Analysis of Loss of Flow Transients in the Conceptual Design of a PGSFR with MARS-LMR

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1. Introduction

KAERI has been developing a conceptual design of the PGSFR, which is the pool type sodium cooled fast reactor with the thermal power of 392.6 MW and the core loaded with metal fuel. The PGSFR consists of the Primary Heat Transport System, the Intermediate Heat Transport System, the Steam Generating System including Balance of Plant, and a decay heat removal system.

The PHTS is placed in a large pool similar to the demonstration fast reactor. The IHTS transfers reactor-generated heat from the intermediate heat exchanger (IHX) of the PHTS to the steam generator system. The IHTS consists of two loops and each loop has two IHXs, one pump, and a steam generator. The IHTS loop is thermally coupled to the PHTS by the IHXs and to the steam generator system by a steam generator. The SGs consists of two independent steam generation loops and converts a sub-cooled water to a superheated steam by transferring the heat from the intermediate sodium to the water/steam. The size of the steam generator is decreased compared with the demonstration fast reactor.

The DHRS is composed of 2 units of PDHRS and 2 units of ADHRS and each loop is equipped with single sodium-to-sodium decay heat exchanger, single sodium-to-air heat exchanger, a surge tank to control the pressure of the loop and the pipes. In addition, a damper driven by the emergency generator (Diesel Generator) is attached to the AHX and the FHX. The damper is designed with the concept of the fail-open similar to the passive type. The DHRS has the heat removal capability of 400 % by the 4 units.

This study presents the safety analysis results concerned with the safety assurance of the PGSFR design. The analyses of Loss of flow (LOF) event are followed using the MARS-LMR code.

2. Methods and Results

The LOF means the loss of core cooling capability due to a pumping failure of the primary pumps. The cause of this accident is a mechanical pump failure or a power loss. If the loss of off-site power occurs, PHTS coolant flow rate decreases rapidly. A representative of the initial incident is caused by all the PHTS pump failure at the same time.

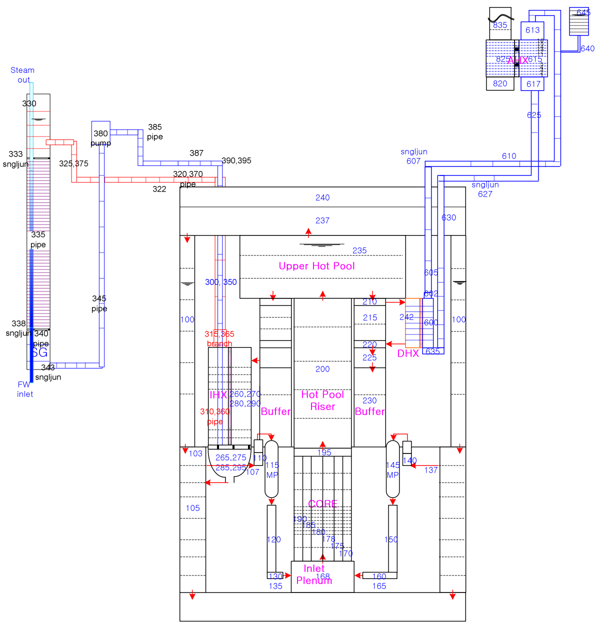

Figure 1 MARS-LMR Nodalization for PGSFR

The event is assumed to start at the full power condition. The ANS-79 model is used for core decay power after a reactor scram. Two independent PDRC’s and one ADRC are assumed available by applying a single failure criterion.

The imbalance between the reactor power and the primary flow rate is a main safety concern in the LOF event. To prevent the occurrence of severe imbalance between power and flow, the PGSFR is designed so as far the reactor to be tripped by a high power/flow trip.

In this simulation, all the primary pumps are tripped at 10 seconds, and the reactor scram occurs at 17 seconds. The reactor power and flow rate decrease. The decrease of the core specific power is greater than the specific flow reduction in the early period of pump coastdown and the core outlet temperature decreases nearly vertically.

Figure 2 shows the coolant temperature behaviors during the LOF accident. The core outlet temperature decreases nearly vertically and the core inlet and outlet temperature rise due to decreased flow rate by the PHTS pump coastdown and the reduced heat transfer to IHTS by the feedwater isolation.

Figure 3 shows the decay heat removal rate of DHRS compared with reactor power. The AHX Dampers are assumed to open at 5 seconds after reactor trip. The

DHX heat removal exceeds core power after 3400 seconds, and the core outlet temperature decreases continuously.

Figure 4 shows the cladding temperature behaviors with time. The peak cladding temperature was calculated at 645.04℃. The temperature satisfies the safety criteria.

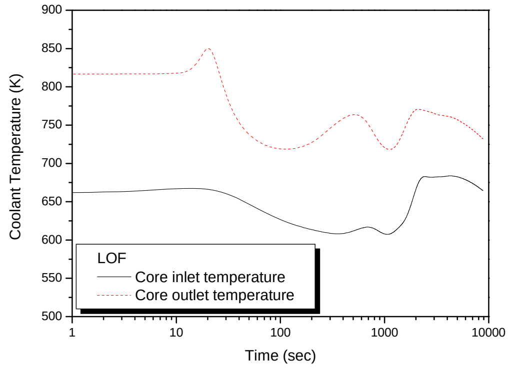

Figure 2 Coolant Temperature Behaviors for a LOF

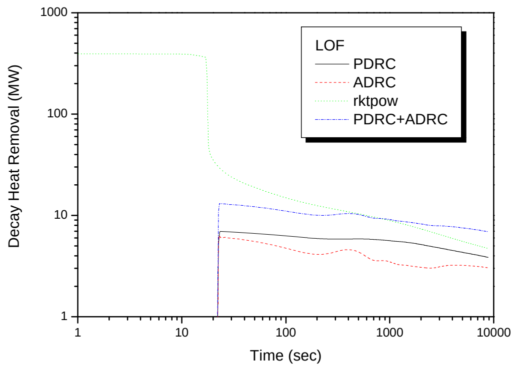

Figure 3 Reactor Power Compared with DHRS Heat Removal at LOF

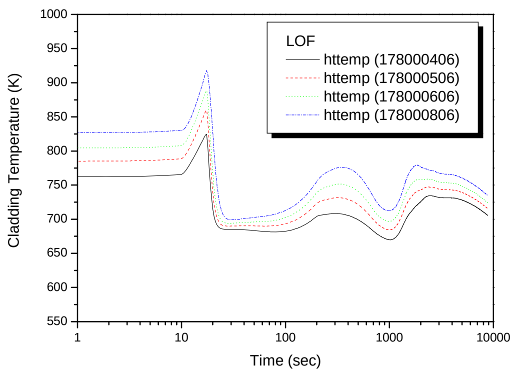

Figure 4 Cladding Temperature Behaviors for a LOF

3. Conclusions

The reactor safety is determined by the peak cladding temperature just before the reactor trip and the peak core exit temperature during a long-term cooling. The PGSFR has a sufficient heat removal capability during long-term cooling period and the peak cladding temperature is also evaluated to meet the safety criteria. As a consequence, the PGSFR can carry out its safety functions required for preventing the LOF accident with an appropriate margin.

REFERENCES

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[2] K. S. Ha et al, Safety Evaluation for Transients of DFR, KAERI/TR-4288/2011, Korea Atomic Energy Research and Institute, 2011.