INTRODUCTION

Over the past few years, the rapidly rising interest in Molten Salt Reactor (MSR) technology has led to an increased need to develop advanced computational tools for MSR modeling and simulation. Compared with some of the other advanced reactor technologies, there is a lack of capabilities to conduct the transient safety analysis for MSR systems as no mature system analysis codes are available in the literature. Besides the salt chemistry and thermal-hydraulic properties, the fuel moving with the salt in the primary cooling system introduces extra challenges on the MSR transient simulation.

System Analysis Module (SAM) is being developed at Argonne National Laboratory as a modern system-level modeling and simulation tool for advanced non-light water reactor safety analyses [1]. It utilizes the object-oriented application framework MOOSE [2] to leverage the modern software environment and advanced numerical methods. In support of DOE’s missions, the capabilities of SAM are being extended to enable the transient modeling, analysis, and design of MSR systems. The objective of this summary is to present the preliminary development of MSR transient simulation capability in SAM.

APPROACHES

The MSR transient simulation approaches being developed for SAM include four parts: Point-Kinetics model with delayed neutron precursor (DNP) drift; DNP drift model in the cooling system; decay heat generation in the core and cooling system, and reactivity feedback modeling.

Once fully developed and implemented in SAM, the new capabilities will be examined with MSR design basis accidents. The initial analysis will focus on two types of transients: 1) loss of flow in the primary and/or secondary cooling systems; 2) reactivity driven power change. The Unprotected Loss of Flow (ULOF) transient is initiated by the loss of forced circulations in primary and secondary systems due to pump failure. The mass flow rate decreases from the full level to the natural circulation level. The loss of flow results in two effects on MSR [3]: the reduction of the loss of delayed neutrons from the core adds positive reactivity feedback; the temperature of fuel salt increases. Under the Unprotected Transient Overpower (UTOP) scenario, extra fissile materials are inserted into the primary loop and reactor core, and the positive reactivity leads to the power increase. The new capabilities of SAM are expected to simulate MSR transient behaviors under ULOF and UTOP scenarios.

Point-Kinetics Model for MSR

The Point-Kinetics model is shown in Equations 1 and 2 and has been widely used for the transient safety analysis of stationary fuel reactors. It was used to simulate MSR transients in [3-5]. The derivation of the Point Kinetics equations assumes that the fission power shape is not changing during the transient. This assumption may not be valid for MSR transients, especially under the loss of flow scenarios. Coupling between SAM and neutronics codes is also being planned so that the spatial effects can be continuously updated during transients. The Spatial-Kinetics model will also be able to model potential fissions outside the active core and more accurate boundaries of the active core.

Equations 1 and 2 are tightly coupled, and \( n \) and \( C_i \) are solved simultaneously in SAM. The last term in Equation 2 represents the incoming and outgoing delayed neutron precursors and are provided as boundary conditions to the solver. \( c_i \) is the delayed neutron precursor density in the cooling system and is calculated from the mass transport module (described in Equation 3). The effective delayed neutron fraction \( \beta_{\text{eff}} \), where \( \beta_{\text{eff}} = \beta_{\text{static}} - \beta_{\text{loss}}(t) \) in Equation 1, is considered as the loss of delayed neutron fraction varying with the flow rate. \( \beta_{\text{loss}} \) is a time-dependent variable and is continuously updated during the transient calculation based on the DNP loss in the primary loop.

\[
\frac{dn}{dt} = \frac{\rho_{\text{ext}} - \beta_{\text{eff}}}{A} n + \sum_{i=1}^{6} \lambda_i C_i \quad (1)
\]

\[
\frac{dC_i}{dt} = \frac{\beta_i}{A} n - \lambda_i C_i - \int uc_i dA \quad (2)
\]

The Point-Kinetics module above has been implemented in SAM and was verified against forward Euler method and the calculation by SAS4A/SASSYS-1 [6]. The comparisons of the normalized fission power show good agreement for stationary fuel reactors. The coupling between Point-Kinetics solver and DNP drifting model has been completed and is being tested at the time of writing.

Delayed Neutron Precursor Drift in Coolant System

Simulating MSR transients requires the capability of tracking delayed neutron precursors in the primary system.
The DNP drift is governed by Equation 3, which has been implemented in SAM as a general mass transport module. It leverages SAM’s existing system level fluid flow dynamics and heat transfer simulation capability. The source term, $S_i$, represents both user-defined source and fission product source. This mass transport module can be used to model the transport of Tritium, delayed neutron precursors, and decay heat precursors in the reactor systems.

$$\frac{∂\rho c_i}{∂t} + \nabla \cdot (\rho u c_i) - \nabla \cdot (D_i \nabla \rho c_i) = S_i - \lambda_i \rho c_i \quad (3)$$

The DNP drift modeling capability has been tested using a simple MSR primary loop model. The primary loop (Fig. 1) is composed of a 0.8m active core with fission power of 10MW, a heat exchanger with a constant secondary inlet coolant temperature, a pump providing pump head at steady state, and associated pipes. This simplified primary loop is used to demonstrate the delayed neutron precursor drifting in the primary loop and evaluate their behaviors during transient events.

Fig. 1 Schematic of Simplified MSR Primary Loop Model

Fig. 2 compares the distributions of delayed neutron precursors in the primary system under steady state. Due to the different half-lifetimes, DPNs in the six groups have different concentrations in the primary loop: most Group 1 DNP circulate together with the salt while Group 5 and 6 DPNs decrease quickly outside the core.

Based on the loss of delayed neutron precursors outside the core, the $\beta_{eff}$ value in Equation 1 during steady-state conditions is plotted corresponding to the flow velocity at the core outlet (Fig. 3). It should be noted that each point in Fig. 3 is obtained for a new steady state with a specific flow velocity. It is observed that the effective delayed neutron fraction in MSR decreases with the increase of system flow rates.

The new DNP drift capability has also been tested for the transient scenario when the primary flow coasts down to the natural circulation level starting at 20 seconds. For simplification and isolating the effects of reactivity feedbacks, the fission power is fixed at the initial value for this verification test problem. Fig. 4 compares the concentrations of Group 1 and 6 DNP at core outlet, IHX inlet, and core inlet. During the pump coastdown, the concentrations of DNP in the active core increase, and therefore the concentrations at the core outlet increase constantly. Then, the peak value propagates via IHX, and continuously circulates in the primary loop. After about 200 seconds, the concentrations of DNP converge to a new steady state. For Group 6, most DNP decay out before they reach the IHX. Unlike Group 1 DNP, the concentration of Group 6 DNP approaches steady state quickly. This study indicated that DNP in MSR show very complex behaviors during transients and impose significant impact on $\beta_{eff}$ value for the Point-Kinetics model.

<table>
<thead>
<tr>
<th>Group Number</th>
<th>Decay Constant $\lambda_i$</th>
<th>Delay Neutron Fraction $\beta_i$ (pcm)</th>
</tr>
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<tbody>
<tr>
<td>1</td>
<td>0.0124</td>
<td>21.1</td>
</tr>
<tr>
<td>2</td>
<td>0.0305</td>
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<td>3</td>
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<td>4</td>
<td>0.3013</td>
<td>252.8</td>
</tr>
<tr>
<td>5</td>
<td>1.1400</td>
<td>74.0</td>
</tr>
<tr>
<td>6</td>
<td>3.0100</td>
<td>27.0</td>
</tr>
</tbody>
</table>

Fig. 2 Normalized Delayed Neutron Precursor Distribution in the Primary System on Steady State
Novel feature of MSR is the moving decay heat precursors in the primary system. Decay heat is continuously generated in the core and cooling system, which affects the fluid calculation. The decay heat precursor $h_{n,k}$ is solved by the mass transport module (Equation 4), and the source term $S_{JL}$ is obtained from fission reaction rate of the active core. Note that Equation 4 is basically the same as the Equation 3.

$$\frac{\partial \rho h_{n,k}}{\partial t} + \nabla \cdot (\rho u h_{n,k}) - \nabla \cdot (D_i \nabla h_{n,k}) = S_{JL} - \lambda_{n,k} h_{n,k}$$ (4)

Based on the decay heat precursors distributions, the decay heat power at position $r$ in the primary system is obtained by Equation 5 and is coupled with the fluid dynamics calculation.

$$P_d(r^2) = \sum_{k} \sum_{n} \lambda_{n,k} h_{n,k}(r)$$ (5)

Reactivity Feedback

MSR design features the moving fuel together with coolant. During the MSR transient, the temperature change of the salt causes the fuel density change as well as Doppler effect, which contribute reactivity feedback and affects the fission power. Additionally, the graphite temperature (for thermal spectrum designs) and thermal structure displacements may also affect the fission power during the transient and will be investigated in the future.

As an example, the reactivity feedback effects from coolant density changes are calculated using the user-defined coolant density reactivity coefficient $R_{Z\%}$ and the coolant density $\rho_{salt}$ calculated by SAM at time $t$ in the unit of kg/m$^3$. The resulted coolant density reactivity feedback is used in the Point-Kinetics model in Equation 1.

$$R_{salt} = \left( \frac{\Delta k}{\Delta N/N} \right) x \frac{\rho_{salt}^{t,t} - \rho_{salt}^{t,0}}{\rho_{salt}^{t,0}}$$ (6)
CONCLUSION

Compared with stationary fuel, the moving fuel together with the salt introduces extra difficulties for MSR transient safety analyses. This paper presents a number of new capabilities available in SAM for MSR transient simulations. A modified Point-Kinetics model has been developed with boundary conditions of the delayed neutron precursors concentrations at core inlet and outlet. A mass transport module in SAM has been implemented to model the transport of various species in the cooling system. The demonstration case indicates that the delayed neutron precursors shows complex behaviors during the loss of flow transient. Their impacts on the effective delayed neutron fraction should be taken into account in calculating the fission power during the transient. The decay heat generation outside the core and coolant reactivity feedback will also be considered for MSR safety analysis. These new capabilities in SAM are expected to enable MSR simulations under postulated transient scenarios, including loss of flow in the cooling systems and reactivity driven transients.

NOMENCLATURE

\( n \) = prompt fission power  
\( C_i \) = total delayed neutron precursors \( i \)  
\( c_i \) = density of delayed neutron precursors \( i \)  
\( D_i \) = diffusion coefficient for species \( i \)  
\( h_{nk} \) = decay heat precursor density for group \( n \) and decay heat curve \( k \)  
\( \lambda_i \) = decay constant for delayed neutron precursors \( i \)  
\( \rho_{sat} \) = fluid density at time \( t \) in position \( i \)  
\( \rho_{ext} \) = external reactivity feedback  
\( R_{sat} \) = coolant reactivity feedback    
\( \beta_{eff} \) = effective delayed neutron fraction  
\( \lambda \) = prompt neutron lifetime  
\( S_i \) = source term for delayed neutron precursor \( i \)  
\( S^n_k \) = decay heat precursor source term for group \( n \) and decay heat curve \( k \)  
\( u \) = velocity

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