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A Study on Level 1 Internal PSA of Prototype SFR Conceptual Design

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1. Introduction

A prototype SFR(Sodium Fast Reactor) is under design with defense in depth concept with active, passive, and inherent safety features. In this paper, we tried to evaluate the safety level of the prototype SFR with quantitative measure of Core Damage Frequency(CDF), with conventional PSA methodology. A basic PSA was done on KALIMER-600 [1]. However, since some design changes were done in the prototype SFR, a new Level 1 PSA was performed. The core damage scenarios and the frequency of the prototype SFR are identified. Sensitivity studies on the design alternatives of safety systems and PSA assumptions are also performed.

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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. The prototype SFR 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 ADRC (Active Decay Heat removal Circuit), which is also installed in reactor vessel. The prototype SFR 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).

Using the conventional event tree and fault tree method which is used in LWR PSA, level 1 PSA are under development now for the prototype SFR 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 ADRC 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 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 paper. Sensitivity study results on the design alternatives of safety systems and PSA assumptions are briefly described.

3. Sensitivity Study on the Design Alternatives

The prototype SFR 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 CDF. This kind of study would help to decide the configurations of safety systems of the prototype SFR. It is assumed that the base model has 50% x 2 PDRC, and 50% x 2 ADRC for the decay heat removal function, and recirculation for the long term cooling. Also, it is assumed that the failure rate of the recirculation is 0.1. In this condition, CDF of the base model is calculated as 5.4E-7 as shown in Table 1. If the recirculation shown in Fig. 1 is modeled by a fault tree, the failure rate of the recirculation becomes 0.028, and which make CDF be 4.5E-7. A sensitivity study is performed for the cases for various design alternatives, and an example of which is that the capacities of PDRC and ADRC are increased to 100% x 2, respectively. In this case, CDF can be improved to 2.5e-7 as shown in Table 1.

Since the rated core power of the prototype SFR was lowered to 393 MWth, RVACS (Reactor Vessel Aux Cooling System) as shown in Fig. 2 can be used for cooling of the residual heat removal. Fig. 3 shows that only small capacity SFR can afford the RVACS [2]. If RVACS is added to the prototype, the general transient ET is modified like Fig. 4, and CDF becomes 5.3E-8 as shown in Table 1.

In conclusion, a level 1 internal PSA for the prototype SFR was performed. If RVACS and recirculation features are modeled, then better CDF can be acquired.

Acknowledgement

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REFERENCES

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Table 1. A Sensitive Study for the Alternative Design for Residual Heat Removal in the Prototype SFR

Design Options for Residual Heat Removal	CDF
o Basic Model	5.4e-7
o If Recirculation System Modeled by FT	→ 4.5e-7
o and If PDRC/ADRC Capacity Changed(50% → 100%)	→ 2.5e-7
- 2 Passive + 2 Active	
. Passive : 2 x 100%	
➢ Damper (pneumatic) Open required	
. Active : 2 x 100%	
➢ Damper (Pneumatic) Open required	
➢ Pump, Blower Start + Emergency Power	
➢ Transition to Passive Mode 2 x 50%	
o and RVACS Added	→ 5.3e-8

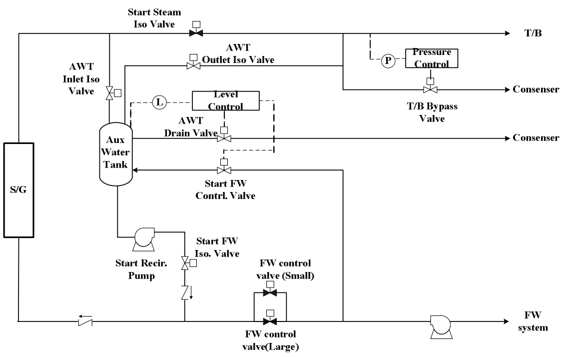

Fig. 1 Recirculation system

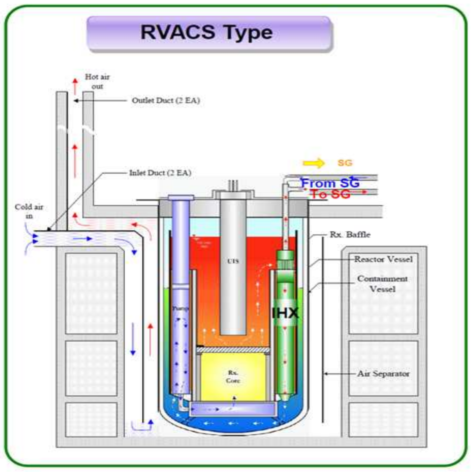

Fig. 2 RVACS of the prototype SFR

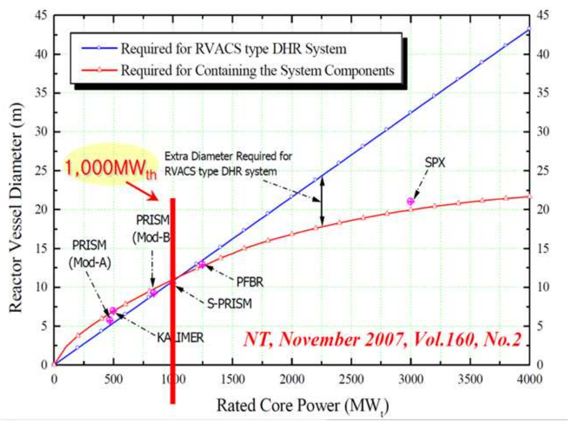

Fig. 3 RVACS possibility based on reactor core power

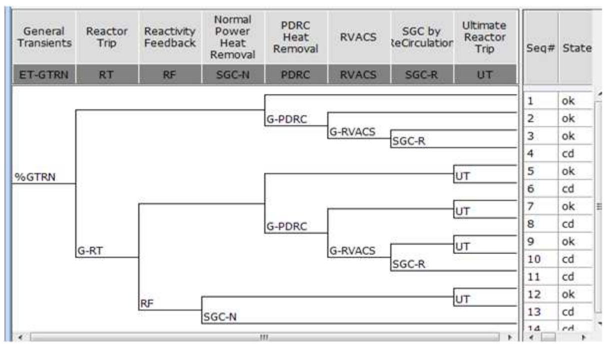

Fig. 4. An Example of Level 1 System Event Tree of General Transient Accident for the Prototype SFR.