Safety Function Evaluation for the Decay Heat Removal System of a Sodium Cooled Fast Reactor

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1. Introduction

Korea Atomic Energy Research Institute (KAERI) has been developing an advanced sodium-cooled fast reactor (SFR) that meets the design goals of GEN IV reactors.

The sodium-cooled fast reactor is the pool type sodium cooled fast reactor with the thermal power of 1548.2 MW and the core loaded with metal fuel, which is consists of primary heat transport system, intermediate heat transport system, steam generating System, and decay heat removal system.

The decay heat removal system is composed of 2 units of passive decay-heat removal circuits (PDRC) and 2 units of active decay-heat removal circuits (ADRC) considering 200% heat removal capacity. Decay heat removal system removed the decay heat produced in the core during planned and unplanned shutdown of the reactor to respect design limits.

For the safety function evaluation for the decay heat removal system, the 4 DBE's(Design Bases Events) are analyzed using MARS-LMR.

2. Method and Results

The overview of the plant is shown in Fig.1 and the overall composition is similar to the KALIMER-600.

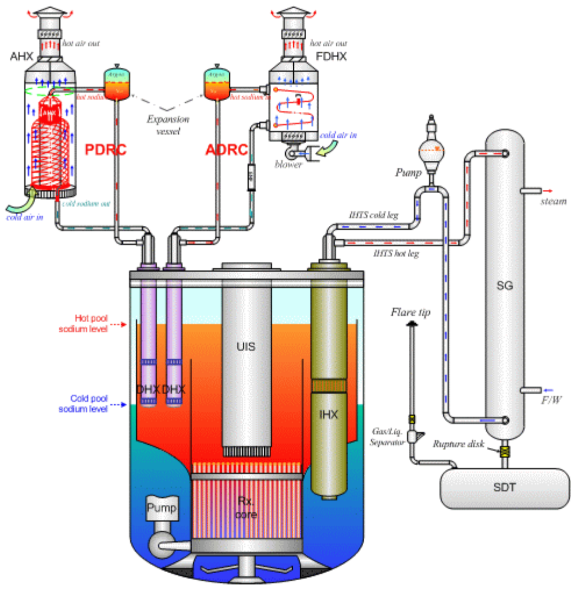

Fig. 1. The schematic diagram of a sodium cooled fast reactor

Fig.2. shows the hot rod temperatures for variations of the number of DRC loop to verify the design margin. The results indicate that the two-loop DRC operation is sufficient to meet safety standards. However the actual accident considered only a single failure mode, the two-loop operation with DRC ensured the safety.

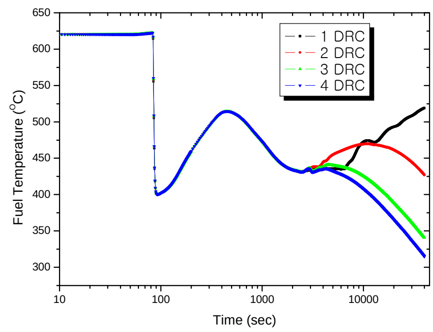

Fig. 2. Hot rod temperatures behaviors at LOHS

Following figure 3~6 show the results of 4 representative DBE's: TOP (Transient of Over Power), LOF (Loss Of Flow), LOHS (Loss Of Heat Sink) and Pipe Break.

The TOP accident is initiated by a possible malfunction of the reactivity controller due to control rods withdrawal and the core power is rapidly increased by a positive reactivity insertion. The LOF is initiated by the loss of core cooling capability due to the pumping failure of primary pumps. The LOHS is caused by a loss of feedwater to all SG's or all pumps trip in IHTS (Intermediate Heat Transport System). The accident of the pipe break is assumed to occur at between one of inlet pipes and the inlet plenum.

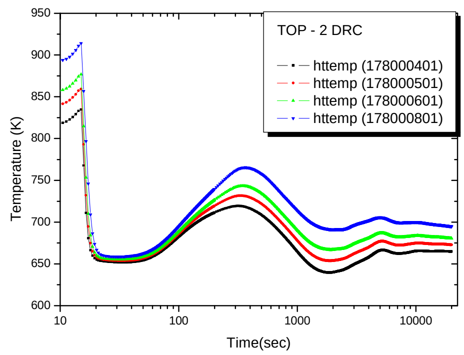

Fig. 3. Hot rod temperatures behaviors at TOP

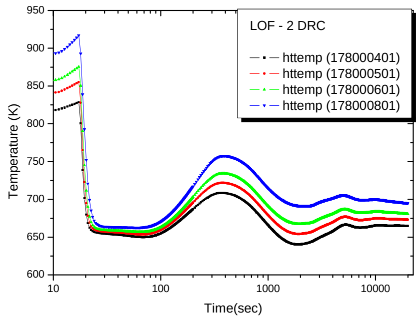

Fig. 4. Hot rod temperatures behaviors at LOF

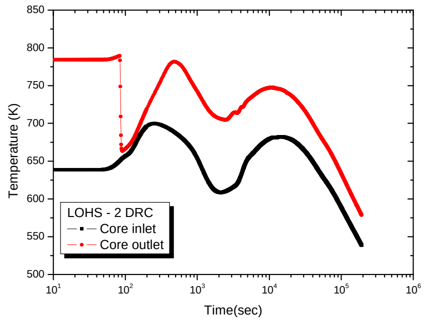

Fig. 5. Coolant temperatures behaviors at LOHS

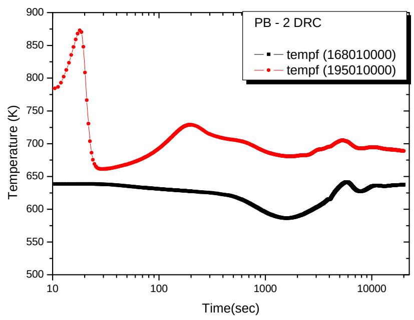

Fig. 6. Coolant temperatures behaviors at Pipe Break

3. Conclusions

For the safety function evaluation for the decay heat removal system, the 4 DBE's are analyzed using MARS-LMR and the results are summarized in table 1. The results indicate that the two-loop DRC operation is sufficient to meet safety standards.

Table I: Summary of Analysis Results

Number of DRC Loop	Fuel Temperature (℃)			
	TOP	LOF	LOHS	PB
2-DRC	640.75	643.81	622.42	701.70
3-DRC	640.74	643.79	622.40	701.70
4-DRC	640.72	643.78	622.39	701.67

REFERENCES

[1] K. S. Ha, Safety Evaluation for Transient of Demonstration reactor, Korea Atomic Energy Research and Institute, 2011.
[2] D. H. Hahn, KALIMER-600 Conceptual Design Report, Korea Atomic Energy Research and Institute, 2007.