepts.

International Journal of Current Science Research and Review
ISSN: 2581-8341
Volume 09 Issue 08 August 2026
DOI: 10.47191/ijcsrr/V9-i8-04, Impact Factor: 8.048
IJCSRR @ 2026
www.ijcsrr.org

5. CONCLUSION

The analysis conducted in this study indicates that the thorium fuel cycle can be considered an effective alternative for nuclear energy production from the perspective of neutron economy. The conversion of Th-232 into U-233 through neutron absorption provides the basis for the production of new fissile fuel and enhances the breeding capability when an appropriate neutron balance is maintained. Uranium-233 exhibits favorable neutron production characteristics, releasing 2.28 neutrons for thermal neutrons and 2.50 neutrons for fast neutrons, thereby providing suitable conditions for sustaining the thorium fuel cycle. Although the relatively low delayed neutron fraction of U-233 presents a challenge for reactor control, neutron economy, conversion ratio, fuel purity, and the control of neutron losses remain the key factors determining the success of the thorium fuel cycle. Overall, the thorium–U-233 fuel cycle offers significant potential for future molten salt reactors because of its reduced production of long-lived actinides, improved neutron efficiency, and long-term fuel sustainability.

Based on the synthesis of the reviewed studies, this review proposes an integrated conceptual framework linking thorium conversion, U-233 production, neutron economy, breeding ratio, reactor safety, and fuel sustainability. This framework demonstrates that neutron economy acts as the central connecting parameter governing overall reactor performance. The proposed framework represents an original analytical contribution of this review because previous studies generally discussed these parameters independently rather than within a unified evaluation model.

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Cite this Article: Rahimi, A., Pyawarai, A.K., Saqib, A. (2026). Neutron Economy and the Thorium Fuel Cycle in Thorium Molten Salt Reactors: A Systematic Review. International Journal of Current Science Research and Review, 9(8), pp. 4253-4263. DOI: https://doi.org/10.47191/ijcsrr/V9-i8-04