Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 24-25, 2024

Control Assembly Withdrawal Dynamics: A Comparative Verification of GAMMA+ with MARS-LMR

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1. Introduction

The GAMMA+ code, originally developed at Korea's KAERI to simulate air ingress in high-temperature gas-cooled reactors (HTGRs), has been substantially enhanced for broader applications in nuclear system analyses. Now supporting advanced capabilities for system transients and safety assessments of very high-temperature reactors (VHTRs), the latest version, GAMMA+ 2.1, enables detailed simulations of thermo-fluid phenomena and dynamic system components. This versatile tool has supported numerous national and international projects, including the design of various reactor models and collaborative research with U.S. institutions. Recent upgrades have extended GAMMA+ functionalities to sodium-cooled fast reactors (SFRs), emphasizing its ongoing improvements and essential role in the nuclear research community through rigorous verification and validation efforts.

The AOO TOP accident analysis was performed on the SALUS reactor, which is under development at KAERI. The SALUS (Small Advanced Long-cycled and Ultimate Safe SFR), being developed by KAERI, is characterized by its ability to operate for approximately 20 years without the need for nuclear fuel replacement.

The objective of these studies is to verify the GAMMA+ code by comparing its analysis results with those from MARS-LMR.

2. Methods and Results

2.1 Accident Analysis Methodology

The Reactivity Anomalies category includes the control rod withdrawal event (TOP). Acceptance criteria for preventing fuel failure are based on the fuel's melting temperature and the Cumulative Damage Fraction (CDF) of the cladding, with the safety acceptance criterion for AOOs set to maintain all CDFs below 0.05.
Initial conditions are conservatively determined by selecting parameters either above or below design conditions to evaluate severe scenarios. Conservative assumptions for SALUS safety analysis include a +2% allowance for core power calorimetric error, a +12°C allowance for temperature, and a ±8% flow variance.
The SALUS design employs dual control rod assemblies for emergency shutdown, supported by the conservative application of the ANS-79 model with a 20% uncertainty for decay heat generation. Setpoints and response times for the Plant Protection System (PPS) are specified in Table I. Assumptions include single failures in safety components and operational reliability of non-nuclear systems. A loss of offsite power is presumed immediate upon a reactor trip, with a 30-minute delay credited for operator safety actions post-event notification in design basis events. Initial conditions and key input parameters for reactivity feedback are detailed in Table II.

Table I: Problem Description

Parameters	Analysis Setpoints	Delay (sec)	Function
Overpower	118%	0.5	Reactor Trip
Variable Overpower	10%/min	0.5	Reactor Trip
High Power to PHTS Flow Ratio	120%	0.8	Reactor Trip
High Core Inlet Temperature	384°C	6.0	Reactor Trip ESF Actuation
High Center Fuel Assembly Outlet Temperature	674°C	6.0	Reactor Trip ESF Actuation

Table II: Initial conditions and Assumptions

Parameter	Assumed Value (% relative to nominal values)
Core Power	102 %
Core Inlet/Outlet Coolant Temperature	103.3 % (inlet) 105.4 % (outlet)
Core Coolant Flow Rates	92.0 %
Doppler Reactivity	Most Negative (-)
Sodium Density Reactivity	Least Positive (+)
Fuel Axial Expansion Reactivity	Most Negative (-)
Core Radial Expansion Reactivity	Most Negative (-)
CRDL/RV Expansion Reactivity	Most Negative (-)
Single Failure	Single Failure of PDHRS

2.2 Analysis Results

The sequence for this accident, as shown in Table III, indicates that in GAMMA+, a reactor trip occurs due to a High Central Subassembly Outlet Temperature signal, while in MARS-LMR, it is triggered by a High Power to PHTS Flow Ratio. As depicted in Figure 1, both factors are positioned close to the set point post-accident, and the variance in reactivity feedback alters the trip signals. The results from each code reveal that trip signals shift within a few seconds of each other. In GAMMA+, the

HCSOT signal leads to simultaneous reactor trip and DRHRS actuation. Due to heat removal by the DHRS, the temperatures at the core outlet and inlet post-trip are lower compared to those in MARS-LMR.

Table III: Initial conditions and Assumptions

Time	GAMMA+	Time	MARS-LMR
0.0	- Control rod withdrawal starts	0.0	- Control rod withdrawal starts
127.1	- PPS High HCSOT Signal	139.7	- PPS High P/Q Signal
133.1	- Rx Trip by PPS - DRHRS Actuation by PPS	140.5	- Rx Trip by PPS
133.3	- PPS High P/Q Signal	141.0	- Control Rods Insert
133.6	- Control Rods Insert	142.3	- PPS High HCSOT Signal - DRHRS Actuation by PPS

As the trip signal initiates, the flow rates in the PHTS, IHTS, and Feed water systems decrease due to the shutdown of pumps. However, some flow is sustained due to natural convection within the reactor pool, driven by the residual heat of the core. Initially, the natural convection flow rate in GAMMA+ PHTS is relatively high, but as the temperature difference between the core inlet and outlet decreases, it diminishes compared to the MARS-LMR results. The trends in core power output and DHRS heat removal are similar between the two codes. Variations in core inlet and outlet temperatures occur depending on the timing of the DRHRS actuation signal. The CDF is calculated to be 0.00149, similar to that of MARS-LMR (=0.0014)..

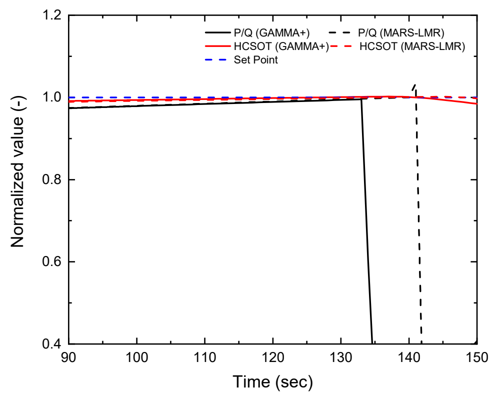

Fig. 1. Temporal Variation of Trip Signals in GAMMA+

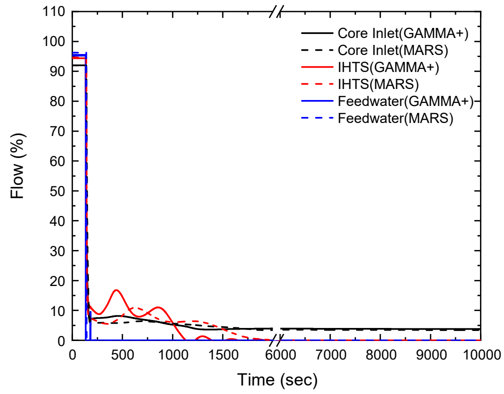

Fig. 2. Comparison of Flow Changes in TOP

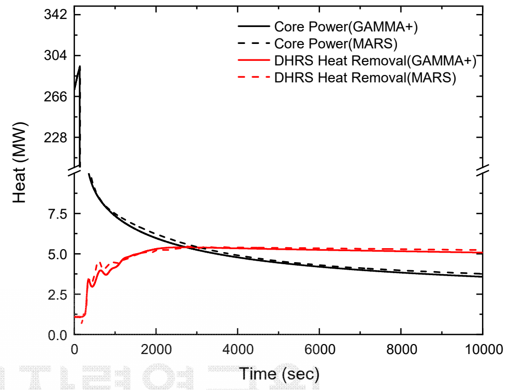

Fig. 3. Comparison of Core Power and DHRS Heat Removal Changes in TOP

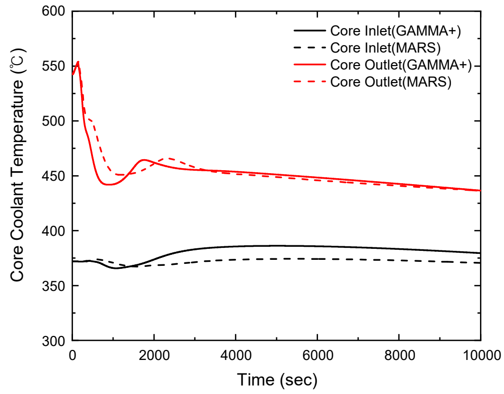

Fig. 4. Comparison of Core Inlet/Outlet Temperature Changes in TOP

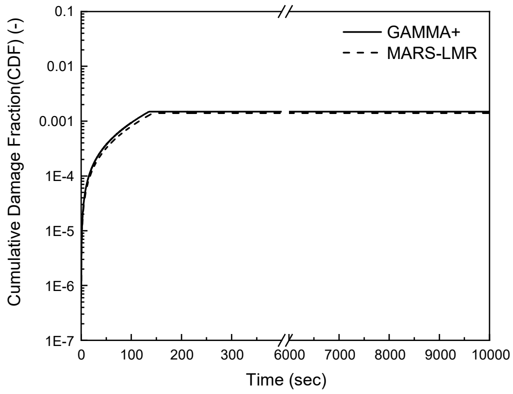

Fig. 5. Comparison of CDF Changes in TOP

3. Conclusions

The GAMMA+ analysis results for the AOO TOP accident indicate that a reactor trip and DRHRS actuation occur due to the HCSOT signal. The PPS signal and heat removal by the DHRS ensure that the reactor safely shuts down. The moment when the heat removal by the DHRS surpasses the residual heat in the core occurs at 2730 seconds, and the CDF is 0.00149, confirming that the reactor design safety criterion of CDF < 0.05 is comfortably met.

ACKNOWLEDGMENTS

This work was supported by a National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (No. 2021M2E2A2081061, 2021M2E2A1037871].

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