Sensitivity Analyses of Core Damage Frequency on the Design Alternatives for SFR-600 Conceptual Design

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1. Introduction

KALIMER-600 is under design with defense in depth concept with active, passive, and inherent safety features. In this report we tried to evaluate the safety level of KALIMER-600 with quantitative measure of CDF, with conventional PSA methodology. To quantify the CDF in quantitative way is the goal of Level-1 PSA.

The PSA methodology has been used in conventional nuclear power plants which mainly have active safety systems. Here we tried to develop PSA models for KALIMER-600 with inherent, passive as well as active safety systems. Even though there are still some limitations in developing PSA models for plant such as KALIMER-600 with its inherent and passive systems, the core damage scenarios are identified and they are developed by using the event tree and fault tree models.

With the reliability data mainly quoted from the database of conventional Light Water Reactor and with some assumptions and expert judgments which cannot exist in LWR database. The core damage scenarios and the frequency of KALIMER-600 are identified. Sensitivity studies on the design alternatives of safety systems and PSA assumptions are also performed.

2. Preliminary Level-1 PSA Models and Results

Accident scenarios which lead to the core damage should be identified for the development of a Level-1 PSA model. KALIMER-600 is under design using safety systems with passive as well as active safety features. It has passive safety features such as passive shutdown functions, passive pump coast-down features, and passive decay heat removal systems. The passive decay heat removal system is called PDRC (Passive Decay Heat removal Circuit), which is installed in reactor vessel. The active decay heat removal system is called IRACS (Intermediate Reactor Auxiliary Cooling System), which is installed in intermediate loop. The KALIMER-600 has also inherent reactivity feedback effects such as Doppler, sodium void, core axial expansion, control rod axial expansion, and core radial expansion, etc. For the reactor trip functions, independent and diverse features are assumed among the primary, the secondary reactor trip systems and SASS (Self-Actuated Shutdown System).

The accident scenarios, which lead to core damage, are under investigation for the KALIMER-600 reactor concept. Even though the occurrence frequency of multiple failure events is very low, we try to understand the accident spectrum in metallic fuel SFR by available core thermal hydraulic analysis computer codes. The transient simulation codes such as SSC-K and MARS-LMR are currently under development in KAERI.

A few accident scenarios have been simulated for the ATWS (Anticipated Transient Without Scram) events such as the Unprotected Transient Overpower (UTOP), the Unprotected Loss of Flow (ULOF), and the Unprotected Loss of Heat Sink (ULOHS) events with degrade safety systems or functions. Each accident scenario is under simulation and analysis with available cases and unavailable cases of the safety systems such as reactor shutdown systems, pump coastdown feature, and inherent reactivity features (Doppler, sodium voiding, core axial expansion, control rod axial expansion, and core radial expansion, etc.).

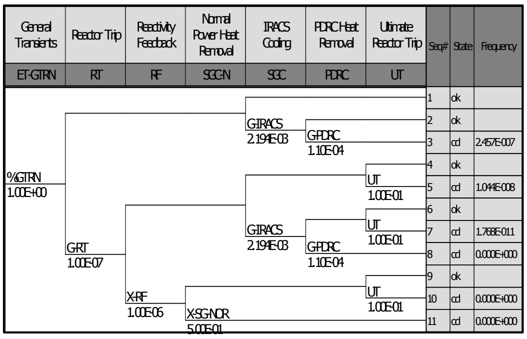

Fig. 1. An Example of Level 1 System Event Tree of General Transient Accident for SFR-600

Using the conventional event tree and fault tree method which is used in LWR PSA, level 1 PSA are under development now for the metal fuel KALIMER-600 conceptual design. The event categories which are considered as initiating events are Reactivity Insertion Accident (RIA), Loss of Primary Flow Accident (LOPF), Loss of Intermediate Flow Accident (LOIF), Loss of Secondary Flow Accident (LOSF), Loss of Electrical Power (LOEP), Sodium water Reaction in Steam Generator (SWR), and Reactor Vessel Rupture (RVR). The fault trees for PDRC and IRACS are made using the conceptual design information. Reliability data for the initiating event frequencies and component failure rates are quoted from the available sources for the fast reactor design report such as PRISM and current light water reactor PSA reports. For the initiating events and components, the reliability data cannot be obtained from the available sources, most of them are assumed based on the on the current generation LWR experience and practices. For the quantification of the core damage frequency the AIMS PSA Tool is used, which is developed by KAERI (Korea Atomic Energy Research Institute) for the streamlining of the PSA woks.

Preliminary level 1 PSA models and the results are described in this report. Sensitivity study results on the design alternatives of safety systems and PSA assumptions are briefly described.

Table 1 shows initiating events frequencies and relative core damage frequencies contributions for SFR-600. Table 2 shows the results (increasing ratio of CDF to base case) of the various design alternatives on the safety systems.

Table 1. Initiating Events Frequencies and Relative Core Damage Frequencies Contributions for SFR-600

Initiating Event	IE Freq/yr	CDF(%)
General Transients	1.00E+00	25.58
Vessel Leak	1.00E-04	21.67
Sodium Water Reaction in SG	1.00E-03	20.84
PDRC Unavailable	3.00E-03	7.85
Loss of Primary Flow	3.00E-01	7.66
Loss of Secondary (Feedwater)	3.00E-01	7.66
Loss of Intermediate Flow	3.00E-01	7.66
Loss of Electric Power	3.00E-02	1.05
Reactivity Insertion Accident	1.00E-03	0.02
Fast Reactivity Insertion Accident	1.00E-10	0.01
Total		100

3. Sensitivity Study on the Design Alternatives

KALIMER-600 is in the design stage where various configurations are under consideration now. A lot of assumptions are used in performing this PSA. Therefore we try to evaluate the impact of configuration change and assumptions on the core damage frequency. This kind of study would help to decide the configurations of safety systems of KALIMER-600. Table 2 shows the results (increasing ratio of CDF to base case) of the various design alternatives on the safety systems.

There are one passive and one active system for the decay heat removal function. The sensitivity studies are performed for the cases for various design alternatives when only passive or active system is installed (Cases 1, 2 and 6). The CDF is increased beyond the acceptable level in every case. The results show that both passive and active features are essential.

The base case assumes that there are two independent groups of the RPS and there are two gas turbine generators to support the safety grade electric power system. The CDF also increases beyond the unacceptable level if any feature is removed from the base case (Cases 7 and 8).

The several sensitivity analyses are performed for assumptions made such as the solidification frequency (Case 3), the phenomenological uncertainty of PDRC (Case 4) and the reactivity feedback probability (Case 9). These sensitivity analyses show that the phenomenological uncertainty of PDRC has a big impact on the CDF and should be studied in detail in the future.

Because of assumptions made, the following areas are identified for future studies to improve the quality of the PSA;

- Revise the PSA model as the design of KALIMER-600 is finalized.
- Develop the methodology to estimate the reliability for newly introduced system or component for the SFR.
- Develop the methodology to estimate the passive system reliability. The method can be applied to estimate the phenomenological uncertainty for the PDRC and reactivity feedback.

In conclusion, we identified that the current design features on the safety systems are the most acceptable in terms of risk as well as cost.

Table 2. Sensitivity study results of the various design alternatives on the safety systems

System	Base Case Assumptions	Assumptions in Sensitivity Study	Increasing ratio
PDRC	With PDRC	*(1) No PDRC	5176
	2 x 50% Passive	(2) 2 x 100% Active System	10.2
	Solidification frequency (0.003/yr)	(3) 10 times increase (0.03/yr)	1.76
	PDRC reliability (1e-4)	(4) 10 times increase (1e-3)	5.66
	2 x 50% Passive	(5) 2 x 100% Passive	0.86
IRACS	2 x 100%, Safety class electric power (2 Gas Turbine backup)	(6) No IRACS	364
EPS	2 Gas Turbine backup	(7) No Gas turbine	11.4
RPS	2 diverse systems (1st, 2nd)	(8) No 2nd RPS	23.4
	Reactivity Feedback Failure Prob. (1e-6)	(9) Reactivity Feedback Failure Prob. (0.1)	1.03

*(Case Number)

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