Safety Analysis of All PHTS Pumps Failure Event in the SALUS

In Sub, Jun*, Huee-Youl Ye
Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, 305-353, Korea
**Corresponding author: isjun@kaeri.re.kr*

1. Introduction

The SALUS (Small Advanced Long-cycled and Ultimate Safe SFR) is designed by KAERI to operate for about 20 years without refueling. In this paper, a safety analysis of all primary heat transfer system (PHTS) pumps failure in the SALUS was performed using the MARS-LMR code [1].

2. Methods

2.1 Identification of Causes

All PHTS pumps failure event may result from a simultaneous loss of power supplies for PHTS pumps or malfunctions of PHTS pump motors. This event is classified into an Anticipated Operational Occurrence (AOO). The maximum cumulative damage fraction (CDF) shall be less than the design limit of 0.05 to avoid fuel cladding damage [2].

2.2 Sequence of Event and System Operation

Once all PHTS pumps stop, the PHTS pumps begin to coast down according to the inertia of pump flywheels. This results in the rapid decrease in the PHTS flow. The reactor trip signal by the high 'power-to-flow-ratio (P/Q)' from the PPS is generated and the control rod assemblies begin to drop into the core.

Since a loss of off-site power (LOOP) concurrent with a turbine trip following a reactor trip is considered as a basic assumption, all intermediate heat transfer system (IHTS) pumps and feedwater pumps stop at the same time. The core coolant temperature increases due to a loss of heat removal through the steam generators.

The engineered safety feature (ESF) actuation signal is generated by the high 'core inlet temperature (CIT)' from the PPS, then the intact decay heat removal system (DHRS) dampers are fully opened and the blowers are operated.

Finally, the core decay heat is removed through the intact DHRS.

2.3 Input Parameters and Initial Conditions

Fig. 1 shows the nodalization of the MARS-LMR code for the safety analysis in the SALUS.

The conservative input parameters and initial conditions are considered for a safety analysis [3, 4]. An initial core power of 102% of full power is assumed to consider measurement uncertainty and conservative analysis.

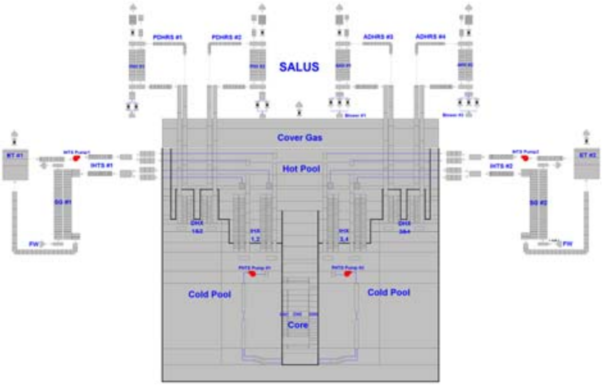

Fig. 1. MARS-LMR Nodalization for SALUS

Table I: Initial Condition Combinations

No.	Power	PHTS Flow	T_{in}	T_{out}
1	102%	92%	348℃	519℃
2	102%	92%	372℃	493℃
3	102%	108%	348℃	542℃
4	102%	108%	372℃	517℃

In addition, the most conservative initial condition among the combinations of the core inlet temperature and PHTS flow rate described in Table I is considered.

The additional assumptions used in this analysis are as follows: 1) It is assumed that a LOOP occurs immediately upon a reactor trip. The IHTS pumps and the feedwater pumps therefore assumed to stop at the time of a reactor trip. 2) The minimum available scram rod worth is considered to maximize the heat flux after a reactor trip. 3) One control rod assembly is assumed to be stuck at the out-position (N-1). 4) The delay time of the rod drop is assumed to be 0.5 seconds. 5) There is no single failure for this event that affects the acceptance criteria. However, the failure of one PDHRS is assumed to minimize the decay heat removal through the DHRS after a reactor trip.

3. Results

3.1 Sensitivity Study Results

Based on the parametric studies, these initial conditions (i.e., core power, core inlet coolant temperature, and PHTS flow) are chosen in order to maximize the CDF. In addition, the most conservative combination of the reactivity feedbacks from the doppler, the coolant density change, and the axial/radial expansion effects in the core and structures is selected to maximize the CDF.

The CDF is maximized in the conditions of the least coolant density reactivity feedback and the most other reactivity feedbacks as shown in Fig. 2.

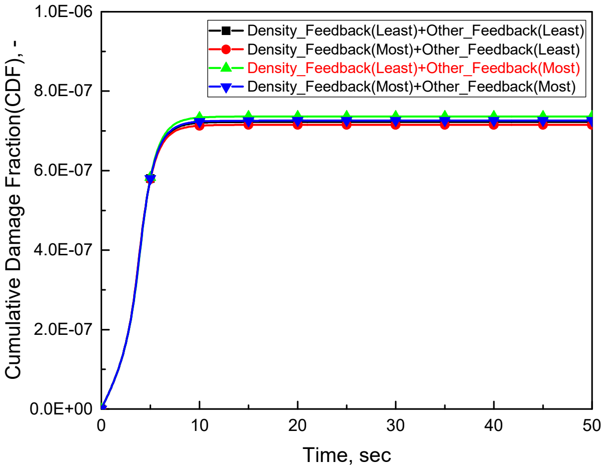

Fig. 2. Sensitivity Study Result for Feedback Reactivities

The CDF is maximized in the conditions of high core inlet temperature and low PHTS flow rate as shown in Fig. 3.

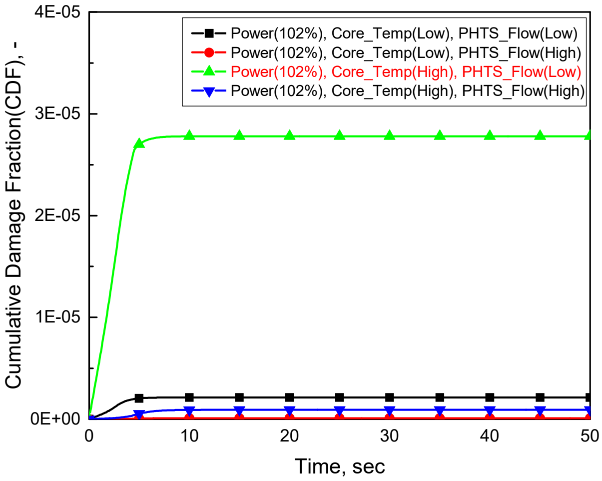

Fig. 3. Sensitivity Study Result for initial Conditions

3.2 Analysis Results

Table II: Sequence of Event

Time (sec)	Events	Values
0.0	One PHTS pump stop	-
1.73	Rx trip by high ‘P/Q’ - Loss of off-site power - All PHTS pumps stop - All feedwater pumps stop	120%
2.23	Control rod assemblies drop	-
2494	ESF Actuation by high ‘CIT’ - DHRS dampers open	384℃
∞	Core cooling by DHRS	-

The sequence of event due to the failure of all PHTS pumps is summarized in Table II.

Once all PHTS pumps stop, the core flow decreases very rapidly as shown in Fig. 4. It results in the increase of the core coolant temperature and the fuel temperature. The ‘P/Q’ reaches the reactor trip setpoint of the PPS. A reactor trip signal by the high ‘P/Q’ from the PPS is generated at 1.73 seconds. Also, a loss of offsite power occurs at this time.

Subsequently, the control rod assemblies begin to drop into the core. The core power decreases slightly prior to the reactor trip due to the reactivity feedback effects. After the insertion of the control rod assemblies, the core power rapidly decreases as shown in Fig. 5.

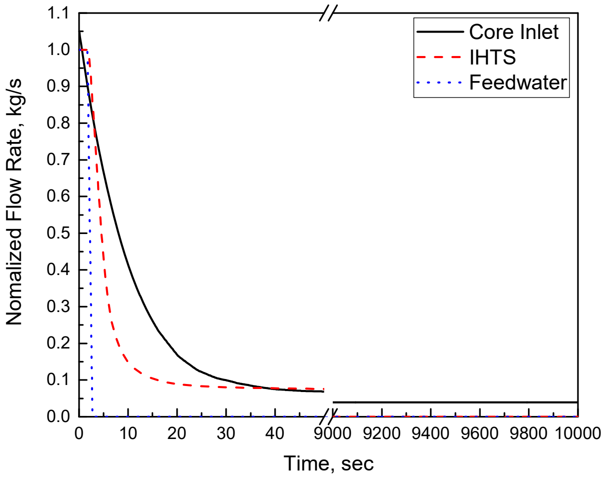

Fig. 4. Core Inlet, IHTS, and Feedwater Flow Rates

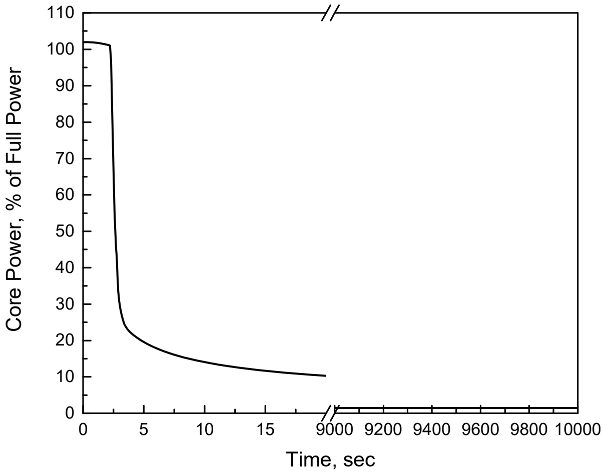

Fig. 5. Core Power

The heat removal through the steam generators is lost after a reactor trip as the IHTS pumps and the feedwater pumps stop due to a LOOP. This leads to the core coolant temperature increases as shown in Fig. 6.

At about 2,494 seconds, the ESF actuation signal by the high ‘CIT’ from the PPS is generated. The intact DHRS dampers are fully opened and the blowers begin to operate.

At around 3,200 seconds, the removed heat through the intact DHRS exceeds the core decay heat as shown in Fig. 7 and thereafter the core coolant temperature continuously decreases.

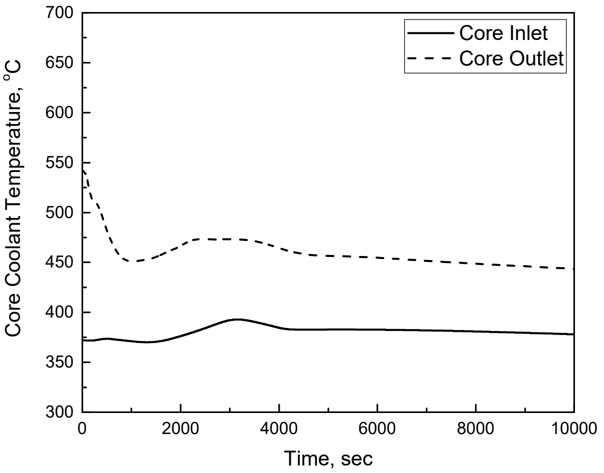

Fig. 6. Core Coolant Temperatures

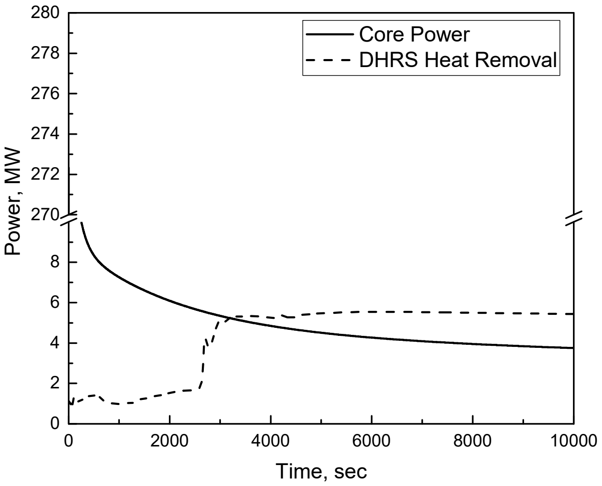

Fig. 7. Core Power and Decay Heat Removal

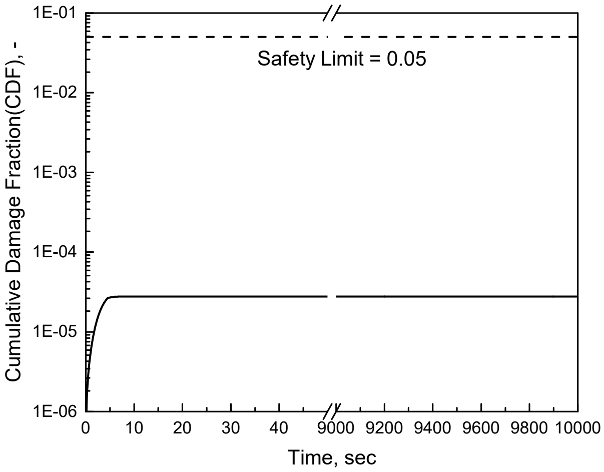

Fig. 8. Cumulative Damage Function (CDF)

The CDF behavior at the hot fuel assembly is shown in Fig. 8. The CDF rises drastically due to the increase of the coolant and fuel temperatures. After a reactor trip, the CDF remains below the safety limit.

4. Conclusions

A safety analysis of all PHTS pumps failure was performed using the MARS-LMR code. The maximum CDF remains less than the design limit CDF of 0.05 during the transient. Hence, no fuel failures are predicted.

ACKNOWLEDGMENTS

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

REFERENCES

[1] H. Y. Jeong et al., "Thermal-hydraulic model in MARS-LMR", KAERI/TR-4297/2011, KAERI, 2011.

[2] Ji-Woong Han et al, "Parametric study on the cumulative damage fraction for SALUS," Transaction of the Korean Nuclear Society Autumn Meeting, 2023.

[3] Sun Rock Choi, "SALUS Design Data for Safety Analysis," SAL-900-E2-301-002, Rev. 01, 2023.

[4] In Sub Jun, "Initial Conditions for Safety Analysis," SAL-900-E2-301-003, Rev. 0, 2024.