Safety Analysis of Design Basis Events 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 2018, 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.

In this study, safety analysis of DBEs (Design Basis Events) such as AOOs (Anticipated Operational Occurrences) and DBAs (Design Basis Accidents) for PGSFR specific design is implemented using MARS-LMR code [3].

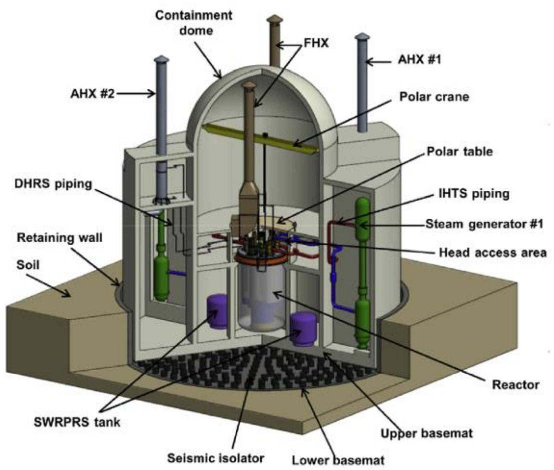


Fig. 1. Overall configuration of the PGSFR.

2. Safety Analysis Methodology

DBEs (Design Basis Events) such as AOOs and DBAs are analyzed with a conservative deterministic evaluation method (a best-estimate code and conservative input and BCs) considering sensitivity analysis of LCOs (Limiting Conditions for Operation) and design uncertainty.

The LCOs sensitivity analysis is performed to determine the most conservative initial condition among initial, transition and equilibrium cycle core. The safety acceptance criteria related variables such as CDF (Cumulative Damage Fraction), fuel center temperature, cladding temperature, and coolant temperature are maximized after conservative consideration of the effects of the uncertainty in the reactor kinetic parameters.

For considering conservative approach, LOOP (Loss Of Offsite Power) is assumed to occur with the reactor trip, and then the PHTS, the IHTS, and the feedwater pump trips follow at the same time. A failure of one train of PDHRS (Passive DHRS) and ADHRS (Active DHRS) due to maintenance and single failure is assumed, respectively. It is also assumed that all control assemblies except the maximum worth one are inserted to shutdown.

DBEs are conventionally classified based on their occurrence frequency and safety acceptance criteria as shown in Table I. Table II shows representative event classification based on the event frequency data of other SFRs.

Table I: Occurrence Frequency and Safety Acceptance Criteria

Frequency/r·y	Plant Condition	Fuel, Cladding, Structure, Containment Damage Limit
F≥ $1x10^{-2}$	AOO	- No fuel melting, Maintain clad integrity, Core coolability - $CDF_{\sum AOO}$ < 0.05 - ASME Service Level B - Maintain design leakage rate
$1x10^{-2}$ >F≥ $1x10^{-4}$	DBA Class I	- No fuel melting, Small fraction of fuel pin failure, Core coolability - CDF_{each} < 0.05 - ASME Service Level C - Maintain design leakage rate
$1x10^{-4}$ >F≥ $1x10^{-6}$	DBA Class II	- Pin coolable geometry - Fuel T < 1,237 ˚C, Clad T < 1,075 ˚C, No bulk sodium boiling - ASME Service Level D - Maintain design leakage rate

Table II: Representative Event Classification

	Initiating Events	Event Frequency
Reactivity Anomalies	Control Rod Assembly Withdrawal with Full Speed	DBA Class I
	Seismic Reactivity Insertions-SSE	DBA Class II
Loss of Flow	Spurious PHTS Pump Trip	AOO
	PHTS Pump Pipe Break	DBA Class II
	Single PHTS Pump Seizure	DBA Class II
Loss of Heat Sink	Single Feedwater Pump Seizure	DBA Class II
	Steam Generator Tube Large Leaks	DBA Class II
	Station Black Out	DBA Class II

Fig. 2 shows the safety analysis nodalization for the specific design of the PGSFR. The core is modeled by fifteen 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 III shows the plant protection system parameters and their setpoints.

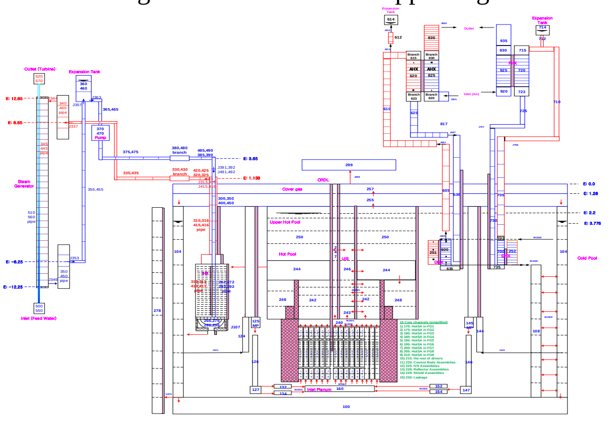


Fig. 2. Safety analysis nodalization for specific design of PGSFR.

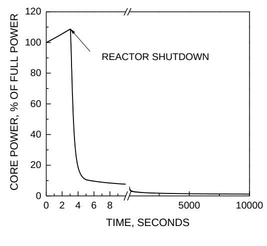
Table III: Plant Protection System Parameters and Setpoints

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

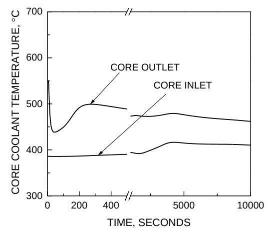
3. Safety Analysis Results

3.1 Control Rod Assembly Withdrawal with Full Speed [Reactivity Anomalies, DBA Class I]

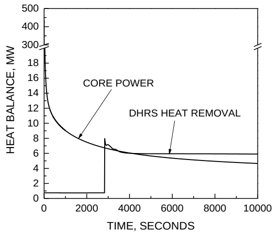
The event is initiated by the insertion of positive reactivity as a result of the withdrawal of the control rod assembly at full speed, which is caused by failures in the control rod assembly driving device and the reactor control system, or an operator mistake. A reactivity insertion of 0.489 $ for 26.6 seconds is adopted to conservatively consider the uncertainties of reactivity insertion amount and time. Fig. 3 shows the safety analysis results. At 0.0 seconds, the core power increases, since a positive reactivity is inserted due to the withdrawal of the control rod assembly at full speed. As the core power increases, the 'high neutron flux



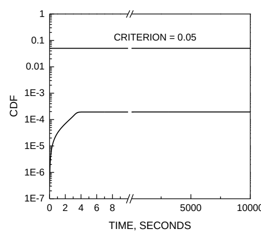
(a) Core power



(b) Core average temperature



(c) Decay heat removal



(d) CDF

Fig. 3. Safety analysis results of control rod assembly withdrawal with full speed.

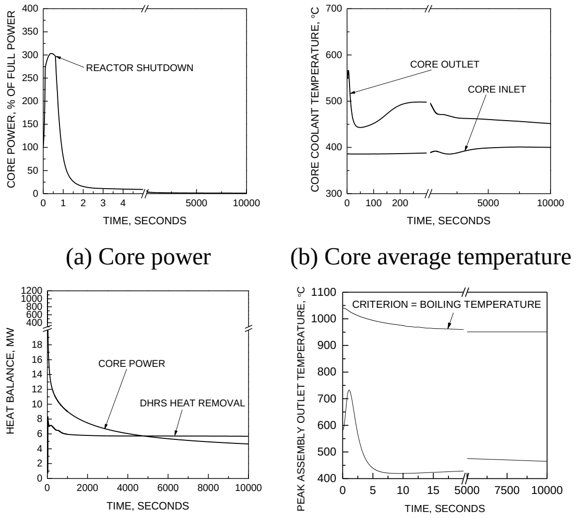
change trip' signal reaches the trip setpoint at 2.44 seconds, and the reactor trip signal is generated at 2.49 seconds, and the insertion of control assemblies starts at 3.04 seconds. At the same time the reactor trip signal is generated, the PHTS pumps stop, and the reactor coolant flow decreases with the coastdown operation of the PHTS pumps. As the core coolant flow decreases, in about 20 seconds the core power to flow ratio increases, and the core outlet coolant temperature rises. Heat removal through the steam generator is lost as the IHTS pumps and feedwater pumps stop. This leads to an increase in the core inlet coolant temperature at about 2,000 seconds. At about 2,800 seconds, the 'high core inlet temperature' ESF actuation signal reaches the setpoint. Then, the DHRS dampers are fully opened and the blower begins to operate. At around 4,200 seconds, the DHRS heat removal rate exceeds the decay heat generation rate, and core outlet temperature continuously decreases. The CDF is less than 1.94E-04.

3.2 Seismic Reactivity Insertions-SSE [Reactivity Anomalies, DBA Class II]

The event is initiated by the insertion of positive reactivity as a result of the core structure compaction due to an earthquake, leading to increases in the core power and the core outlet temperature. Reactivity insertion of 0.637 $ for 0.1 seconds is adopted to maximize the peak coolant, cladding, and fuel temperatures considering the uncertainties of reactivity insertion amount and time. Fig. 4 shows the safety analysis results. At 0.0 seconds, the core power increases since positive reactivity is inserted due to a seismic reactivity insertions-SSE. As the core power increases, the 'high neutron flux change rate trip' signal reaches the trip setpoint at 0.01 seconds, and the insertion of control assemblies starts at 0.61 seconds. At the same time as the reactor trip signal is generated, then the PHTS pumps stop, and the reactor coolant flow decreases with the coastdown operation of the PHTS pumps. Heat removal through the steam generator is lost as the IHTS pumps and feedwater pumps stop. At about 28 seconds, the 'high central fuel assembly outlet temperature' ESF actuation signal reaches the setpoint. Then, the DHRS dampers are fully opened and the blower begins to operate. At around 4,800 seconds, the DHRS heat removal rate exceeds the decay heat rate, and thereafter the core outlet temperature continuously. The peak assembly outlet temperature is 732.5 ℃.

3.3 Spurious PHTS Pump [Loss of Flow, AOO]

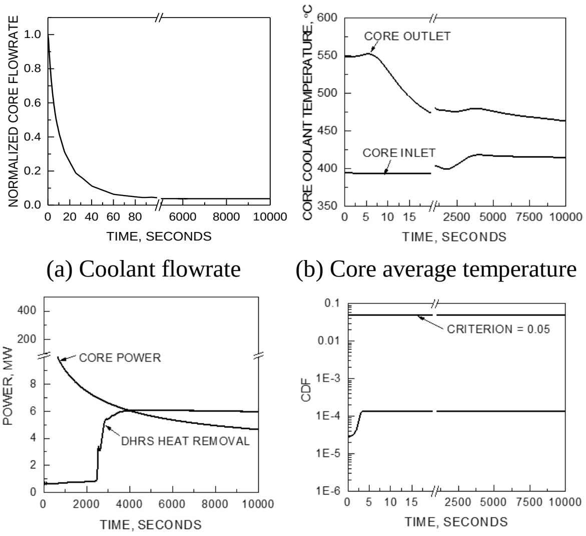
A spurious PHTS pump trip with coastdown results from the mechanical failure of both PHTS pumps or the simultaneous loss of electrical power to PHTS pumps. Fig. 5 shows the safety analysis results. At 0.0 seconds, the core outlet temperature increases due to the decrease of the core flow rates since 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 at 1.21 seconds, and the reactor trip signal is generated at 2.01 seconds, and the insertion of control assemblies starts at 2.56 seconds. At the same time as the reactor trip signal is generated, heat removal through the steam generator is lost as the IHTS and the feedwater pumps stop. Thus the heat removal is only available through the partially operating DHRS. This leads to the core inlet temperature increases. At about 2190 seconds, the 'high core inlet temperature trip' ESFAS signal reaches the setpoint. Then, the DHRS dampers are fully opened and the blower begins to operate. Around 4100 seconds, the DHRS heat removal rate exceeds the decay heat rate, and thereafter the core outlet temperature continuously decreases. As a result, CDF maintains less than 1.36E-04.



(a) Core power (b) Core average temperature

(c) Decay heat removal (d) Peak assembly temperature

Fig. 4. Safety analysis results of seismic reactivity insertions-SSE.



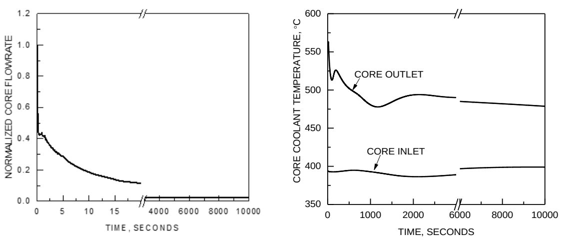
(a) Coolant flowrate (b) Core average temperature

(c) Decay heat removal (d) CDF

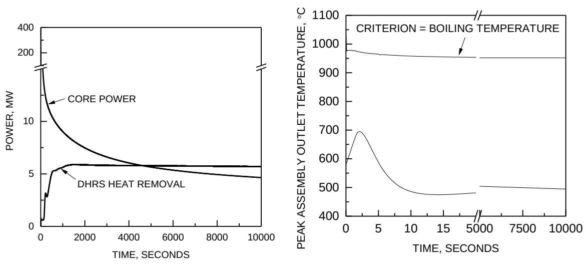
Fig. 5. Safety analysis results of spurious PHTS pump

3.4 PHTS Pump Pipe Break [Loss of Flow, DBA Class II]

This event is initiated by a decrease in core coolant flow rates as a result of a postulated DEGB (Double Ended Guillotine Break) at one of the four PHTS pump discharge pipes. Fig. 6 shows the safety analysis results. At 0.0 seconds, the core coolant flow rates rapidly decrease since a DEGB at one PHTS pump discharge pipe occurs. The core coolant temperature locally increases due to the decrease in the core coolant flow rates. As the core coolant flow rates rapidly decrease, the 'high power to PHTS flow ratio trip' signal reaches the trip setpoint at 0.025 seconds, and the insertion of control assemblies starts at 1.37 seconds. In about 31.2 seconds, the 'high central fuel assembly outlet temperature trip' ESF actuation signal reaches the trip setpoint. Then, the DHRS dampers are fully opened and the blower begins to operate. At around 4,627 seconds, the DHRS heat removal rate exceeds the decay heat rate, and thereafter the core outlet temperature continuously decreases. As a result, the peak assembly outlet temperature is 694.8 ℃.



(a) Coolant flowrate (b) Core average temperature

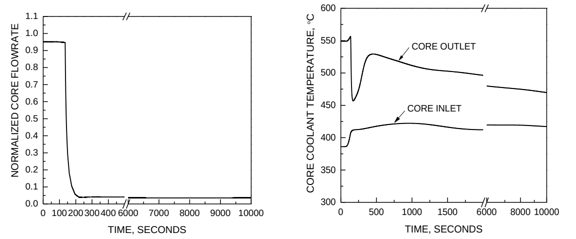


(c) Decay heat removal (d) Peak assembly temperature

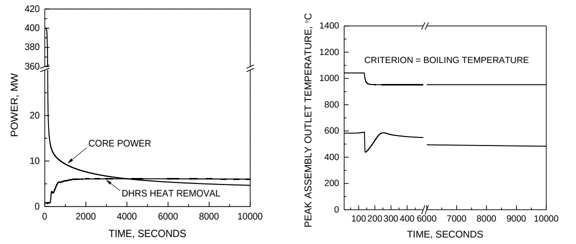
Fig. 6. Safety analysis results of PHTS pump pipe break.

3.5 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. 7 shows the safety analysis results. A high core inlet temperature trip signal is generated at 134.78 seconds. At 135.33 seconds, the reactor is tripped. The peak assembly outlet temperature is 591.9 ℃. At 154.88 seconds, the DHRS dampers are fully opened and the blower begins to operate. At around 4,069.0 seconds, the amount of DHRS decay heat removal exceeds that of core decay heat.



(a) Coolant flowrate (b) Core average temperature



(c) Decay heat removal (d) Peak assembly temperature

Fig. 7. Safety analysis results of single feedwater pump seizure.

4. Conclusions

Safety analysis of DBEs such as AOOs and DBAs for PGSFR specific design has been carried out using MARS-LMR code. As a results, representative DBEs fulfill the safety acceptance criteria with sufficient margin.

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