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An Accident Sequence Analysis for PGSFR

Hoyoung Shin and Moosung Jae*
Department of Nuclear Engineering, Hanyang University, Seoul, 04763, Korea
**Corresponding author: jae@hanyang.ac.kr*

1. Introduction

Sodium-cooled fast reactor (SFR), one of the fourth-generation reactors, has many advantages in terms of inherent safety, efficient use of uranium resources, and nuclear non-proliferation. Especially, the safety of SFR is greatly improved compared with the light water reactor.

The United States nuclear regulatory commission (U.S.NRC) utilizes probabilistic risk assessment (PRA) in almost all areas of regulatory activity, and requires the PRA to be carried out to evaluate the design safety of new advanced reactors such as the fourth-generation reactors. In Korea, the needs to evaluate whether the safety goals are met and to utilize the evaluation results in design improvement and regulation (risk-informed design and regulation) using PRA from the early stage of designing SFR, are being raised [1].

In this study, for the introduction of risk-informed design and regulation systems, the initiating events were identified using master logic diagram (MLD), and various accident sequences were developed for prototype gen-IV sodium-cooled fast reactor (PGSFR). In addition, the core damage frequency (CDF) of the initiating event was evaluated by quantifying the developed accident sequences. However, the scope of this study was limited to the quantification of LOOP accident sequences among the identified initiating events, not the entire PRA of PGSFR.

2. Methods and Results

2.1 Reference System

PGSFR was selected as a reference system in this study. It is a prototype reactor developed by the Korea atomic energy research institute (KAERI). The characteristics of PGSFR are shown in table I [2]. PGSFR is a 150 MWe pool-type SFR using metal fuel, and is being designed by KAERI. The pool-type SFR adopted in Korea has a large heat capacity. In addition, the safety is improved because the leakage in the primary heat transport system is restricted to inside the reactor vessel. And since the piping is shortened, the economic efficiency is improved [3].

Table I: The design characteristics of PGSFR [2]

Item	Specification
Designer	KAERI (Korea Atomic Energy Research Institute)
Reactor Type	Pool-Type
Reactor Power	150 MWe
Coolant Type	Sodium
System Pressure	~ 1 bar
System Temperature	390～545 °C
Fuel Material	U-Zr (initial core) U-TRU-Zr (reload core)
Fuel Cycle	～10 Months
Residual Heat Removal Systems	PDHRC (Passive Decay Heat Removal System) ADHRC (Active Decay Heat Removal System)

The conceptual design of PGSFR used in this study is shown in figure 1. The heat transport system of PGSFR consists of primary heat transport system (PHTS), intermediate heat transport system (IHTS), and power conversion system (PCS). PGSFR is characterized by the introduction of the intermediate heat transport system so that the sodium-water reaction in the steam generator does not directly affect the primary heat transport system. The decay heat removal system of PGSFR consists of two passive decay heat removal systems (PDHRS) and two Active Decay Heat Removal Systems (ADHRS) [2].

2.2 Initiating Events Identification

Based on the conceptual design data, the initiating events that could occur in PGSFR was analysed. The MLD methodology, used to identify the initiating events that cause an accident, was utilized to analyse the initiating events. The MLD is a logic diagram similar to fault tree but without formal mathematical properties [4].

The top event of the MLD developed in this study means an abnormal release of radioactive material into environment. The MLDs developed for PGSFR are shown in figure 2 to figure 7.

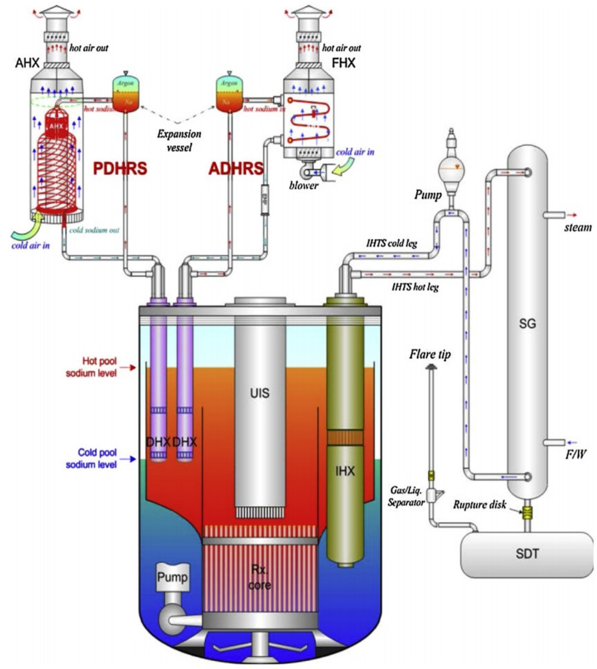

Fig. 1. The conceptual design of PGSFR [5].

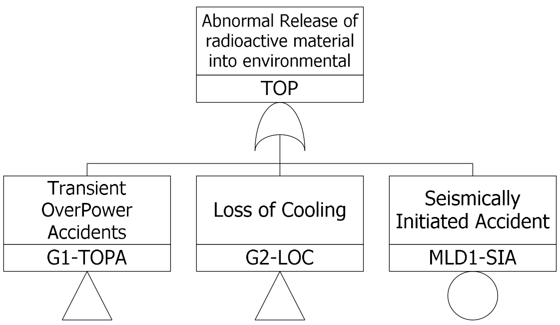

Fig. 2. The Master Logic Diagram (MLD) of PGSFR (1/6).

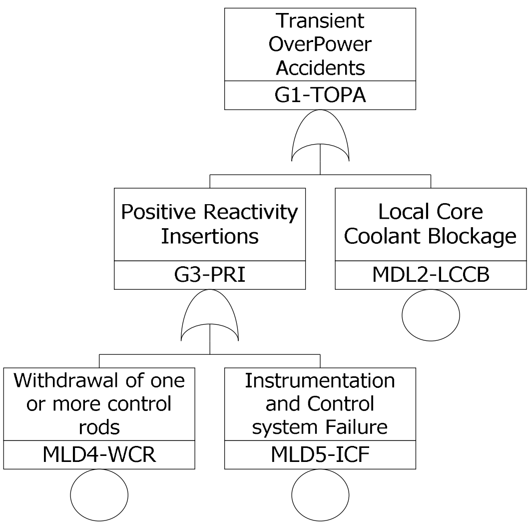

Fig. 3. The Master Logic Diagram (MLD) of PGSFR (2/6).

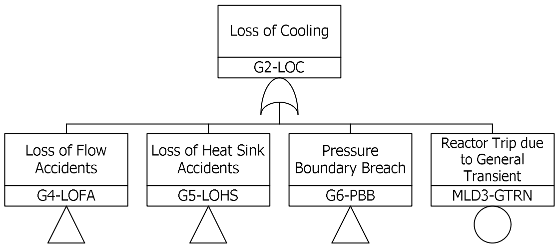

Fig. 4. The Master Logic Diagram (MLD) of PGSFR (3/6).

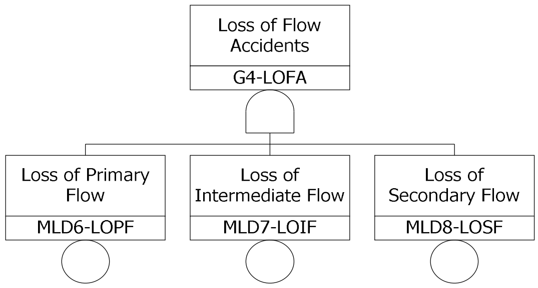

Fig. 5. The Master Logic Diagram (MLD) of PGSFR (4/6).

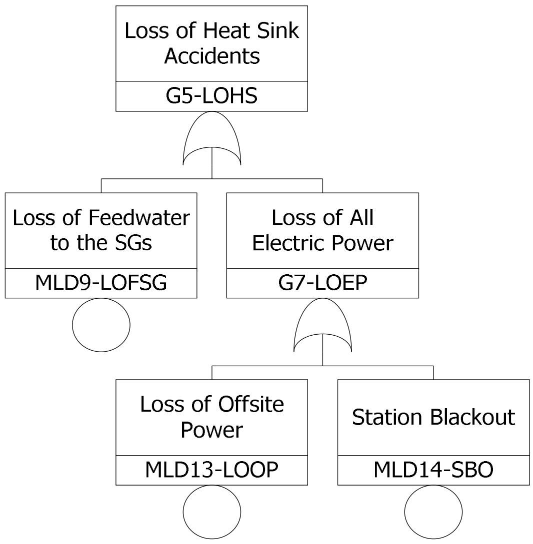

Fig. 6. The Master Logic Diagram (MLD) of PGSFR (5/6).

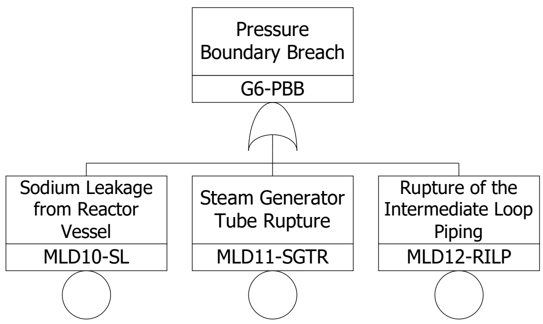

Fig. 7. The Master Logic Diagram (MLD) of PGSFR (6/6).

The identified initiating events are divided into transient overpower accidents, loss of cooling, and seismically initiated accidents. A total of 14 initiating events were identified.

1. Seismically Initiated Accident
2. Local Core Coolant Blockage
3. Reactor Trip due to General Transient
4. Withdrawal of one or more control rods
5. Instrumentation and Control System Failure
6. Loss of Primary Flow
7. Loss of intermediate Flow
8. Loss of Secondary Flow
9. Loss of Feedwater to the SGs
10. Sodium Leakage from Reactor Vessel
11. Steam Generator Tube Rupture
12. Rupture of the Intermediate Loop Piping
13. Loss of Offsite Power
14. Station Blackout

The general transients were defined as the transients causing the reactor shutdown without any failure of safety systems or components. Transient events caused by flow loss and inadvertent reactivity insertion were considered as independent initiating events because the accident sequences of them may different with those of general transients.

2.3 Accident Sequences Analysis

After the initiating event has been identified, various accident sequences, depending on whether the safety system is activated after the initiating event, should be developed and expressed as an event tree. The end states of the event tree are divided into OK and core damage (CD) state. Among the initiating events, loss of offsite power (LOOP) was selected as a representative case to model the event tree because it is an event that can occur in light water reactors, and the frequency of the initiating event is relatively high in OPR1000 [6]. The modeled event tree of LOOP is shown in figure 8.

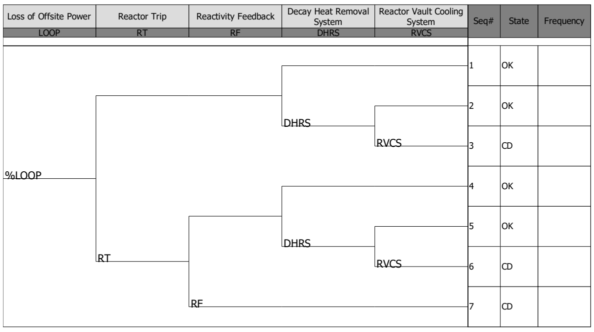

Fig. 8. The loss of offsite power (LOOP) event tree.

In order to quantify the accident sequences, reliability data on the initiating event frequencies, component failure probabilities, common cause failure probabilities, human error probabilities, and so on are required. Based on these reliability data, the fault tree of the safety systems corresponding to each heading of the event tree can be modeled, and the occurrence probability of each accident sequences can be quantified.

In this study, the CDF of LOOP accident was evaluated by quantifying the accident sequences of LOOP. The data and assumptions used to quantify the accident sequences are shown in table II.

Table II: Source of data used to quantify LOOP sequences

Headings	Frequency /Probability	Data source
LOOP	0.05 /y	NUREG/CR-5750
RT (Reactor Trip)	5.057×10^{-6}	Shin Kori 1&2 PSA
RF (Reactivity Feedback)	1.0×10^{-6} /demand	PRISM PSA
DHRS (Decay Heat Removal System)	2.55×10^{-4}	Reference [1]
RVCS (Reactor Vault Cooling System)	1.0×10^{-4}	PRISM PSA

The initiating event frequency of LOOP was assumed to be 0.05 /y for a more conservative evaluation than 0.046 /y presented in the NUREG/CR-5750 report [7].

The failure probability of reactor trip (RT) was calculated by modeling fault trees (FTs) based on Shin Kori 1&2 PSA report [6]. The modeled FT and calculation results are shown in figure 9. The calculated RT failure probability using the FTs was 5.057×10^{-6}.

Inherent reactivity feedback system reliability, RF heading, was assumed to be 1.0×10^{-6} /demand as presented in the PRISM PSA [8].

The failure probability of the decay heat removal system (DHRS), 2.55×10^{-4}, was assumed to be equal to that of unprotected loss of heat sink (ULOHS) accident of PGSFR presented in reference 1 [1].

The failure probability of reactor vault cooling system (RVCS) was assumed to be 1.0×10^{-4} for a more conservative evaluation than 4.2×10^{-5} presented in the PRISM PSA [8].

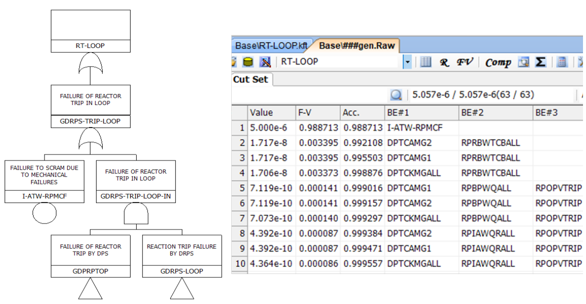

Fig. 9. The fault tree and calculated failure probability for RT.

The total CDF for LOOP accident sequences were evaluated as 1.28×10^{-9} /ry as shown in figure 10. This is about 0.15% of 8.09×10^{-7} /ry, the CDF for LOOP of Shin Kori 1&2 [6]. The evaluated CDF proved the high safety of PGSFR.

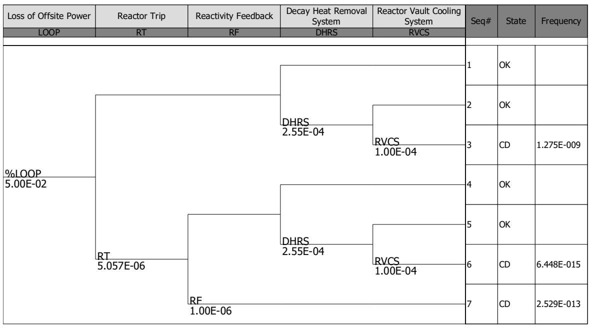

Fig. 10. The accident sequences quantification result for LOOP.

3. Conclusions

SFR is one of the reactors selected at the international forum on fourth-generation nuclear power systems held in 2002, and it is being studied in various countries including Korea. In Korea, the PRA is performed from the early stage of PGSFR design and efforts are being made to utilize the results in design improvement and regulation.

In this study, the initiating events that can occur in PGSFR were identified using the master logic diagram. As a result of initiating event analysis, a total 14 initiating events were identified, and the event tree for LOOP accident sequences were modeled among the identified initiating events. In addition, the CDF of initiating event for LOOP, was evaluated by quantifying the developed accident sequences. As a result, the CDF was evaluated as 1.28×10^{-9} /ry. This is about 0.15% of that of the Shin Kori 1&2, proving the high safety of PGSFR. The results of the analysis of the initiating events and accident sequences derived from this study can be used as a basic data for the risk assessment of PGSFR.

Acknowledgements

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