Journal of Energy Research and Reviews

Volume 15, Issue 4, Page 18-33, 2023; Article no.JENRR.110642
ISSN: 2581-8368

Proposal of Application of a Simple Analysis Code DYMOS for Accident Analyses of Molten Salt Reactors

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Authors' contributions

This work was carried out in collaboration among all authors. All authors read and approved the final manuscript.

Article Information

DOI: 10.9734/JENRR/2023/v15i4321

Open Peer Review History:
This journal follows the Advanced Open Peer Review policy. Identity of the Reviewers, Editor(s) and additional Reviewers, peer review comments, different versions of the manuscript, comments of the editors, etc are available here: https://www.sdiarticle5.com/review-history/110642

Review Article

Received: 15/10/2023
Accepted: 21/12/2023
Published: 27/12/2023

ABSTRACT

The Molten Salt Reactor (MSR) possesses unique characteristics, including the circulation of liquid fuel salt in-core and ex-core, and the absence of cladding tubes of fuel rods. To perform safety analyses for accidents and transients, the selection of appropriate models is crucial. Proposed models vary from the coupling of a three-dimensional (3-D) nuclear reactor model and a detailed thermal/hydraulic (T/H) model to a simple point reactor model coupled with a lumped parameter model of the T/H system. The authors have been investigating the development of a simple and accurate model that can be utilized during the design stage and licensing evaluation, while also providing transparency, which means that models can be easily understandable by experts and reproduced in licensing evaluation. Given the distinctive features of the MSR, the peak heat flux or fuel cladding peak temperature is not a requirement. Instead, the most decisive parameter for safety evaluations is the fuel salt temperature in the fuel salt boundary. As demonstrated in this paper, the outlet of the MSR core consistently displays the highest temperature. Based on the afore-mentioned prospects for both nuclear and T/H systems, the authors have

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developed a simple safety and transient analysis code for MSR (DYMOS). The DYMOS code has been verified for MSRE experiments, as described in this paper. However, the previous verification was limited to small experimental MSRs, and there is a lack of verification for large reactor systems. This paper shows that the DYMOS code is applicable to these larger reactors. In other words, the main objective of this article is not to claim the originality of the model of DYMOS code, but to propose applicability of such simple code to large reactor systems.

Keywords: Molten Salt Reactors - MSR; accident analysis; DYMOS code; reactivity initiated accident; loss of flow accident.

1. INTRODUCTION

Molten Salt Reactor (MSR) concept has recently been considered as one of the candidates for the generation IV nuclear power systems. MSRs have many advantages such as improved safety, proliferation resistance, resource sustainability and waste reduction. Recently, efforts to develop MSR have been in progress in many countries [1,2]. One of these activities is to perform safety analyses for accidents and transients, which is indispensable for MSR design and licensing. For these purposes, many analytical codes have been proposed [3,4]. Some studies have been reported on transient behaviour for MSFR [5,6]. Transients are analysed using mainly neutronics and thermal-hydraulics coupled computational fluid dynamics (CFD) code, e.g. COUPLE code [7], HEAT code [8], COMSOL code [9,10], OpenFOAM code [11–14]. In some cases, system codes like TRACE [15] is used. Recently, the PROTEUS-NODAL code which is a 3D nodal transport code has been developed at Argonne National Laboratory and benchmarked with Trace and CFD [16].

As described above, MSR has several unique features, where liquid fuel salt is circulating in-core and ex-core and there are no cladding tubes of fuel rods. To perform safety analyses for accidents and transients, the selection of models is crucial. Models of these codes vary from one that uses coupling of a three-dimensional (3-D) nuclear reactor model and detailed thermal/hydraulic (T/H) CFD model to a simple point reactor model coupled with lumped parameter model of T/H system.

The validity of fine structures of the transients is quite important for conventional reactor systems using solid fuel rods. Because in these reactors, 3-D evaluation for time-dependent local heat flux and cladding temperature of fuel rods is required to confirm that these parameters are within safety limits of fuel integrity during accidents or transients.

Meanwhile, in safety analyses for MSR, there are no requirements for peak heat flux or fuel cladding peak temperature, because there are no cladding tubes in MSR. For the safety evaluation of MSR, the most decisive parameter is the fuel salt temperature in the fuel salt boundary. As is explained in Section-2, the highest temperature is expected at the outlet of the MSR core under accidents with fuel salt flow. That is, 3-D power distribution is not required. Also, this suggests that a point kinetics model can be applied to the codes.

Based on the above prospects for both nuclear and T/H models, a simple safety and transient analysis code for MSR (DYnamics for MOlten Salt reactors: DYMOS) has been developed by the authors, and DYMOS code is verified for MSRE experiments as described later.

However, these verifications are for a small experimental MSR, known as MSRE. MSRE operated at the maximum thermal output of 10MW. The size of the reactor is such that the height is 1.7m and the diameter is 1.4m [17]. Verification for large reactors is missing. This paper provides the information that the DYMOS code can be applied to these large reactors. The target plant is the MCSFR, which has 700 MWt reactor power with a reactor volume similar to that of the MSFR in the EU [18]. Brief description is given in Appendix.

The main objective of this article is not to claim the originality of the model of DYMOS code, but to propose applicability of such simple code to large reactor systems. Then, this code can be applied in the actual licensing process by both reactor designers and licensing bodies.

2. FEATURES OF SAFETY ANALYSIS OF MOLTEN SALT REACTOR SYSTEM

As discussed above, there is no requirement for local heat flux or temperature in transients of MSRs with fuel salt flow. The most important information is the highest fuel salt temperature in the fuel boundary, which is expected as the outlet fuel temperature. Based on heat balance in an MSR core, the outlet temperature is simply obtained as follows.

$$T_{out} = T_{in} + \frac{P}{CG}, \quad (1)$$

Where

T_{out}: Outlet temperature
T_{in} : Inlet temperature
P: Reactor power= $\int_{whole\ core} p(r,z)drdz$
$p(r,z)$: Power distribution in the reactor
C: Specific heat of fuel salt
G: Mass flow rate

The highest fuel temperature is obtained through the integration of reactor power in the whole reactor region, which is the total power generated in the reactor core. This fact means that detailed power distribution is not required when molten salt flow is maintained. The highest fuel temperature is the most decisive parameter for the integrity of the fuel salt boundary. From this point of view, it is expected that the highest fuel salt temperature could be estimated using a simple system model of a point reactor model coupled with heat transfer systems of lumped parameter models.

Based on these prospects, DYMOS code has been developed by the authors, and its accuracy is verified for MSRE experiments as shown in Section 3.4. But these verifications are for small experimental MSR, and verification for large reactors is needed. Recently, safety analyses using 3-D power distribution and detailed T/H models for a large reactor system are reported [18]. A comparison between the results of this detailed model and DYMOS results for the large reactor system is provided in Section 5.

3. DESCRIPTION OF DYMOS CODE

3.1 Plant Model in DYMOS Code

A simplified plant model in the DYMOS code is shown in Fig. 1. The concept of MSR is quite different from those of conventional concept, which use solid type nuclear fuels. As described above, the advantages of the concept are improved safety, proliferation resistance, resource sustainability and waste reduction. MSR under consideration in this article uses high temperature molten fluorides or chlorides salts as nuclear fuel and secondary cooling system. Fuel salt contains fissionable materials. In the reactor fuel salt becomes critical and make fissions to generate energy. Generated nuclear heat in the reactor core is transferred to a (primary) heat exchanger by a fuel salt loop, and its heat is transferred to, for an example, a steam generator (SG) by a coolant salt loop, and finally used at a turbine/generator, if this plant applies a steam turbine system.

As the fuel salt is circulating the primary loop, part of delayed neutron precursors flow out of the reactor and part of them will return to the reactor. This feature distinguishes the nuclear kinetics of MSRs (See section 3.2). However, after the heat is transferred to the secondary system and thereafter, the system can be understood as ordinary heat transferring systems.

From these basic consideration, DYMOS code focuses on the fuel salt loop in Fig. 1.

3.2 Reactor Kinetics

DYMOS code adopts one-point kinetic equations for the reactor power as shown below, which is almost identical to the model used in LWR licensing. The following equations are common in textbooks of reactor physics [19]. The unique difference in MSR is that it considers the circulation of delayed neutron precursors in fuel salt, as is shown in the third and fourth terms of the second equation proposed by ORNL [20]. The well-known kinetic equations for solid fuel reactor are modified by considering the flow-out loss and the flow-in gain of delayed neutron precursors [19]. They are expressed in the third and fourth terms in equation (3), respectively.

The newly defined parameters, namely, β_{eff} and β_{loss} are for the convenience of simplification of equation (2).

$$\frac{dn(t)}{dt} = \frac{\rho - \beta_{eff}}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \quad (2)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_{s,i}}{\Lambda} n(t) - \lambda_i C_i(t) - \frac{1}{\tau_C} C_i(t) + \frac{1}{\tau_C} C_i(t-\tau_L)\exp(-\lambda_i \tau_L) \quad (3)$$

$$\rho = (tf - tf_d) * \alpha_f + \rho_{ex} \quad (4)$$

$$\beta_{eff} = \sum_{i=1}^{6} \beta_i = \sum_{1=1}^{6} \left(\beta_{s,i} - \beta_{loss,i}\right) \quad (5)$$

$$\sum_{i=1}^{6} \beta_{loss,i} = \sum_{i=1}^{6} \beta_{s,i} \left[1 - \frac{\lambda_i}{\lambda_i + \frac{1}{\tau_C}\{1 - \exp(-\lambda_i \tau_L)\}} \right] \quad (6)$$

$n(t)$: Number of neutrons
$C_i(t)$: Number of delayed neutron precursor of i-th group
λ_i : Decay constant of i-th group delayed neutron precursor
β_{eff} : Effective delayed neutron fraction
$\beta_{s,i}$: Delayed neutron fraction for static reactor of i-th group
β_{loss} : Loss fraction of delayed neutron by fuel salt flow
Λ : Neutron generation time
τ_C : Fuel transit time in the reactor core
τ_L : Fuel transit time in the loop
ρ : Reactivity
t_f, t_{fd} ; Fuel temperature and design value
α_f ; Fuel temperature reactivity coefficient
ρ_{ex} ; External reactivity

3.3 Heat Transfer Model

Energy generated by fission is transferred to a primary heat exchanger through a fuel salt loop and then transferred to an SG or a secondary heat exchanger and finally to a power conversion system. The respective temperatures can be calculated based on energy flow. The present model is a quite simple one expressed with each system as one point. Although the current DYMOS code can calculate SG, SG is simply treated as the final heat sink in this study to compare the reference design. DYMOS code assumes only one loop, however, it can treat partial loss of fuel salt flow by adjusting the effective flow rate in the reactor core.

Related equations are expressed as follows. The equations are taken after or further simplified from the models presented in the references as follows [21,22].

In the following equations, there are no parameters and heat transfer for graphite moderators used in thermal reactors. However, the DYMOS model for thermal reactors with graphite moderators is verified as described in Section 3.4. This model includes similar equations of heat transfer between fuel salt and the graphite.

$$M_f C_{pf} \frac{dT_{out}(t)}{dt} = F_l C_{pf} (T_{in}(t - \tau_L) - T_{out}(t)) + P_r(t) \quad (7)$$

$$M_f C_{pf} \frac{dT_{in}(t)}{dt} = F_l C_{pf} (T_{in}(t) - T_{hx}(t)) - A_{hx} H_{hx} (T_{in}(t) - T_{hx}(t)) \quad (8)$$

$$M_f C_{pf} \frac{dT_{in}(t)}{dt} = F_l C_{pf} \left(T_{hx}(t) - T_{in}(t)\right) - Q_{sink}(t) \quad (9)$$

T_{out}, T_{in}, T_{hx} : Fuel outlet, inlet and heat exchanger temperature, respectively

M_f : Fuel mass
C_f : Specific heat of fuel
F_l : Fuel flow mass rate
P_r : Thermal power output=fission power + decay heat
Q_{sink} : Heat sink in the primary heat exchanger

Note: The term, $C(t - \tau_L)$, in Equation (3) is included to consider of the piping effect due to the circulation of molten slat in the system. Several models were used to represent salt transport in the piping. In this study, the simplest model of a first order time lag of τ_L is used as in explained in Ref [21]. This model is also applied to the returning salt temperature, $T_{in}(t - \tau_L)$, presented in Eq. (8) above.

The above differential equations (2) to (9) are solved by numerical integration and its time interval of 0.01 seconds is confirmed to be short enough.

Also, these simple nuclear models and heat transfer models would provide transparency, which means that models can be easily understandable by experts and reproduced in licensing evaluation.

3.4 Verification for MSRE Experiments

Verification of the DYMOS code against experimental data of MSRE, which was operated at Oak Ridge National Laboratory (ORNL) in the 1960s adopting fluoride fuel salt with graphite moderator, is already reported by the authors [23,24,25]. In a summary, DYMOS

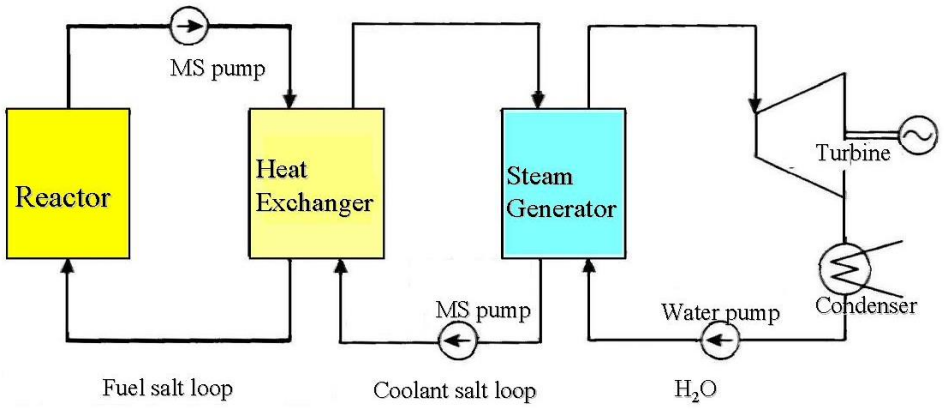


Fig. 1. DYMOS model for MSR power plants [18,19,20]

code shows good agreement with MSRE results, for the MSRE startup experiment and coast-down experiment at zero-power, and also reactivity insertion experiments at 5MW and 8MW operations.

There is a benchmark report for MSRE zero-power experiments, where various models are compared with MSRE data [26]. DYMOS results are also within the variation of various models. As for experiments at power operations, there is another paper using a 3-D detailed multi-physics model [27]. DYMOS shows similar results. The above observation would support that a simple one-point model can be applied to MSR safety analysis.

However, MSRE is a small experimental reactor with a power output of around 10MWt. Thus, it is desirable to verify DYMOS code for a large power plant. Transient analyses by 3-D T/H detailed codes for such plants are available in the reference [18] shown in the next section. The authors provide a comparison of both results in the following sections.

4. COMPARISON OF THE RESULTS AND DISCUSSIONS

Results of accident analyses and the analytical mode is presented in Ref. [18]. For the convenience of readers, main body of the reference is re-cited in the Appendix.

In Ref. [18], it is claimed that there are four typical transients to show safety characteristics of MSR, which are Loss Of Fuel Flow (LOFF), Loss Of Heat Sink (LOHS), Toal Black Out (TBO), and Reactivity Initiated Accident (RIA). The reason of selecting these transients is briefly discussed in Section6 of the present paper [28].

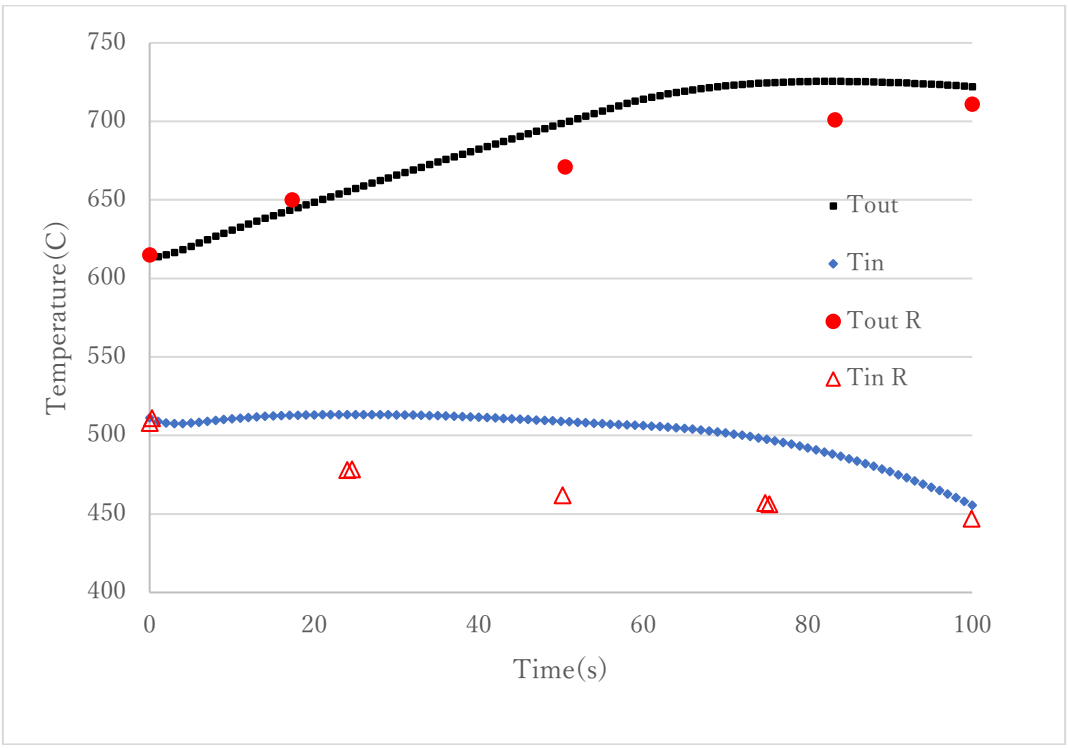
As is explained in Section 2, the outlet temperature of fuel salt is increased when power is increased or flow is decreased. Therefore, it is enough to study RIA as a representative power increase accident and LOFF, LOHS and TBO as representative of flow decrease accidents.

As for RIA analysis, one of the authors has experience with MSR power plant FUJI [27], and as for LOFF at a larger plant MSBR, there is another paper [28, 29]. Based on these studies, a similar approach is applied in the present study.

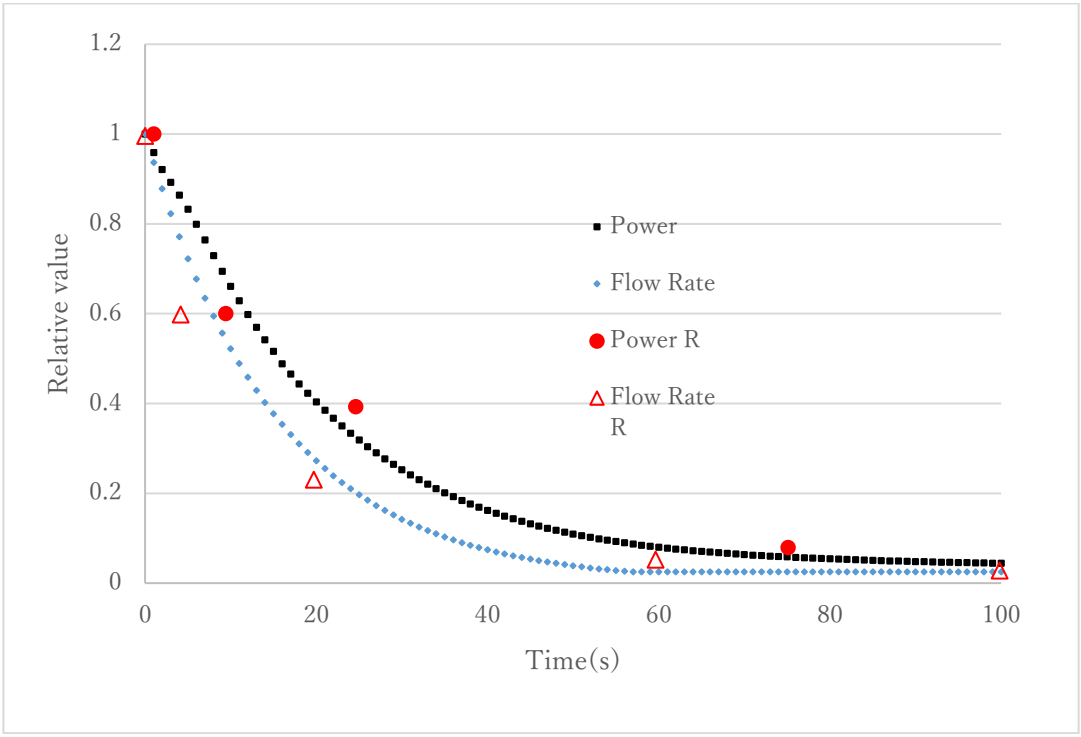
The numerical results of the reference simulation code are obtained by a digitizer from the figures in the reference [18], which are indicated by character “R”. The accident transients by DYMOS are calculated using the same conditions as those of refence [18].

Additional explanation is given briefly as follows. The flow rate of the fuel salt for DYMOS calculation is obtained from the data in reference [18]. Also, the transient of heat sink is approximated to follow the power reduction given in reference [18]. Note that these parameters are design dependent and could be known in advance.

The transients for LOFF (100% loss of flow) are compared in Fig. 2. In Fig. 3, the transient for the loss of flow of 25% is shown. Figs. 4 to 6 show comparison of LOHS, TBO and RIA, respectively. Note that the comparisons TBO are made for a short time intervals after the initiation of accidents in which the nuclear power has died out. These accidents could be caused by loss of electricity or troubles in electrical circuits.

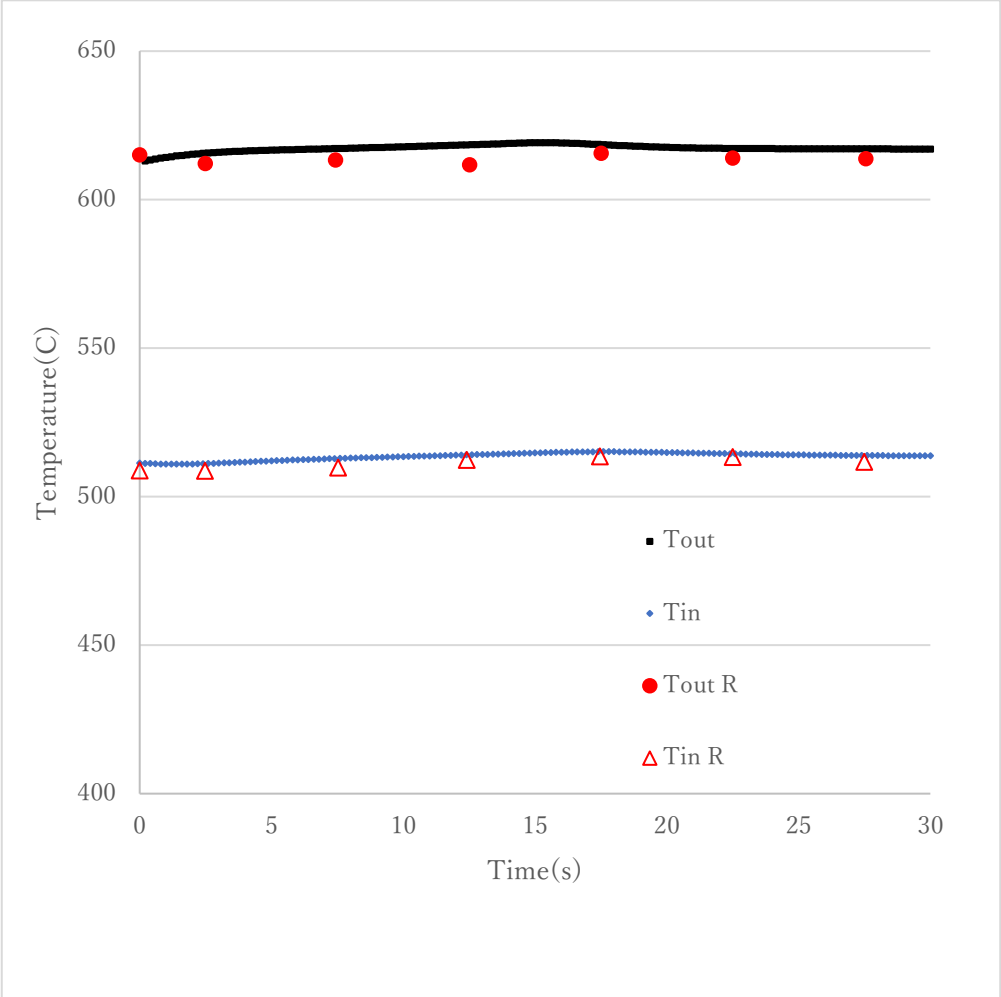


(a) Fuel salt temperature

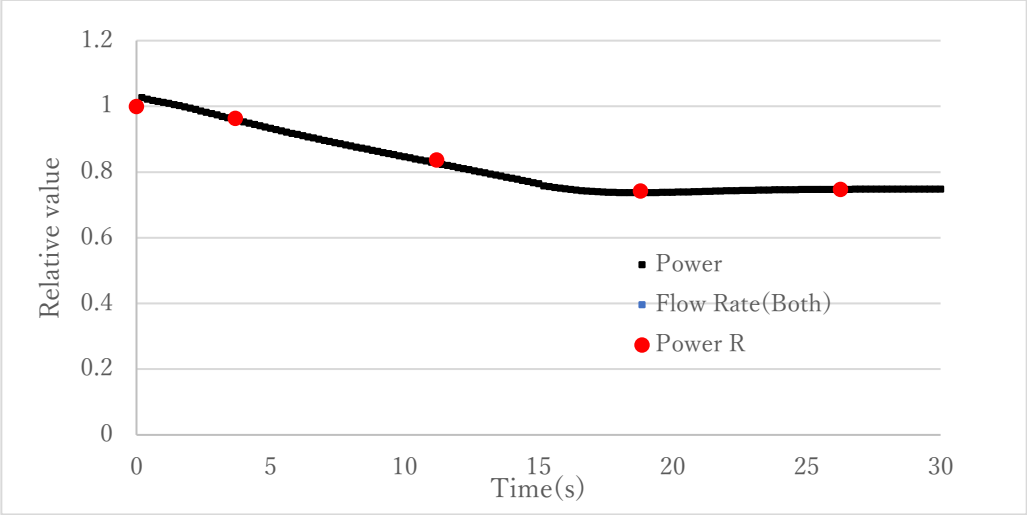


(b) Relative power and fuel flow-rate

Fig. 2. Comparison of system transient of Loss of Fuel Flow

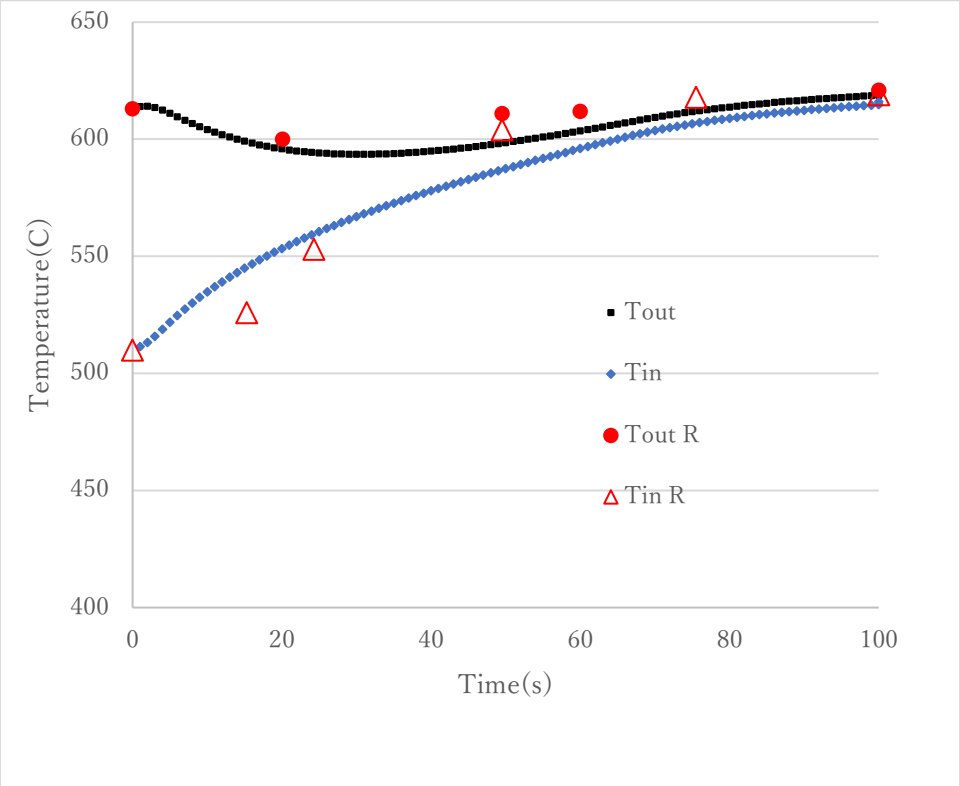


(a) Fuel salt temperatures

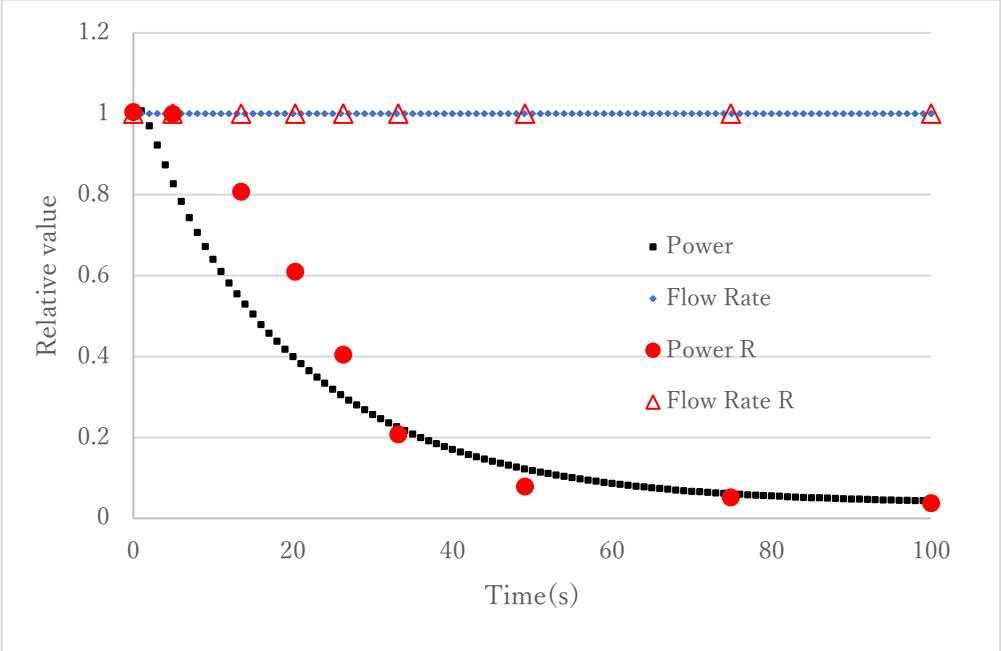


(b) Relative power and fuel flow-rate

Fig. 3. Comparison of transient of Loss of 25% Fuel Flow

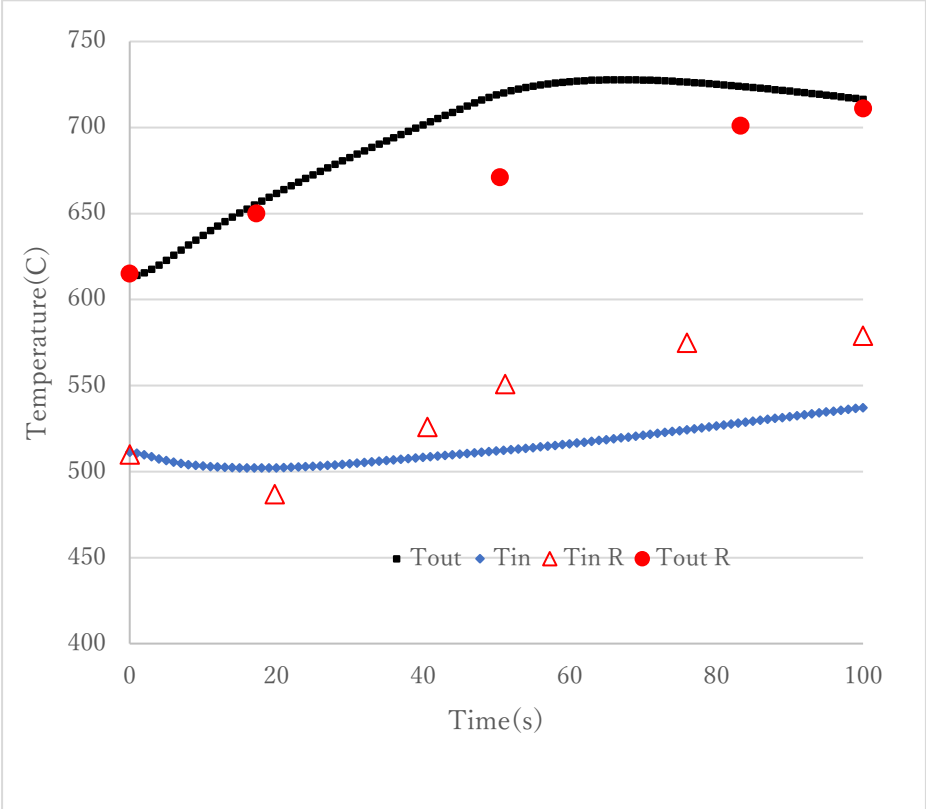


(a) Fuel salt temperatures

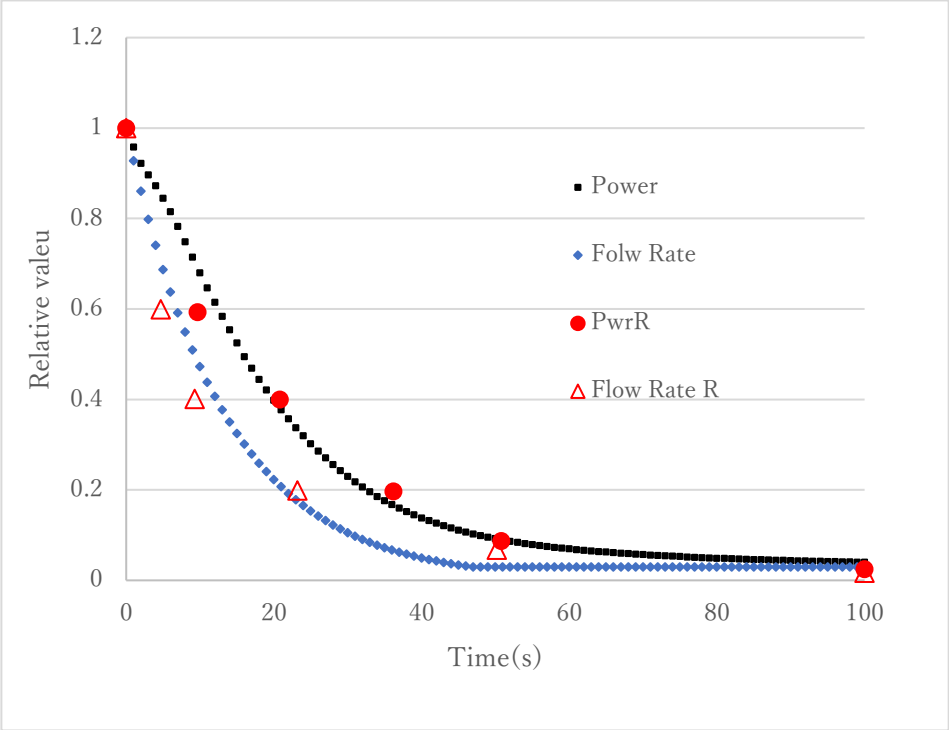


(b) Relative power and flowrate

Fig. 4. Comparison of transient of Loss of Heat Sink

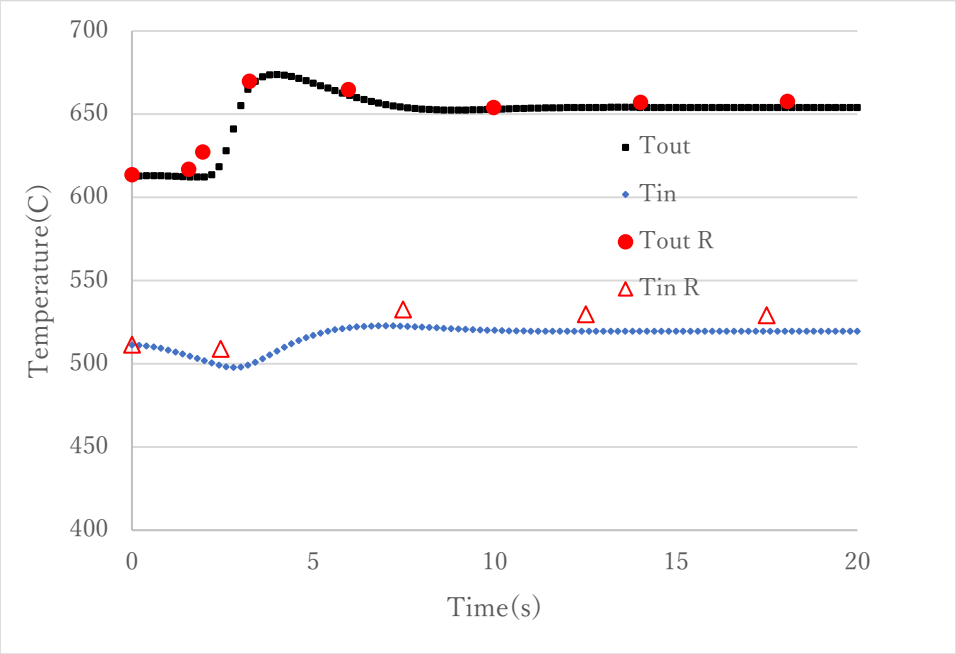


(a) Fuel salt temperatures

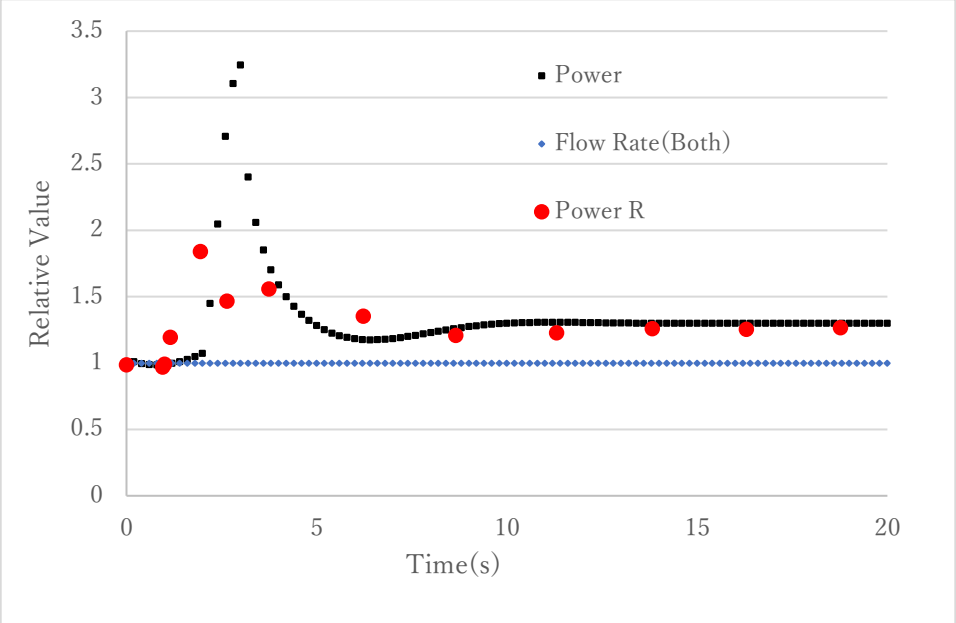


(b) Relative power and flowrate

Fig. 5. Comparison of transient of Total Blackout



(a) Relative power and fuel flow-rate



(b) Relative power and flow-rate

Fig. 6. Comparison of transient of Reactivity Initiated Accident
(Reactivity of 0.00218 is added in one second and maintained)

On the other hand, RIA in MSRs could be caused by malfunctions of fuel composition adjustment system or control rod drive system in the plants if the plant is equipped with control rods. In the RIA transient, when the heat sink is stay constant, the reactor power is finally stabilized at the same level of the heat sink. However, the result in Ref. [18], the reactor power increases and stays at about 30% higher than the nominal power.

This might be caused by the fact that the analytical model in Ref. [18] consists of other heat transfer systems. Thus, it might have large thermal inertia and can absorb more heat than the nominal power. Based on this consideration, RIA transient is analysed by increasing the heat sink about 30% of the nominal value.

As can be seen, highest fuel temperatures in all transients agree within a few degrees Celsius except for the LOFF, which result shows about 13 degrees difference. In other words, the results obtained by DYMOS all show good agreement with small discrepancies from those obtained by the detailed models. However, the authors would like to show that DYMOS can evaluate the highest fuel temperature within smaller discrepancy form that of detailed analysis codes. The authors understand that the discrepancy still exists but would continue to improve the accuracy when comparable data are available.

5. POSSIBLE ACCIDENTS IN MSR

In order to apply DYMOS code to transient or accident analysis, it is necessary to investigate possible accidents in MSRs at first. Based on previous authors study [30,31], possible accidents in MSRs are categorized as shown in Table 1. In this table, accidents by external causes such as earthquake or tsunami are not included, because they are studied by other codes. That is, only internal cause accidents are considered.

Regarding accidental causes such as overpressure or overheating, they are caused by power increase accident (Type-1) or flow decrease accident (Type-2), because temperature rise or enthalpy rise is proportional to power and inversely proportional to flow. Besides these two accidents, fuel salt leak accident (Type-3) may be caused by other mechanical failures of the primary loop boundary. Therefore, the above three types of accidents must be considered. Of course, some of the second and the third types of accidents may cause a reactivity increase as a result. Meanwhile, the first accident type is usually called a reactivity initiated accident (RIA), because it is initiated by the insertion of positive reactivity at first.

Besides these three categories, the fourth category, named “other accidents” (Type-4) is considered. These accidents are mostly specific to the MSR.

Based on this table, DYMOS code can evaluate accidents mostly for Type-1 and Type-2. That is, Type-3 and 4 should be evaluated by other codes. From this point of view, studied cases in Section-5 are selected representing Type-1 and Type-2.

Table 1. Accident category

Type	Possible accidents in MSRs
1	Power increase accident or RIA (Reactivity initiated accident)
	Control rod withdrawal/ejection accident
	Cold-loop startup accident
	Secondary salt flow increase accident
	Secondary salt temperature decrease accident
	Fuel salt flow increase accident
	Fuel salt filling accident
	Excessive fissile addition or fertile removal ccident
	De-pressurization accident
	Fissile precipitation accident
	Graphite loss accident
	Fissile penetration to graphite accident
2	Flow decrease accident
	Pump trip accident
	Pump seizure accident
	Flow blockage accident
	Loss of secondary salt cooling accident
	Loss of decay heat cooling accident (in core)
	Loss of decay heat cooling accident (in drain tank)

3	Fuel salt leak accident Primary loop break accident
4	Other accidents Steam-generator break accident Recriticality accident in drain tank Fuel salt freeze accident Secondary salt freeze accident Remelt accident Freeze valve failure accident Graphite fire accident Wigner effect accident Off-gas system failure accident Reactor oscillation accident Fuel salt or beryllium release accident Rupture of containment accident

6. CONCLUSION

As explained above, DYMOS code already shows good agreement for MSRE experimental results. The present study shows that DYMOS code also shows good agreement with the results of detailed T/H 3-D codes for large-size plant. As shown and discussed above, the most important parameter in MSR safety design and licensing is the highest fuel salt temperature, and DYMOS code shows good agreement with that obtained by the referenced detailed model. However, it could be pointed out that some discrepancies are seen. As the detailed information of heatsink is not known, it was approximated based on engineering judgement in this analysis. The authors believe that when the heatsink is correctly given or evaluated in the other means of non-nuclear heat transfer analysis, the DYMOS can evaluate the MSR transients with higher accuracy.

Thus, safety analyses for MSR could use a simple point model. Because they require no local heat flux or cladding temperature of the fuel rod but the highest fuel temperature in the primary system.

The great advantage of such simple models is to provide transparency in safety evaluation in licensing procedures, besides reduction of computer burden. For example, the computing time required for each of the transients is in seconds on the personal computers at hand. Thus, it can be used to perform various parameter studies in a short time.

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COMPETING INTERESTS

Authors have declared that no competing interests exist.

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APPENDIX

Reference plant and the accident analysis model [18]

A.1 Plant description of MCSFR

The MCSFR referenced without safety rods in the present study has 700 MWt reactor power with a reactor volume similar to that of the MSFR in the EU. The MSFR using fluoride fuel assumes a thermal power density of about 300 MW/m3, which is close to the power density of an LMFBR with solid fuel. However, in the present study a low thermal power density of about 70 MW/ m3 is assumed. The fuel salt is assumed to be chloride, which has a melting point lower than that of the fluoride fuel salt. The molten fuel salt and the molten coolant salt assumed in the present study consist of NaCl-MgCl2UCl3/4-(PuCl3-HNCl3) and NaCl-KCl-MgCl2, respectively. In the secondary heat transport system (HTS), potassium chloride is contained to lower the melting point.

A.2 Detailed analysis model in reference

In the reference analytical model, two well-known codes, namely RELA5-3D and FLUENT, are coupled in such a way that the fuel salt behavior in the core is calculated by the FLUENT code with inlet boundary conditions calculated by the RELAP5-3D, and the reactor power computed by FLUENT is transferred to RELAP5-3D to calculate the plant system. Both codes are modified or implemented of the function to calculate the delayed neutron behavior and the obtained results are validated against the experimental data of MSRE at ORNL. A detailed description is available in reference [18].

The behavior of neutrons in the core is based on one-point kinetics. The kinetic parameters under the steady state conditions are analyzed using the values shown in Table A1 and A2 [18]. The reactor power distribution is assumed to be a function of cosine in the axial direction and a Bessel function in the radial direction of a cylindrical core model. The decay heat is calculated using a well-known equation proposed by the American Nuclear Society [32] for light water reactors.

A.3 Target system

A schematic diagram of the conceptual Molten Chloride Salt Fast Reactor in the present study is illustrated in Figure A1[18]. This study assumes a four-loop system. Table A1 shows the dimensions of the primary Heat Transport System (HTS). The reactor core is a simple cylinder with conical components at the top and bottom. Hot legs are provided at the top of the reactor core. The primary HTS consists of the reactor core, gas treatment equipment (Not shown in Figure A1), a fuel salt pump, and a fuel salt to coolant salt heat exchanger (FCHX). The secondary HTS consists of an FCHX of the secondary side, a coolant salt pump, heat storage tanks, a gas heat exchanger (HX), and a decay heat removal system (DHRS) with an air cooler (AC). The heat storage tanks are omitted in reference [18] because the volume is undetermined and the tanks are required for load-following operations except for daily load-following. Thus, this system is also also omitted in DYMOS.

The discretized equations are coded using the C-language in the user-defined function (UDF) and incorporated into FLUENT by compiling the UDF. The changes in the core transit time and the exterior loop transit time are calculated using the flow rate in the combined UDF, and the kinetic parameters are updated in every time step. The temperatures of mesh elements in the core calculated by FLUENT are collected to calculate the average temperature of the core. The reactivity is calculated based on the change in the average fuel salt temperature, which is reflected in the reactor power.

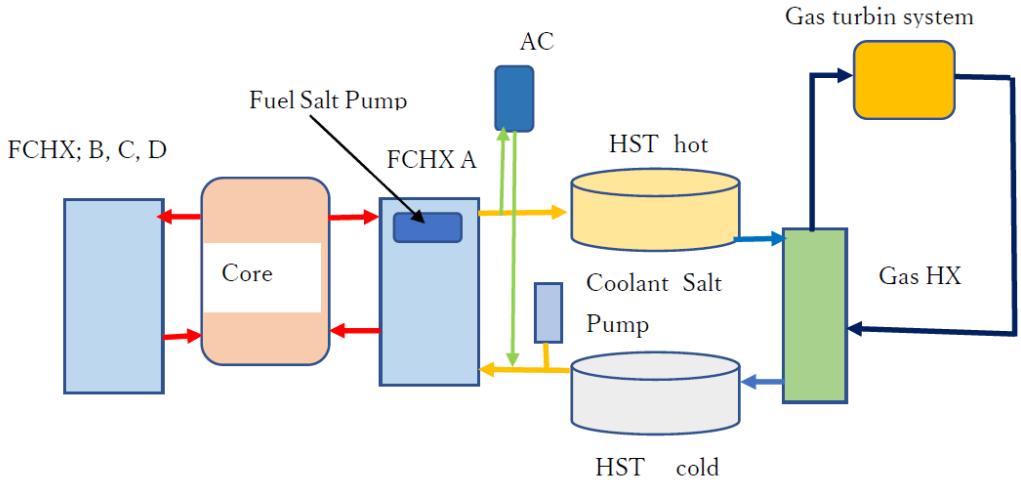

Fig. A1. Schematic of a referenced conceptual molten chloride fast reactor [18]
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Table A1. Kinetic parameters [18]

Group	β_i (Static)	$\lambda_i (s^{-1})$
1	0.00009	0.01338
2	0.00062	0.03068
3	0.00056	0.11671
4	0.00135	0.30604
5	0.00078	0.87593
6	0.00026	2.94183
Prompt neutron lifetime: 5.65×10^{-7} (s^{-1})		

Table A2. Dimensions of the primary heat transfer system [18]

Component	Dimension	Remarks
Core diameter	2.3 m	
Core height	2.4 m	Cones with 0.1 m
Fuel transit time in the reactor core, τ_c	2.46 s	At full flow
Fuel transit time in the loop, τ_L	2.69 s	At full flow
Hot-leg inside diameter (ID)	0.4 m	4 Loops, 4.6 m in length
FCHX channel (width/height)	10 mm/7 mm	Number of channels 7200 Length 3m
Cold-leg ID	0.4 m	2.4 m in length
Total volume	9.7 m^3	
Mass flow rate	3075 kg/s/loop	

Peer-review history:
The peer review history for this paper can be accessed here:
https://www.sdiarticle5.com/review-history/110642