# Modeling and Simulation with AMESIM Code and Comparison with MARS-LMR Code for a PGSFR IHTS

Min Gyu Park [a], Seon Gon Kim [a], Jae Ho Jeong [a*]
*[a]Chung-Ang University, 84, Heukseok-ro, Dongjak-gu, Seoul*
*[*]Corresponding author: jaehojeong@cau.ac.kr*

***Keywords :*** PGSFR, IHTS, Steam Generator, AMESIM, MARS-LMR

## 1. Introduction

Sodium-cooled Fast Reactors (SFRs) have been widely recognized as a promising Generation-IV reactor concept due to their potential for efficient uranium utilization and reduction of high-level radioactive waste. In Korea, the conceptual design of the 392 MWt Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) has been completed, and continuous efforts have been made to evaluate its thermal-hydraulic safety performance.[1]

Conventional safety analyses of PGSFR have primarily relied on the MARS-LMR code, which has been extensively applied to sodium-cooled fast reactor systems. However, recent advances in multiphysics simulation platforms have raised interest in the application of Simcenter AMESIM, which enables integrated system-level modeling across thermal-hydraulic, mechanical, and control domains. Despite its versatility, AMESIM has not been fully validated as a dedicated nuclear safety analysis tool. Therefore, verification against established safety codes such as MARS-LMR is essential to ensure modeling consistency and reliability.

In this study, the Intermediate Heat Transport System (IHTS) and Steam Generator (SG) of PGSFR were modeled using AMESIM. The same conditions were implemented in the MARS-LMR code to establish a coupled comparative framework.[2]

The research process is detailed in Figure 1.

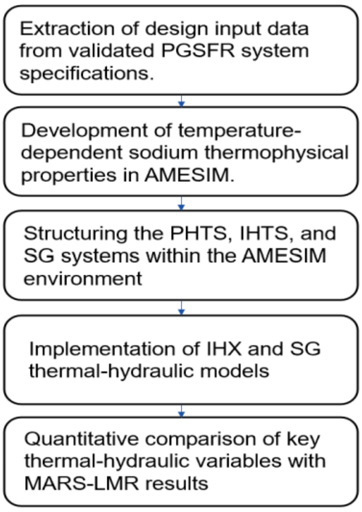


Fig 1. Research Process

## 2. Methods and Results

The present study establishes an AMESIM-based thermal-hydraulic model of the PGSFR Intermediate Heat Transport System (IHTS) and validates it through comparison with the MARS-LMR code under steady-state operating conditions.

### *2.1 Sodium Properties in AMESIM*

The AMESIM code provides various functional libraries for system simulation, among which the Thermal-Hydraulic Library is used to model fluid flow and heat transfer in reactor systems. This library is suitable for representing the thermal-hydraulic behavior of SFR subsystems such as the Intermediate Heat Transport System.

However, high-temperature liquid sodium properties are not included in the default AMESIM database. Therefore, user-defined thermophysical properties must be implemented to perform thermal-hydraulic analysis for Sodium-cooled Fast Reactors.

### *2.2 Governing Equations in AMESIM Thermal-Hydraulic Modeling*

The AMESIM Thermal-Hydraulic Library solves the one-dimensional, transient form of the conservation equations using a control-volume-based formulation.

Each component (pipe, heat exchanger, pump, or volume) is discretized into lumped nodes, and the conservation laws of mass, momentum, and energy are applied to each control volume.

The continuity equation for compressible flow is expressed as:

$$\frac{\partial \rho}{\partial t}+\frac{\partial \rho u}{\partial x}=0 \tag{1}$$

temperature-dependent density modeling is essential for accurately predicting mass flow redistribution and pressure behavior in the IHTS.[3]

The one-dimensional momentum equation is given by:

$$\frac{\partial \rho u}{\partial t}+\frac{\partial (\rho u^2+p)}{\partial x}+\eta_{friction}=0 \tag{2}$$

Frictional losses were calculated using conventional pressure drop correlations, including Reynolds-number-dependent friction factors. This formulation allows prediction of mass flow rates in the primary and intermediate sodium loops under steady-state conditions.[4]

The energy equation is implemented in terms of temperature:

$$\frac{\partial T}{\partial t} = \frac{\dot{m}h_{in} - \dot{m}h_{out} - \dot{m}h_{avg} + \dot{Q}}{m \cdot c_p} + \frac{\mathrm{T} \cdot \alpha}{\rho \cdot c_p} \cdot \frac{dp}{dt} \qquad (3)$$

### *2.3 IHTS Modeling in AMESIM Code*

The IHTS model was developed in the AMESIM environment using component-based hydraulic and thermal elements to represent the heat transfer pathway from the reactor core to the steam generation system.

The primary side of the Intermediate Heat Exchanger (IHX) was modeled to receive high-temperature sodium from the core and transfer thermal energy to the secondary sodium loop through a heat exchanger block.

The secondary loop was implemented as a closed circuit of pipe and volume elements, allowing calculation of mass flow, pressure drop, and temperature distribution along the loop.

An intermediate sodium pump component was introduced to provide the required pressure head and maintain the design flow rate. The loop is thermally connected to the Steam Generator (SG), where a coupled heat exchanger block links the sodium and water domains while ensuring overall energy balance.

All components were connected within AMESIM through hydraulic and thermal ports, enabling integrated solution of flow and heat transfer throughout the Core–IHX–IHTS–SG system.

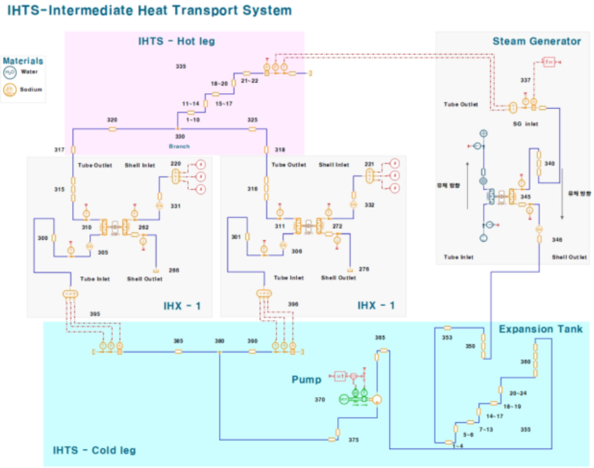
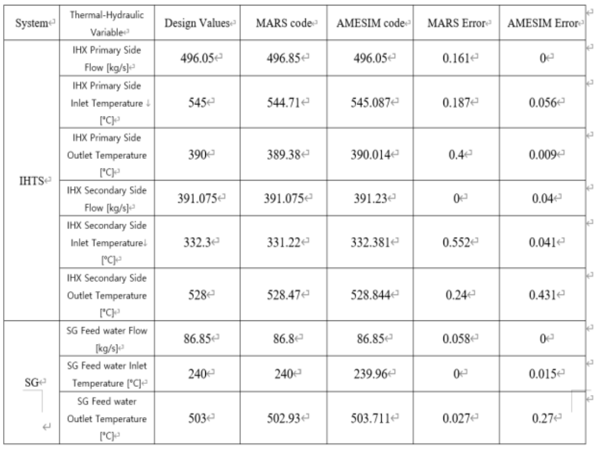


Fig. 2. Modeling of Intermediate Heat Transport System

Table I: Comparison Amesim and MARS-LMR



| System | Thermal-Hydraulic Variable | Design Values | MARS code | AMESIM code | MARS Error | AMESIM Error |
|---|---|---|---|---|---|---|
| IHTS | IHX Primary Side Flow [kg/s] | 496.05 | 496.85 | 496.05 | 0.161 | 0 |
| | IHX Primary Side Inlet Temperature [°C] | 545 | 544.71 | 545.087 | 0.187 | 0.056 |
| | IHX Primary Side Outlet Temperature [°C] | 390 | 389.38 | 390.014 | 0.4 | 0.009 |
| | IHX Secondary Side Flow [kg/s] | 391.075 | 391.075 | 391.23 | 0 | 0.04 |
| | IHX Secondary Side Inlet Temperature [°C] | 332.3 | 331.22 | 332.381 | 0.552 | 0.041 |
| | IHX Secondary Side Outlet Temperature [°C] | 528 | 528.47 | 528.844 | 0.24 | 0.431 |
| SG | SG Feed water Flow [kg/s] | 86.85 | 86.8 | 86.85 | 0.058 | 0 |
| | SG Feed water Inlet Temperature [°C] | 240 | 240 | 239.96 | 0 | 0.015 |
| | SG Feed water Outlet Temperature [°C] | 503 | 502.93 | 503.711 | 0.027 | 0.27 |

### *2.4 Comparison of Steady-State Results*

The steady-state simulation results of the PGSFR IHTS developed using Simcenter AMESIM were compared with the MARS-LMR results to verify the model's reliability. As summarized in Table 1, the temperature discrepancies at the inlet and outlet of the IHX and SG were remarkably low, staying within a 1% to 2% range.

Beyond mere numerical alignment, this agreement underscores the versatility of AMESIM in handling complex sodium-cooled reactor systems despite not having a dedicated nuclear library. Unlike conventional safety codes that are often rigid in their nodalization, AMESIM's component-based approach allows for a highly intuitive and flexible construction of the IHTS loop. This environment facilitates the seamless integration of different physical domains—such as the mechanical characteristics of the pump and the thermal-hydraulic behavior of the sodium—within a single simulation platform. Such a unified modeling capability is a distinct advantage for evaluating the performance of the entire heat transport pathway from the core to the SG..

## 3. Conclusions

In this study, the Intermediate Heat Transport System (IHTS) of the PGSFR was modeled using the AMESIM Thermal-Hydraulic Library with user-defined sodium thermophysical properties. The validation against MARS-LMR confirmed that AMESIM can reliably reproduce the steady-state behavior of the system with high accuracy.

The primary impact of this research lies in demonstrating AMESIM's potential as a multiphysics simulation platform for Gen-IV reactors. The integrated modeling environment provides a significant advantage for future Design Basis Accident (DBA) analyses. Specifically, AMESIM can simultaneously simulate the complex feedback loops between the reactor's control systems and its thermal-hydraulic response, a task that often requires manual coupling in standalone safety codes. Furthermore, the flexibility of the AMESIM environment allows for future expansion into structural or control-specific libraries to evaluate multi-domain interaction effects during transient scenarios. Future work will leverage these multiphysics capabilities to conduct comprehensive safety evaluations, further validating AMESIM's role in the advanced reactor design and optimization process

## REFERENCES

[1] Yoo, J., Chang, J., Lim, J.-Y., Cheon, J.-S., Lee, T.-H., Kim, S. K., Lee, K. L., and Joo, H.-K., 2016, "Overall System Description and Safety Characteristics of Prototype Gen IV Sodium Cooled Fast Reactor in Korea," *Nuclear Engineering and Technology*, Vol. 48, No. 5, pp. 1060–1061.
[2] Jeong, J. H., Lee, J., Park, S. D., and Kim, Y. I., 2020, "Comparative Safety Analysis with MARS-LMR and SAS4A/SASSYS-1 Codes for PGSFR," Nuclear Engineering and Technology.
[3] Siemens Digital Industries Software, *Simcenter Amesim 2022.1 – Thermal-Hydraulic Component Design Library User's Guide*, 2022
[4] Siemens Digital Industries Software, *Simcenter Amesim 2022.1 – Two-Phase Flow Library User's Guide*, 2022.