Comparative Safety Analysis with MARS-LMR and SAS4A/SASSYS-1 Codes for PGSFR

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1. Introduction

The PGSFR (Prototype Gen-Ⅳ Sodium-cooled Fast Reactor) which has thermal power of 392.2MW has been developed in Korea Atomic Energy Research Institute (KAERI) under a National Nuclear R&D program since 2012 to reduce a high-level waste and use a uranium resource efficiently [1]. KAERI has developed a specific design of the PGSFR from 2016 to 2017, which is the pool type SFR (Sodium-cooled Fast Reactor) with metallic fuel of U–10%Zr for a core having inherent reactivity feedback mechanisms and high thermal conductivity. The PGSFR consists of the PHTS (Primary Heat Transport System), the IHTS (Intermediate Heat Transport System), and the DHRS (Decay Heat Removal System) [2] as shown in Fig 1. The PGSFR has inherent safety features accord with the goal of generation-Ⅳ nuclear power plant. PGSFR has inherent negative reactivity during the plant operation time. Also, it has passive safety system to prevent the loss of power in operation time by utilizing a natural circulation in DHRS.

The MARS-LMR code [3] has been used as a safety analysis tool for the PGSFR safety analysis. KAERI has a plan to obtain licensing approval of MARS-LMR code from Korea Institute of Nuclear Safety (KINS). KAERI prepares to submit the topical report of MARS-LMR code to the KINS until the first half of 2019. KAERI already had preliminary presentation in the KINS to explain the overall content of MARS-LMR code TR in 2018. The KINS provided a comment that KAERI would prepare the code-to-code comparative analysis. In this case, comparison between MARS-LMR code and SAS4A/SASSYS-1 code will be the best because SAS4A/SASSYS-1 code is well validated and used for other reactors application such as the Fast Flux Test Facility program in 1970, the Clinch River Breeder Reactor in 1980, Advanced Liquid Metal Reactor program in 1990. SAS4A/SASSYS-1 is a software simulation tool used to perform deterministic analysis of anticipated events as well as design basis and beyond design basis accidents for sodium cooled fast reactors [4].

With the background presented above, the objective of this project was to implement the comparative safety analysis using MARS-LMR and SAS4A/SASSYS-1 codes for three representative design basis events such as Transient Over Power (TOP, Seismic Reactivity Insertions SSE, DBA Class II), Loss Of Flow (LOF, Spurious PHTS Pump Trip, AOO), and Loss Of Heat Sink (LOHS, Single Feedwater Pump Seizure, DBA Class II) for a specific design of PGSFR.

2. Safety Analysis Methodology

Fig. 1 shows the safety analysis nodalization of MARS-LMR for the specific design of the PGSFR. Fig. 2 shows the safety analysis nodalization of SAS4A/SASSYS-1 for the specific design of the PGSFR. The core is modeled by parallel flow channels such as nine hottest subassemblies, the rest of driver fuel assemblies, control rod assemblies, IVS (In Vessel Storage) assemblies, reflector assemblies, shield assemblies, and leakage flow. The PHTS is placed in a large pool, which is divided into hot pool and cold pool zones. The four sodium-to-sodium DHXs (Decay Heat eXchangers) and two pumps are located in the cold pool, whereas four IHXs (Intermediate Heat eXchagers) are located in the hot pool to transfer the reactor generated heat from the PHTS to the SG (Steam Generators). The IHTS consists of the two IHXs tube side, piping, one EM pump, and one SG shell side. The SG inlet feed-water boundary region is adopted with a constant mass flow-rate condition. In addition, the SG outlet boundary region nearby high-pressure turbine is adopted with a constant pressure condition. Each DHRS is modeled by PDHRS and ADHRS, respectively. DHX is located and submerged in the cold pool region and the sodium-to-air heat exchanger is located in the upper region of the reactor building. The air boundary regions are adopted with a pressure condition for simulating natural circulation phenomena. Table I shows the plant protection system parameters and their setpoints.

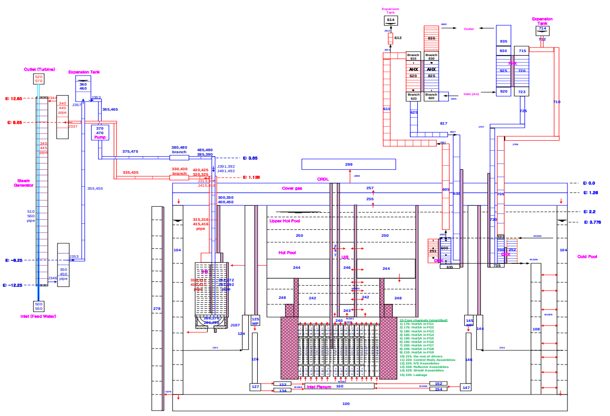

Fig. 1. Nodalization of MARS-LMR code for specific design of PGSFR.

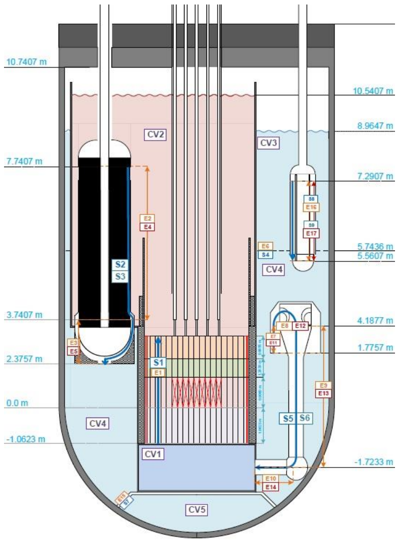

Fig. 2. Nodalization of SAS4A/SASSYS-1 code for specific design of PGSFR

3. Safety Analysis Results

Table II describes the steady state comparison of the analysis values of MARS-LMR and SAS4A/SASSYS-1 codes on each parameter. Based on the steady state of Table II, a comparative safety analysis has been carried out using MARS-LMR and SAS4A/SASSYS-1 codes for three representative design basis events such as TOP, LOF, and LOHS.

3.1 Seismic Reactivity Insertions SSE [Reactivity Anomalies, DBA Class II]

The event is initiated by the insertion of positive reactivity as a result of the radial core structure compaction due to fuel assemblies deformed by an earthquake, leading to increases in the core power and the core outlet temperature. Reactivity insertion of 0.579 $ for 0.1 seconds is adopted at BOC condition. Fig. 3 shows the safety analysis results of MARS-LMR and SAS4A/SASSYS-1 codes. At 0.0 seconds, the core power increases since positive reactivity is inserted due to a seismic reactivity insertions of Safe-Shutdown Earthquake (SSE). Furthermore, SSE induced Loss Of Offsite Power (LOOP) is assumed at 0.0 sec, and then the PHTS, IHTS, and feedwater pump trips follow at the same time. As the core power increases, the 'high neutron flux change rate trip' signal reaches the trip setpoint, and the insertion of control assemblies starts. As shown in Fig. 3, the safety analysis results of MARS-LMR code corresponds well with the results of SAS4A/SASSYS-1 code.

3.2 Spurious PHTS Pump [Loss of Flow, AOO]

A spurious PHTS pump trip with coastdown results from the mechanical failure of both PHTS pumps or the simultaneous loss of electrical power to PHTS pumps. Fig. 4 shows the safety analysis results of MARS-LMR and SAS4A/SASSYS-1 codes. At 0.0 seconds, the core outlet temperature increases due to the decrease of the core flow rates since a loss of offsite power occurs. As the core flow rates rapidly decrease, the 'high power to PHTS flow ratio trip' RPS signal reaches the trip setpoint, and the reactor trip signal is generated, and the insertion of control assemblies starts. At the same time as the reactor trip signal is generated, LOOP is assumed. As shown in Fig. 4, the safety analysis results of MARS-LMR code agrees rather well with the results of SAS4A/SASSYS-1 code.

3.3 Single Feedwater Pump Seizure [Loss of Heat Sink, DBA Class II]

The event is initiated by the seizure of a bearing. Since the rotating shaft of the single feedwater pump stops instantaneously, the feedwater flow rate is rapidly decreased. It is assumed that the affected feedwater flow rate is set to 0.0 kg/s. Fig. 5 shows the safety analysis results of MARS-LMR and SAS4A/SASSYS-1 codes. A high core inlet temperature trip signal is generated, and the reactor is tripped. At the same time as the reactor trip signal is generated, LOOP is assumed. As shown in Fig. 5, the safety analysis results of MARS-LMR code corresponds well with the results of SAS4A/SASSYS-1 code.

4. Conclusions

A comparative safety analysis using MARS-LMR and SAS4A/SASSYS-1 codes has been carried out for three representative design basis events. The safety analysis results of MARS-LMR code are in good agreement with the results of SAS4A/SASSYS-1 code.

ACKNOWLEDGEMENTS

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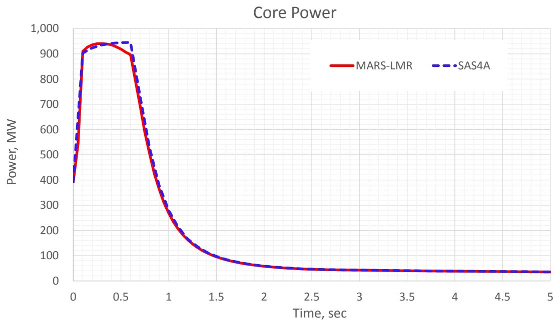
Table I: Plant Protection System Parameters and Setpoints

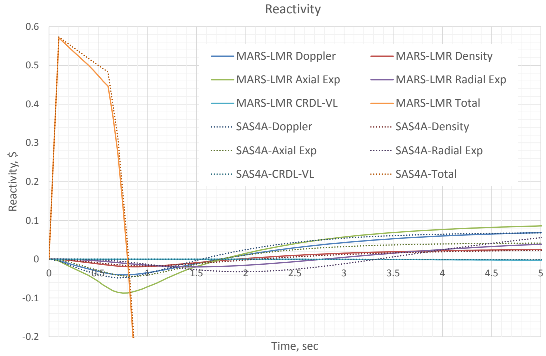
Trip Parameters	Setpoint	Action
Overpower	110 %	Reactor Trip
Variable Overpower	7 %/min	Reactor Trip
High Power to PHTS Flow Rates Ratio	110 %	Reactor Trip
High Core Inlet Temperature	Nominal+15˚C	Reactor Trip
High Center Fuel Assembly Outlet Temperature	Nominal+15˚C	Reactor Trip
High Individual Fuel Assembly Outlet Temperature	Nominal+15˚C	Reactor Trip

Table II: Steady state comparison of the analysis values with MARS-LMR and SAS4A/SASSYS-1 codes

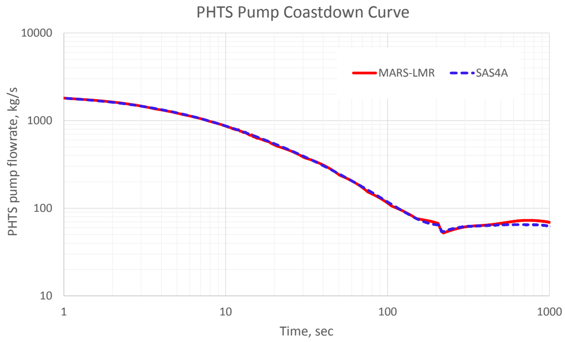
Parameters	MARS-LMR	SAS4A/SASSYS-1
Core condition	BOC	BOC
Core power	392.2	392.20
Core flowrate	1988.1	1992.95
Core inlet temperature	390.12	390.00
Core outlet temperature	545.32	545.03
IHX shell-side flowrate	495.65	498.24
IHX shell-side inlet temperature	544.97	545.03
IHX shell-side outlet temperature	389.52	390.47
IHX tube-side flowrate	391.23	391.08
IHX tube-side inlet temperature	331.3	332.64
IHX tube-side outlet temperature	528.71	529.08
SG steam outlet pressure	16.7	N/A
SG tube-side flowrate	86.84	N/A
SG tube -side inlet temperature	239.97	N/A
SG tube -side outlet temperature	509.02	N/A
IHX heat removal rate	97.685	97.805
SG heat removal rate	195.45	195.609
DHX heat removal rate	0.36/0.37	0.298/0.297
AHX heat removal rate	0.39	0.298
FHX heat removal rate	0.38	0.297
Peak coolant temperature (FA outlet region)	568.91	565.9



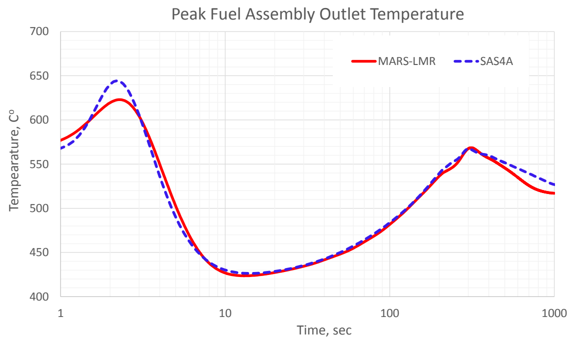
(a) Core power



(b) Reactivity

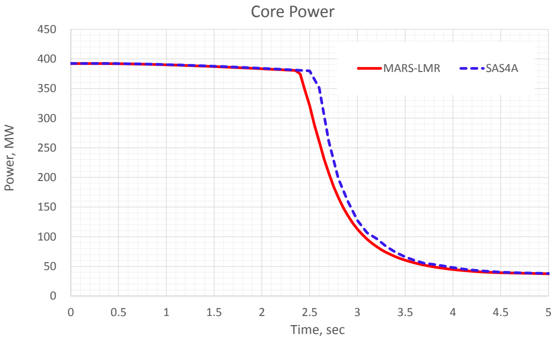


(c) Core coolant flowrate

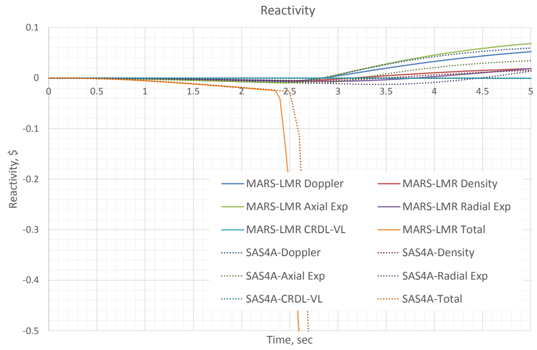


(d) Peak fuel assembly outlet temperature

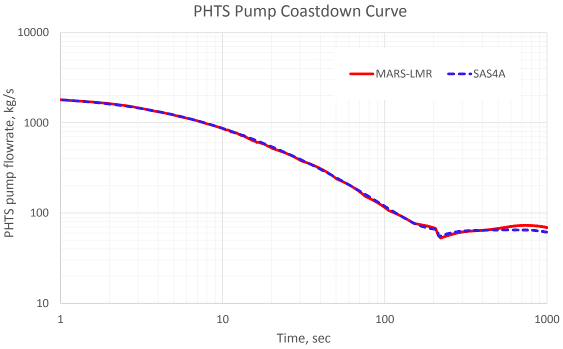
Fig. 3. Safety analysis results of seismic reactivity insertions SSE



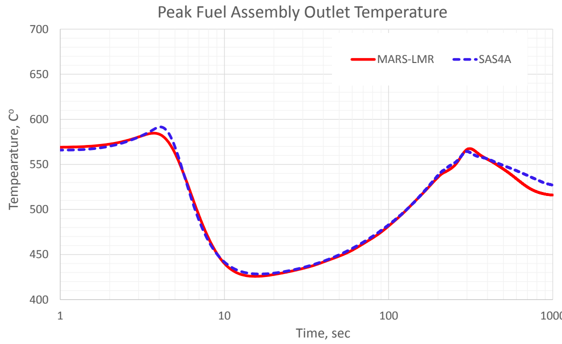
(a) Core power



(b) Reactivity

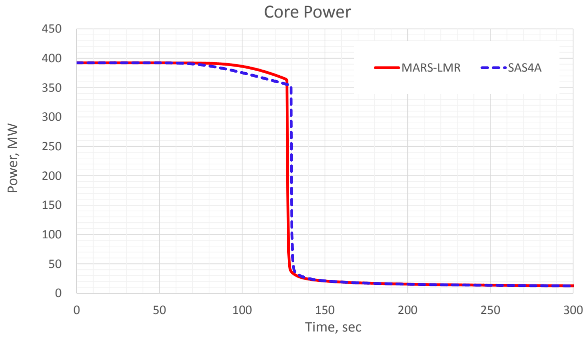


(c) Core coolant flowrate

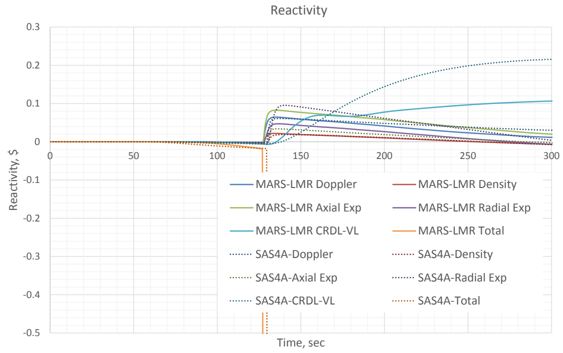


(d) Peak fuel assembly outlet temperature

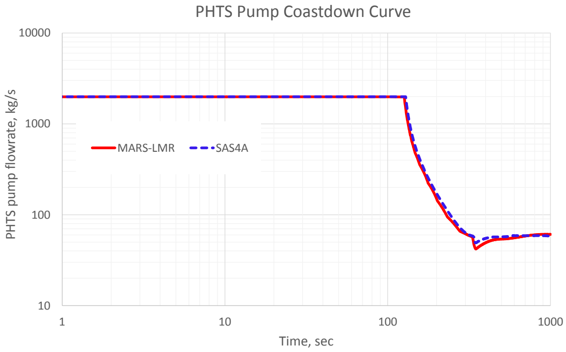
Fig. 4. Safety analysis results of spurious PHTS pump



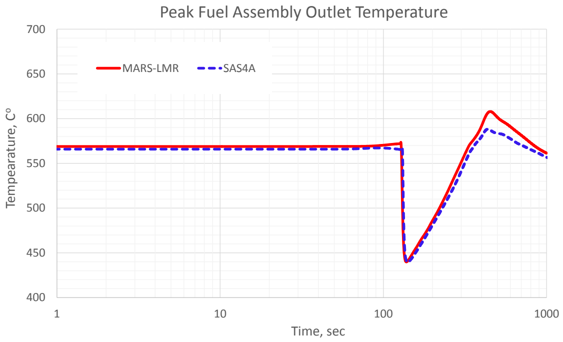
(a) Core power



(b) Reactivity



(c) Core coolant flowrate



(d) Peak fuel assembly outlet temperature

Fig. 5. Safety analysis results of single feedwater pump seizure