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# Core neutronic design of small modular molten salt reactor for submarine propulsion

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**Abstract:** Molten salt reactor (MSR) shows great promise as a Generation IV nuclear reactor concept with high thermal-electric conversion efficiency, inherent safety features, and online reprocessing capability. During operation, the molten salt reactor MSR does not require a long refueling shutdown owing to online fuel reprocessing, making it suitable for watercraft propulsion. This paper presents the neutronic design of the 250 MWt iso-breeder MSR concept for submarine propulsion. The reactor is fueled with LiF-BeF2-ThF4-233UF4 salt with a virtual one-and-a-half fluid configuration. The MSR core is moderated with beryllium oxide and equipped with a high-density graphite reflector to improve neutronic. All calculations on the reactor core design were performed using MCNP6.2 code with ENDF/B-VII.0 neutron cross-section library. As a part of the neutronic analysis, the calculated parameters were the effective multiplication factor ($\mathrm{K_{eff}}$), temperature coefficient of reactivity (TCR), and breeding ratio (BR). The $\mathrm{K_{eff}}$ value of ± 1.005 was obtained using Th: U ratio of 98.42% : 1.58%. The TCR value was obtained at -2.56 pcm/K, while the BR value was 1.057. These initial values indicate that the marine propulsion MSR can achieve iso-breeding with inherent safety characteristics. By using MSR design, it is hoped that the submarines will be able to reach significantly longer operational range than their Diesel-powered counterpart.

*Keywords:* Molten salt reactor, MCNP, neutronics, nuclear submarine

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## 1. Introduction

Fission reactor that uses a mixture of molten salts as the fuel is referred to as molten salt reactor (MSR) (Carelli & Ingersoll, 2015). By using liquid fuel, this reactor has several advantages compared to solid fuel reactors, namely being able to continuously adjust fissile/fertile material fraction in the core, the reactor does not need to be shut down for refueling, and the potential to better use the fuel to produce maximum fuel burn-up.

MSR is one of the six Generation IV nuclear reactor designs proposed in the Generation IV Forum (GIF) (Dwijayanto et al., 2021). The GIF is comprised of the following thirteen nations: 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 types of nuclear reactors that would be appropriate for the fourth-generation nuclear energy system (Wulandari & Permana, 2020). The MSR reactor design demonstrates considerable potential for on-line reprocessing, intrinsic safety, and high thermal-electric conversion efficiency (U.S. Department of Energy, 2002). With the Th-U fuel cycle, which can be utilized with the plentiful resource of thorium, fuel breeding can also be accomplished in both the thermal and fast spectrum (International Atomic Energy Agency, 2005).

Numerical modeling is still being used to develop this sort of reactor, given the lack of experimental facilities at the present day. However, extensive experimental research was carried out at Oak Ridge National Laboratory (ORNL) in the 1950s and 1960s, giving a basis for their viability from an experimental perspective (U.S. Department of Energy, 2002). The typical fuel salt choice, $LiF\text{-}BeF_2\text{-}ThF_4\text{-}^{233}UF_4$, has a good flow and heat transfer characteristics, a eutectic melting point at 499°C, and a low vapor pressure at working temperature (Rykhlevskii et al., 2019). As a converter reactor, the MSR might be used to transmute spent fuel from the present light water reactor (LWR).

MSRs are generally associated with thorium fuel cycle, as the initial development of the molten salt breeder reactor (MSBR) in ORNL was focused on using thorium as the fuel, enabling MSR to attain thermal breeding (Bettis & Robertson, 1970). However, MSR can be used with any type of fuel, either in uranium fuel cycle or thorium fuel cycle. Using uranium fuel cycle, a fast spectrum MSR is required to attain fissile breeding (Dwijayanto & Harto, 2024). If the MSR is designed as a converter reactor instead of breeder reactor, thermal neutron spectrum is sufficient (Carter & Borelli, 2020; Devanney, 2015). Meanwhile, by using thorium fuel cycle, the reactor can be designed either in thermal neutron spectrum or fast neutron spectrum.

Apart from MSBR, the single fluid double zone-thorium molten salt reactor (SD-TMSR) is currently under development as a thermal breeder MSR (Li et al., 2018; Ashraf et al., 2020). Two-fluid iso-breeder MSR is also being developed in the US (Moss et al., 2022), which separates the fuel zone from the breeding blanket zone. The passive compact molten salt reactor (PCMSR), a thermal breeder MSR with higher operational temperature (900 °C) compared to typical MSR (700 °C) is under development in Indonesia (Dwijayanto et al., 2021; Dwijayanto & Harto, 2024). These MSRs exclusively use thorium fuel cycle as thorium is the prerequisite of thermal breeding due to its superior neutronic properties in the thermal neutron spectrum.

This MSR technology is proposed as a potential power source for submarines due to several advantages it offers over traditional pressurized water reactors (PWRs), which are commonly used in nuclear submarines. Several other major countries also implemented maritime transport using nuclear propulsion for both military and/or civilian purposes, such as Russia operating cargo ships NS. Sevmorput and the Taimyr icebreaker, Japan operated the research ship NS Mutsu, Germany with the now-decommissioned cargo ship NS. Otto Hahn (Hirdaris et al., 2014; Khlopkin & Zotov, 1997; Mitenkov & Polunichev, 1997). To date, there are 200 operational nuclear-propulsion maritime vessels (Freire & Andrade, 2014). However, the limitation of PWR for marine propulsion lies in the requirement of highly enriched uranium, which is illegal to use in many countries. Low-enriched uranium will impose a more limited operational time and higher maintenance due to more frequent refueling. Thermal breeder MSR, on the other hand, can be operated under a low-enriched fuel scheme, coupled with thermal breeding capability, so that long operational time without the need of frequent refueling can be achieved. This shows a huge promise of using MSR in marine propulsion, including for submarines.

Various technical challenges and regulatory considerations need to be addressed before MSR technology can be deployed in submarines or any other commercial naval applications. A proper MSR core design must also be designed properly to accommodate its uses in naval territory. One of the required research projects is the reactor physics aspect, which determines whether the core design is sufficient to attain thermal breeding within an allowable size limit for marine vessels.

This research explores the neutronic aspects of thermal breeder MSR intended for submarine propulsion, as the initial research that must be conducted to properly design a nuclear reactor, using Monte Carlo-based code MCNP6.2. The MSR core model is adjusted to the specifications that suit the needs of submarines. The research contributes as a strong initial point to develop the optimum thermal breeder MSR design for marine propulsion.

Previously, MCNP software was used to model the reactor physics features of the FUJI-U3-(0) reactor (Jaradat & Alajo, 2017), MSBR (Park et al., 2015), and PCMSR (Dwijayanto et al., 2021). The effective multiplication factor ($k_{eff}$), temperature coefficient of reactivity (TCR), breeding ratio (BR), and doubling time (DT) were examined in this study.

## 2. Theory

Submarine propulsion requires a small-power nuclear reactor. The small modular MSR (SM-MSR) for submarine propulsion is designed to generate 250 MWt, corresponding to 100 MWe. Figure 1 and Figure 2 are the geometrical designs of SM-MSR in MCNP. A small core size is sufficient to generate this power level, allowing it to be placed in a submarine. The carrier salt is eutectic LiF-$BeF_2$ and the fuel salts are $ThF_4$ and small amount of $^{233}UF_4$. To improve fuel cycle performance, the virtual one-and-a-half fluid configuration is adopted. Beryllium oxide (BeO) is used as the moderator to improve core neutronic in a small-sized reactor, while high-density graphite is used as the radial reflector. Beryllium has a stronger neutron slowing-down power than graphite, ensuring that the reactor core can be designed to be more compact while maintaining a high level of neutron economy. This suits the need of compact power-generating units due to the size limitation in marine vessels, especially submarines. The disadvantage, however, is that beryllium is highly toxic. Thus, the fabrication of core materials and its handling post-irradiation must be conducted very carefully.

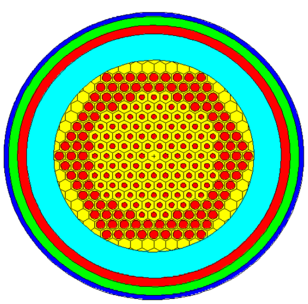

Figure 1. Small modular-molten salt reactor model visualization in MCNP (horizontal cross-section).

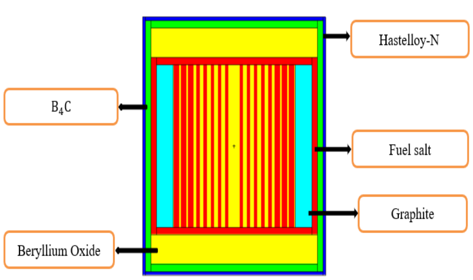


Figure 2. Small modular-molten salt reactor model visualization in MCNP (vertical cross-section).

The core operational temperature is 900 K. Because molten fluoride has a very high boiling point and a very low vapor pressure, the temperature can be achieved under atmospheric pressure. Fuel assembly is unnecessary due to the unique properties of molten salt, and thus radiation damage or fuel cladding failure do not apply to MSR (Status Report-MSR-FUJI, 2016).

The reactor vessel is in a cylindrical shape filled by hexagonal BeO moderator and graphite radial reflector blocks, making up the core structure. The primary pump circulates the molten salt through the cylindrical channel bored through the moderator blocks as the fuel channels. The heated molten salt is then transferred to the secondary coolant salt using a heat exchanger.

Through the online reprocessing system, the fuel composition concentration can be adjusted. Fission products can be removed online by helium sparging (Xe, Kr), plating on the piping surface (semi-soluble metals) and reductive extraction (other fission products including lanthanides). Similarly, fissile and fertile fuel can be added either continuously or in batch without shutting down the reactor. Refueling downtime is not necessary because there are no fuel assemblies in the core, making continuous operation possible. The fuel salt can be left indefinitely within the reactor core, as the salt is highly resistant to damage induced by radiation. Reactivity and/or power level can be managed during operations by adjusting fuel salt flow.

SM-MSR core parameters used in this study are provided in Table 1. The fuel salt composition was adopted from MSBR.

Table 1. SM-MSR core parameters.

| SM- MSR core parameters | Values |
|---|---|
| Thermal power | 250 MWt |
| 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 | 3 cm |
| Blanket channel radius | 5 cm |
| Operational temperature | 900 K |
| Fuel type | Molten salt |
| Composition | LiF-$BeF_2$-$ThF_4$-$^{233}UF_4$ |
| Molar composition (%mole) | 71.76-16-12-0.24 |
| Fuel salt density | 3.3 g/cm$^3$ |
| Fuel dilatation coefficient | 6.7 × $10^{-4}$ g/cm$^3$.K |

## 3. Methodology

The effective multiplication factor ($k_{eff}$), TCR, and kinetic parameters were among the significant reactor physics parameters that will be evaluated for the proposed SM-MSR design, using the MCNP6.2 radiation transport code (Zuhair et al., 2019), and the cross-section data used is adopted from ENDF/B-VII.0 library. The MCNP code, developed and maintained by Los Alamos National Laboratory (LANL), is the widely used code for using the Monte Carlo method (hence MC) to analyze radiation transport including neutrons and gamma rays (Shultis & Faw, 2004). MCNP is commonly used for fission and fusion reactor design, nuclear criticality safety analysis, radiations shielding analysis waste storage/disposal, detector design and analysis, used in medical physics and dosimetry, radiotherapy and burnup calculations. Considered as the "golden standard" of MC codes, the MCNP has been validated for numerous reactor types and is known to be able to model scenarios involving complicated geometry in radiative transport (Alzamly et al., 2020; Sembiring et al., 2018; Zuhair et al., 2019).

For the case of MSR, reactor physics of FUJI-U3-(0) were studied using the MCNP6 code (Jaradat & Alajo, 2017). The findings proved that MCNP6 calculation results are comparable to the original SRAC95 calculations. MCNP was also used to model the neutronic aspects of PCMSR (Dwijayanto et al., 2021), TMSR-500 (Khakim et al., 2021), and IMSR (Carter & Borrelli, 2020). Thus, MCNP is suitable for MSR calculation.

In this study, criticality was observed in the beginning of cycle (BOC). Neutronic simulation was performed using 10,000 neutrons at each cycle for a total of 250 cycles, with the first 50 cycles discarded to obtain source convergence. Five variations of fuel channel radii were simulated to obtain the optimum criticality value. The one considered as the optimum value is the channel radius with the highest $k_{eff}$. After that, the composition of the fuel in the reactor core was adjusted so that the critical core is at a low excess reactivity, around 500 pcm, with the aim is to make it easier to control its reactivity. However, no burnup calculation was conducted for this calculation, as MCNP does not support online reprocessing feature integral to MSR. However, the BR can be calculated from the reaction rate and so does the DT.

After that, TCR calculation was conducted. TCR denotes the change of reactivity against the change of temperature. TCR calculation was divided into moderator temperature coefficient (MTC), Doppler coefficient (DC), and salt density coefficient (SDC). DC was obtained by changing the temperature to 1200 K on the fuel material card. MTC was calculated by changing the temperature of the moderator to 1200 K and adjusting the (α, β) thermal scattering library to 1200 K. SDC was obtained by decreasing fuel salt density to the density at 1200 K using the coefficient listed in Table 1. The total value of TCR constituents must be negative to ensure the inherent safety of the reactor.

## 4. Result and discussion

### *4.1. Effective multiplication factor*

The SM-MSR is intended to have a long fuel cycle. To do this, the outer fuel channels (referred to as "blanket") had larger radius than the inner fuel channels (referred to as "core"). This configuration allows larger fuel loading and thereby larger thorium conversion into U-233. This way, thermal breeding can be achieved with a single fuel stream, though not as large as dual fuel stream.

To obtain the optimum core channel radius, $k_{eff}$ with various core radii were examined. The result is shown in Table 2. The optimum fuel diameter variation, as previously mentioned, is noted by its maximum $k_{eff}$. Here, we observed that the maximum $k_{eff}$ was found at the core channel radius of 3 cm. This denotes that the core is in optimum moderation condition. That is, the moderator-to-fuel ratio (MFR) is at optimal, where both the thermal utilization factor and resonance escape probability are at balanced value. Some MSRs are operated in under-moderated regions (Khakim et al. 2023; Li et al., 2018) to improve fuel utilization, at the expense of poorer SDC and slightly compromised inherent safety. In this research, the optimum MFR condition is chosen to compromise the inherent safety and the fissile breeding capability, as too much excess fissile material is unnecessary anyway.

Since the $k_{eff}$ value of SM-MSR with 3 cm core channel radius is significantly larger than target value, the fuel composition was adjusted and recalculated so that the $k_{eff}$ is approximately 1.005. The optimum fuel composition and core geometry were then used for further calculations.

Table 2. $k_{eff}$ values of SM-MSR with different core channel radii.

| Diameter (cm) | $\mathbf{k_{eff}}$ |
|---|---|
| 2 | 1.00568 |
| 2.5 | 1.03295 |
| 3 | 1.03904 |
| 3.5 | 1.03005 |
| 4 | 1.00536 |

### *4.2. Temperature coefficient of reactivity*

TCR is a critical safety parameter for the reactor and must be negative at all conditions. The overall TCR for MSR can be divided into its three constituents, as shown in Equation 1.

$$\frac{dk}{dT}\text{total} = \frac{dk}{dT}\text{DC} + \frac{dk}{dT}\text{SDC} + \frac{dk}{dT}\text{MTC} \quad (1)$$

The results of the TCR calculation are summarized in Table 3.

Table 3. Temperature coefficient of reactivity.

| Parameter | Value (pcm/K) |
|---|---|
| DC | -2.89 |
| SDC | 0.10 |
| MTC | 0.23 |
| Total TCR | -2.56 |

The DC is negative as the Doppler effect increases fertile capture when temperature increases. Compared to other MSRs, the DC is not as negative due to small core size, causing lower fissile inventory and weakening its Doppler broadening. Compared to PCMSR, which has an even smaller core, the DC is less strongly negative (Dwijayanto & Harto, 2024). The SDC is weakly positive, indicating that the core is slightly under-moderated despite it having maximum $k_{eff}$ compared to other core radii. Compared to SD-TMSR (Ashraf et al, 2020), PCMSR (Dwijayanto & Harto, 2024), and MSBR (Rykhlevskii et al., 2019), the SDC is much more negative, indicating that other MSRs mentioned are in under-moderated conditions. Under-moderated core in MSR will result in an increase of reactivity when fuel salt density decreases as MFR increases. Since the SDC is only weakly positive, the under-moderated region is not severe and can be compensated by negative DC value.

Meanwhile, MTC is also slightly positive, due to the use of graphite reflector that radially surrounds the core. In MSR moderated by graphite, when using U-233 as its fuel, MTC tends to be positive as the spectral shift of graphite coincides with U-233 resonance (Mathieu et al, 2006). This phenomenon also occurs in SD-TMSR, MSBR, and PCMSR, with the exception that the positive MTC is more prominent in those MSRs due to their use of graphite in their full core.

As the primary moderator is BeO, while graphite acts as a radial reflector, the positive MTC induced by graphite is not as large as other MSRs. BeO does not impose positive moderator temperature feedback, unlike graphite. Therefore, apart from improving neutron moderation so that compact reactor core can be designed, using BeO as moderator also improves inherent safety of the SM-MSR.

Overall, the TCR is sufficiently negative, thus satisfying the inherent safety criteria. This way, when the reactor temperature increases, the reactor power decreases, ensuring that power excursion accident due to temperature increase will not happen. The obtained value is more negative compared to graphite-moderated SD-TMSR (Li et al., 2018) and PCMSR (Dwijayanto et al., 2021), which was slightly below -2 pcm/K.

### *4.3. Th-U breeding ratio and doubling time*

BR can be interpreted as a ratio between the capture of fertile fuel and the absorption of fissile fuel, as formulated in Equation 2 (Zou et al., 2015).

$$\mathrm{BR} = \frac{\mathrm{R_c}(^{232}_{90}\mathrm{Th}) + \mathrm{R_c}(^{234}_{92}\mathrm{U}) + \mathrm{R_c}(^{233}_{91}\mathrm{Pa})}{\mathrm{R_a}(^{233}_{92}\mathrm{U}) + \mathrm{R_a}(^{235}_{92}\mathrm{U})} \quad (2)$$

where $\mathrm{R_c}$ stands for the fertile nuclide's neutron capture reaction rate and $R_a$ for the fissile nuclide's neutron absorption rate. Since there are just Th-232 and U-233 at the BOC, Equation 2 can be modified as follows.

$$\mathrm{BR} = \frac{\mathrm{R_c}(^{232}_{90}\mathrm{Th})}{\mathrm{R_a}(^{233}_{92}\mathrm{U})} \quad (3)$$

The geometry of the reactor and the molten salt reprocessing period are two elements that have the most influence on the breeding character of MSR at equilibrium (Li et al., 2018). They influence parameters such as fuel composition, neutron flux, fuel residence time, and the neutron capture to fission ratio. Achieving a breeding ratio greater than 1 indicates that the reactor is producing more fissile material than it consumes, which is a desirable characteristic for sustainable and efficient operation.

In this investigation, the BR at the BOC was calculated to be 1.057. Thus, from the calculated result, the BR satisfied the thermal breeding criterion. This value is lower than SD-TMSR, which clocked at 1.104, and comparable to optimized PCMSR at 1.05. The use of BeO moderator proves to be beneficial in obtaining BR larger than unity, since a similar-sized geometry PCMSR (Dwijayanto et al., 2021) failed to achieve thermal breeding condition using graphite as moderator, despite the fuel fraction (30%mole) is significantly larger than SM-MSR (12.24%mole).

DT is calculated as the time needed for the excess fissile material to be accumulated in sufficient amount to start another identical reactor. At the BOC, DT can be estimated using Equation 4.

$$DT = \frac{M_0}{BG.F.C} \quad (4)$$

Where $M_0$ is the initial fissile inventory, BG is breeding gain (BR – 1), F is annual fissile consumption, and C is load factor. The desired MSR fuel cycle must satisfy low initial inventory and short DT.

From the obtained data, assuming C = 1, the DT value is found to be around 52 years. This long DT is primarily caused by low reactor power, which influences the fissile consumption and in turn reducing excess fissile production. Low BR due to small core size further exacerbated this issue. As a comparison, SD-TMSR has a DT of 16 years (Li et al., 2018), since it has large thermal power (2,250 MWt). Optimized PCMSR, on the other hand, has a DT of 13.67 years (Dwijayanto & Harto, 2024), since the reactor power is twice as large as SM-MSR.

Despite long DT, the SM-MSR is observed to be able to generate more fissile fuel than it consumes. Therefore, SM-MSR is qualified to be considered as thermal breeder reactor. This ensures that the nuclear submarine is self-sufficient in fulfilling its own fuel, without the need of external refueling.

## 5. Conclusions

Study on the SM-MSR has been carried out through a series of calculations using MCNP6.2 software and ENDF/B-VII.0 cross-section library. The calculation results show that the SM-MSR core can achieve thermal breeding with BR value at 1.057, although with long doubling time at 52 years. The TCR is -2.56 pcm/K, satisfying the inherent safety criteria. Therefore, the neutronic of SM-MSR core fulfils both thermal breeding for long operational time and negative TCR as a means of maintain operational safety. With small core size, this SM-MSR can be suited for submarine propulsion, with the intention to reach a significantly longer operational range than their Diesel-powered counterparts.

## Conflict of interest

The authors have no conflict of interest to declare.

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