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Moderator Optimisation in Core Design of Small Modular – Molten Salt Reactor for Military Submarines

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Abstract

The Molten Salt Reactor (MSR) has the potential to be a 4th generation nuclear reactor design due to its high thermal-electric conversion efficiency, integrated safety features, and online reprocessing capacity. The Molten Salt Reactor (MSR) does not require shutdown during operation because the fuel is reprocessed online, making it suitable for submarine propulsion. In this study, beryllium oxide and graphite are used as moderators to increase neutrons in the reactor. Beryllium oxide and graphite are two types of moderators typically employed in nuclear reactors, one of which is in the MSR. Good moderation properties produce many slow neutrons for nuclear fission reactions. The reactor is powered by LiF-U233F3-ThF4 salt in a virtual one-and-a-half liquid configuration. All reactor core design calculations were performed using the MCNP 6.2 code with ENDF/B-VII.0 neutron cross-section. In the context of neutron analysis, the calculated parameters are the effective multiplication factor (K_{eff}) *and the temperature coefficient of reactivity (TCR). The value of* $K_{eff} = \pm 1.005$ *is obtained by using the ratio of Th : U,* $98.180\% : 1.82\%$*. TCR values were obtained as* $-2.50554\ pcm/K$ *for graphite moderator and* $-2.50554\ pcm/K$ *for the beryllium oxide moderator. These initial values indicate that the MSR can reach the critical level with the existing safety characteristics of its features. Using the MSR design, the submarine can achieve a much better range than diesel–powered submarines.*

Keywords*: Molten Salt Reactor; MCNP; Graphite; Beryllium Oxide; Nuclear Submarine.*

Introduction

Molten Salt Reactor (MSR) is a nuclear reactor that uses salt fuel in liquid form (U.S. Department of Energy, 2002). By using liquid fuel, this reactor has several advantages over other reactors, the reactor does not need to be shut down for refueling and has better fuel utilization potential (Dwijayanto et al., 2021) (Serp et al., 2014). MSRs have several advantages over other nuclear reactor designs: the fission products are fast, it has a very strong negative temperature, they can work as a simple converter with excellent uranium utilization they have good cooling, and they have great advantages in waste destruction (Leblanc & Popoff, 2012).

The Molten Salt Reactor (MSR) is one of the six Generation IV nuclear reactor designs proposed in the Generation IV Forum (GIF). The GIF consists of the following thirteen countries: Argentina, Brazil, Canada, China, Euratom (European Atomic Energy Community), France, Japan, South Korea, Russia, South Africa, Switzerland, the United Kingdom, and the United States, and evaluated about 100 different nuclear reactor types suitable for fourth-generation nuclear energy systems (Wulandari & Permana, 2020). MSRs offer many advantages over conventional *Light Water Reactors* (LWRs), such as passive safety systems, atmospheric operating pressures, high operating temperatures, and no need for fuel fabrication. The *online reprocessing* capability allows MSRs to achieve breeding in the thermal spectrum using the thorium fuel cycle while maintaining high fuel burnout (International Atomic Energy Agency., 2005).

Numerical modeling is still used to develop such reactors, given the current lack of experimental facilities. However, extensive experimental research was conducted at Oak Ridge National Laboratory (ORNL) in the 1950s and 1960s (Carelli & Ingersoll, 2021). providing a basis for their viability from an experimental perspective. A common fuel salt choice, LiF-U-233F3-ThF4, has good flow and heat transfer characteristics, a eutectic melting point at 499°C, and low vapor pressure at working temperature. As a converter reactor, MSRs can be used to convert spent fuel from existing light water reactors (LWRs) (Rykhlevskii et al., 2019).

In all countries so far, the use of nuclear propulsion systems on ships has only been applied to aircraft carriers and submarines. Some countries that have submarines that use nuclear technology are the United States, Russia, China, UK, France and India (Khlopkin & Zotov, 1997) (Hirdaris et al., 2014) (Mitenkov & Polunichev, 1997). Many technical challenges and regulatory considerations must be overcome before MSR technology can be applied to submarines or other commercial naval applications. A suitable MSR core design must also be well designed to accommodate its use in the naval sector therefore being able to operate freely in high-pressure water and provide long-term operating safety.

This research discusses the neutronic aspects of MR with 2 moderators beryllium oxide and graphite using the Monte Carlo code MCNP 6.2. beryllium oxide has good resistance to nuclear radiation and, a neutral neutron *cross-section.* And good resistance to nuclear radiation. The MSR core model is customized to specifications that suit the needs of the submarine. Previously, MNCP software was used to model the physics features of the MSFR reactor core (Fiorina et al., 2013). The effective multiplication factor (K_{eff}) and Temperature Coefficient of Reactivity (TCR) were examined in this study.

Literature Review

A submarine propulsion requires a small power nuclear reactor. The Small Modular Molten Salt Reactor (SM-MSR) can be used for submarine propulsion designed to produce 250 MWt equivalent to 100 MWe. MSRs of the small modular type can produce up to 100 MW of thermal energy at temperatures sufficient to generate steam for active turbines that drive electrical generators (Leblanc & Popoff, 2012) (Wu et al., 2022). The small core size is sufficient to produce this level of power, allowing it to be placed on submarines. The carrier salt is eutectic LiF-U233F3 and the fuel salt is ThF4 and a small amount of 233UF4. To improve the fuel cycle performance, a virtual one-half fluid configuration is adopted. Beryllium oxide (BeO) is used as a moderator to enhance core neutronic in small-sized reactors, it is possible to design reactor terraces that have breeding capabilities in the thermal neutron spectrum, thereby increasing the utilization of natural nuclear fuel resources to 100 times the current level. Here are some properties of beryllium oxide as a moderator:

1. In neutron moderation, beryllium oxide has a relatively low neutron absorption cross-section and effectively moderators neutrons.
2. High thermal conductivity, beryllium oxide exhibits excellent thermal conductivity, allowing it to efficiently transfer heat out of the reactor core. This property helps keep operating conditions stable and prevents overheating.
3. Chemical stability, beryllium oxide processes good chemical stability even under high temperature and high radiation environments. It is resistant to corrosion and degradation thus contributing to the long-term performance and safety of nuclear reactors.
4. High melting point, beryllium oxide has a high melting point of about 2,530 degrees Celsius.
5. Low thermal neutron absorption.

Some properties of graphite as a moderator:

1. Neutron propagation power (Moderation): On of the main functions of graphite moderators is to slow down neutrons produced during nuclear reactions. Graphite is able to slow neutrons to a lower speed, increasing the likelihood of neutrons interacting with nuclear fuel.
2. Neutron interaction trace length: Graphite has a fairly large neutron interaction trace length, which means that neutrons can make many elastic collisions with graphite nuclei before changing direction or losing energy.
3. Elastic moderation of neutrons: Graphite can slow down neutrons through elastic collisions, where neutrons lose some of their energy in each collision with the graphite core.
4. Thermal stability: Graphite has good thermal stability, which means it can operate over a wide temperature range without changing its fundamental properties.
5. Heat dissipation capability: Graphite has high thermal conductivity, so it can efficiently dissipate heat generated during nuclear reactions.

The operational temperature of the core is 900 K. Since liquid fluoride has a very high boiling point and a very low vapor pressure, such temperatures can be achieved under atmospheric pressure. The fuel assembly is not required due to the unique properties of molten salts, and thus radiation damage or fuel cladding failure does not apply to MSRs.

The reactor design used BeO and Graphite moderators as a comparison of the criticality of the reactor. The cylindrical reactor vessel is filled with hexagonal BeO moderators and graphite radial reflector blocks, which from the core structure. The main pum circulates the molten salt through a cylindrical channel drilled through the moderator block as a conduit. The heated molten salt is then transferred to the secondary cooling salt using a heat exchanger.

Through the online reprocessing system, the concentration of the fuel composition can be adjusted. Refueling downtime is not required as there are no fuel assemblies in the core, allowing continuous operation. Reactivity and/or power levels can be managed during operation by adjusting the fuel salt flow.

Table 1. SM-MSR core parameters

SM-MSR code parameters	**Values**
Thermal power	250 MW/t
Active core diameter	220 cm
Active core height	240 cm
BeO density	3.01 g/cm^3
Graphite density	2.2 g/cm^3
Hastelloy thickness	5 cm
Core channel radius	2 cm
Blanket channel radius	5 cm
Operational temperature	900 K
Fuel type	Molten salt
Composition	LiF-U233F3-ThF4
Molar composition (%)	
Fuel salt density	3.3 g/cm^3
The volumetric expansion coefficient of the fuel salt	1.78 x 10^{-4} K^{-1}

Methods

The design of SM-MSR on military submarines uses the MNCP simulation method on the reactor model. The effective multiplication factor (K_{eff}) and TCR are some of the significant reactor physics parameters that will be evaluated using MCNP6.2. (Zuhair et al., 2019) and the cross-section data used are adopted from ENDF/B-VII.0. The MCNP code, developed and maintained by the Los Alamos International Laboratory (LANL), is a widely used code for using the Monte Carlo method to analyse radiation transport including neutrons and gamma rays (Shultis & Faw, 2011). MNCP is commonly used for fission and fusion reactor design, nuclear criticality safety analysis, radiation shielding analysis, waste storage/disposal, detector design, and analysis, and used in medical physics and dosimetry, radiotherapy, and combustion. The code has been validated for various reactor types and is known to be capable of modeling scenarios involving complex geometries in radiation transport (Zuhair et al., 2019) (Alzamly et al., 2020) (Kuntoro et al., 2023). For the MSR case, the FUJI-U3-(0) reactor was studied using the MNCP 6.2 code. The results prove that the MNCP 6.2 calculations are comparable to the original SRAC95 calculations.

In this study, a comparison was made between two SM-MSRs with different moderators to determine the critical optimization level of each moderator. Neutronic simulations were performed using 10,000 neutrons in each cycle for a total of 250 cycles, with the first cycles discarded. The fuel composition in the reactor core was adjusted so that the critical core was at a low excess reactivity of about 500 pcm, to make it easier to control the reactivity (Jaradat, 2015).

After that, TCR calculations were performed on two SM-MSR models with a Beryllium oxide (BeO) moderator and Graphite moderator, respectively. The TCR calculation is divided into moderator temperature coefficient (MTC), Doppler coefficient (DC), and salt density coefficient (SDC). DC is obtained by changing the temperature to 1200 K on the fuel card. MTC is calculated by converting the moderator temperature to 1200 K. SDC is obtained by subtracting the salt density of the fuel using the coefficients listed in Table 1.

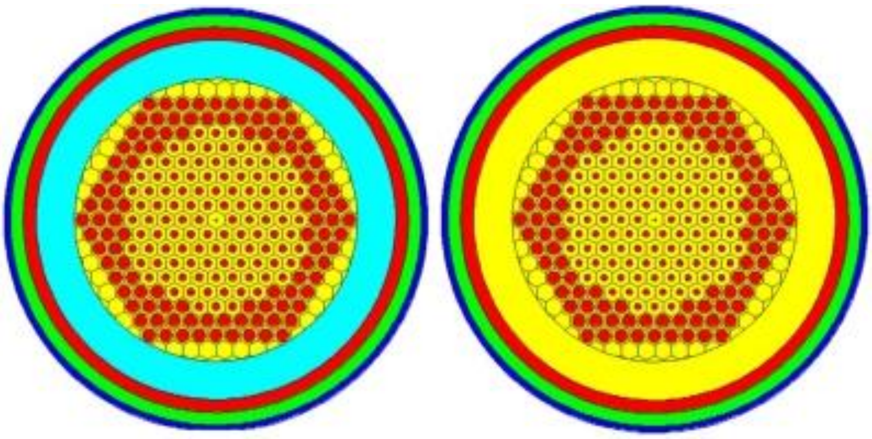

Figure 1. Visualization of *the Small Modular-Molten Salt Reactor* model with beryllium oxide moderator (left) and graphite moderator (right) using MCNP (vertical cross-section)

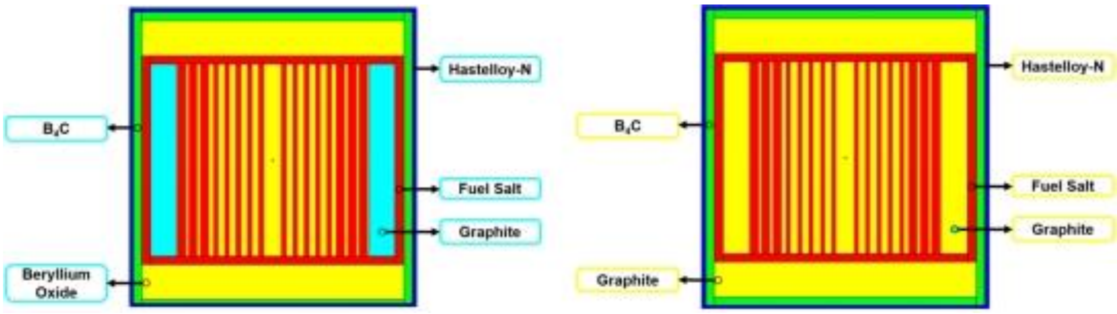

Figure 2. Visualization of *the Small Modular-Molten Salt Reactor* model with beryllium oxide moderator (left) and graphite moderator (right) using MCNP (horizontal cross-section)

Result and Discussion

Multiplication Factor (K_{eff})

The upgrade of the Small Modular-Molten Salt Reactor (SM-MSR) core will extend the combustion cycle. To do this, this outer fuel road ring of the SM-MSR core is enlarged in radius while the overall mass and volume of the core remain constant. As a result, the radius of the fuel roads on the inner ring of the core is reduced. Different scenarios with different radii were examined. Using beryllium oxide and graphite moderators, the optimum fuel diameter variations are shown in Table below:

Table 2. Multiplication factor on beryllium oxide moderator

No	CZ (mm)	K_{eff}
1	2	1,09056
2	2,5	1,11027
3	3	1,11169
4	3,5	1,09067
5	4	1,05591

Table 3. Multiplication factor on graphite moderator

No	CZ (mm)	K_{eff}
1	2	1,0104
2	2,5	1,0053
3	3	1,01906
4	3,5	0,97729
5	4	0,93431

Value K_{eff} is at an effective condition at a radius of 3 mm. This is due to the reactor design, which is affected by the fissile/fertile ratio, the selected reactor core material, and the reactor geometry. Due to the batch -wise nature of this online reprocessing strategy, the coupling factor varies greatly.

From the results shown in the table above, the value of the K_{eff} in the reactor using beryllium oxide moderator is higher by about 1.11169 an the reactor using graphite moderator is lower by about 1.01906. the variation of these two reactors shows the optimisation to criticality in SM-MSR. Some factors that cause the value of K_{eff} on beryllium oxide is higher is better moderation properties that produce many thermal neutrons.

One of the causes of the increase in K_{eff} is the loss of toxins in the reactor core and the addition of fresh fissile material (233U) from the protactinium decay tank. In SM-MSR, the multiplication factor value is set to prevent the reactor from entering a subcritical or supercritical state. Therefore, a value of K_{eff} of ±1.005 was obtained using a Th : U ratio of 98.18%:1.58%.

Table 4. Effective core of Graphite moderator reactor

Effective conditions				
mole	**K_{eff}**	**Stdf**	**rho**	**rho (pcm)**
98,180%	1,00524	0,00046	0,005213	1,737562

Table 5. Effective core of Beryllium Oxide moderator reactor

Effective conditions				
mole	**K_{eff}**	**Stdf**	**rho**	**rho (pcm)**
98,180%	1,00546	0,00046	0,00543	1,810117

Temperature Coefficient of Reactivity (TCR)

TCR is a critical safety parameter for the reactor and must be negative when the reactor is operating. Molten salt TCR and moderator TCR are two different types of TCR for molten salt reactors moderated by beryllium oxide and graphite. The doppler effect plus the density effect form the molten salt TCR. The partial temperature effect and expansion reactivity of beryllium oxide TCR can also result in changes to the geometry of the cell and core. As a result, the whole TCR can be divided into three components.

$$\frac{dk}{dT}total = \frac{dk}{dT}DC + \frac{dk}{dT}SDC + \frac{dk}{dT}MTC$$

Table 6. Total temperature coefficient of reactivity (TCR) of graphite moderator

TCR data						
Parameters	**%mol**	**K_{eff}**	**Stdf**	**rho**	**rho(pcm)**	**TCR**
DC	0,98415	0,9966	0,00044	-0,00341	-1,1372	-2,87476
SDC	0,98415	1,0056	0,00046	0,005569	1,856272	0,11871
MTC	0,98415	1,006	0,0004	0,005964	1,988072	0,25051
Total TCR						-2,50554

Table 7. Total *temperature coefficient of reactivity* (TCR) of beryllium oxide moderator

TCR data						
Parameters	**%mol**	**K_{eff}**	**Stdf**	**rho**	**rho(pcm)**	**TCR**
DC	0,98487	0,9966	0,00044	-0,00341	-1,1372	-2,94731
SDC	0,98415	1,00639	0,00046	0,006349	2,116476	0,30635
MTC	0,98415	1,00745	0,0004	0,007395	2,464969	0,65485
Total TCR						-1,98611

From the results obtained, the TCR values in both reactors are negative, indicating that the safety factor in the reactor is neutronically fulfilled. As the fuel temperatures increases, the salt density decreases, but at some point, the total volume of fuel salt in the core remains constant because it is limited by the moderator. To determine the TCR, the cross-sectional temperatures of the fuel and moderator were changed from 900 K to 1200 K. Three different cases were considered:

1. Increase in fuel salt from 900 K to 1200 K
2. Moderator temperature increased from 900 K to 1200 K
3. Increase in density temperature from 900 K to 1200 K

Conclusion

The SM-MSR study was conducted through a series of calculations using MCNP 6.2 software and ENDF/B-VII.0 neutron cross-section. The calculation results show the TCR value of -2.50554 for graphite moderator and -1.98611 for beryllium oxide moderator. From the MNCP simulation results beryllium oxide works more effectively as a neutron moderator than graphite in the SM-MSR reactor. The results of this study show that SM-MSR can achieve a critical state and neutronically guaranteed safety factor by using beryllium oxide and graphite moderators. It is hoped that research on SM-MSR can be used as literature for the development of nuclear reactors on military submarines replacing diesel engines operating now.

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