A Safety Function Evaluation of the Advanced Sodium-Cooled Fast Reactor in a Loss of Heat Sink Accident

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1. Introduction

Korea Atomic Energy Research Institute(KAERI) has been developing a conceptual design of the advanced sodium-cooled fast reactor.

The Primary Heat Transport System(PHTS) is placed in a large pool similar to KALIMER-600. The Intermediate Heat Transport System(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 structure of the IHX is simpler than the KALIMER-600 by removing the multi-layered horizontal plates, which reduces the flow induced vibration. The steam generator 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 increased compared with the KALIMER-600 caused by the change of the core inlet temperature. The core outlet temperature is 510 ℃ lower than 545 ℃ of the KALIMER-600 due to the change of the fuel cladding material from the modified HT9 to the HT9. Figure 1 shows the configuration of heat transport system of DFR.

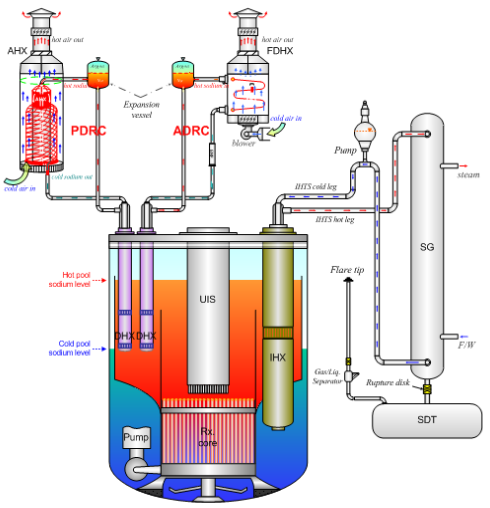

Figure 1.Configuration of Heat Transport System

The decay heat removal system(DHRS) is composed of 2 units of passive decay heat removal circuits(PDRC) and 2 units of active decay heat removal circuits(ADRC) 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 FDHX. The damper is designed with the concept of the fail-open similar to the passive type. The big difference from the decay heat removal system of the KALIMER-600 is the submerged location of the DHXs and the heat transfer mechanism. The DHRS has the heat removal capability of 200 % by the 4 units. The ADRC has been designed to perform a half function by the natural circulation, even if the pump does not work. As a result of the design change of the decay heat removal systems, it is expected that the reliability of the whole plant would be improved compared with that of the KALIMER-600. This study presents the safety analysis results concerned with the safety function evaluation of the safety systems.

2. Method and Results

The MARS-LMR code was used as a system transient safety analysis code. Figure 2 shows the nodalization of the advanced sodium-cooled fast reactor for MARS-LMR calculation. 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. The isolation time of the SG feed water line is the same as the pump trip time. Two independent PDRC's and one ADRC are assumed available by applying a single failure criterion.

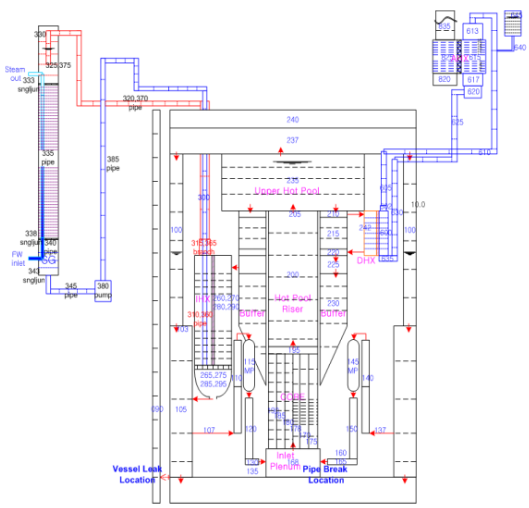

Fig. 2. MARS-LMR Nodalization for SFR 600 MWe

The LOHS accident was assumed to occur from initiated the steam generator feedwater isolation. IHTS pumps and PHTS pumps are also stopped with the assumption that the loss of offsite power occurred at 5 seconds after the reactor trip. Therefore, the residual heat removal is achieved only by the evaporation of water in SG tubes and by the DHRS after the accident.

In this simulation, a loss of feedwater to SG's is assumed to occur at 10 seconds. The IHX inlet and outlet temperatures rise because of the decrease of heat removals by SGs. The reactor is tripped by an abnormal rise of the IHX inlet temperature after the accident. The reactor trip occurs late unlike other accidents. After the pump trip, the coolant temperatures go up rapidly and reach the maximum cladding temperature. The residual heat removal through the steam generators continues until the coolant in the SG tubes runs out due to dry-out. If water inventory in steam generators run out, the natural circulation will be stopped in IHTS. The residual heat removal is achieved only by the DHRS and the core outlet temperature decreases continuously.

Fig. 3 and Fig. 4 show the results of assessing the effect of the AHX air inlet temperature and the AHX damper open time. As the AHX air inlet temperature decreases, the cooling performance is improved. The current design of the air temperature is 40 ℃, which is a conservative value enough to ensure that safety standards were satisfied. According to the result of analyzing the effect of the AHX damper open time, the most conservative value of the damper open time is about 800 sec. The difference of the latter peak fuel temperature compared with the current design of 1800 sec is about 4 ℃. The latter peak fuel temperature is lower by 200 ℃ than the initial peak fuel temperature, thus the impact of damper open time is concluded to be slight at each operating condition.

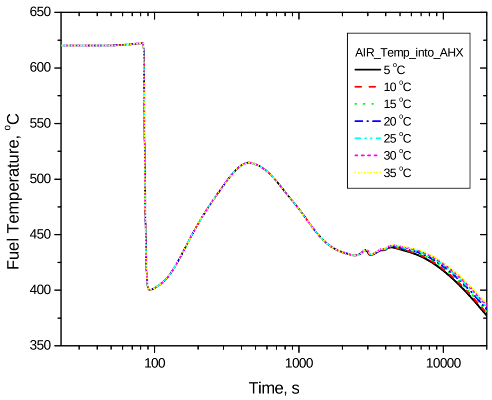

Fig. 3. The effect of the air inlet temperature of the AHX

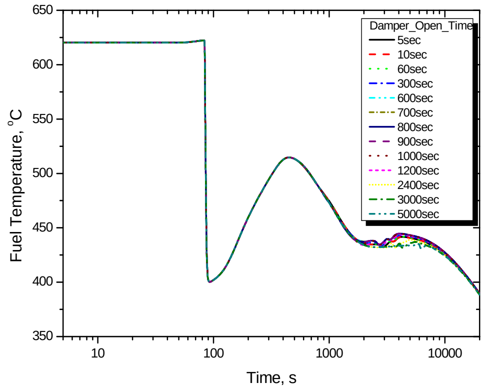

Fig. 4. The effect of the AHX damper open time

3. Conclusions

The safety analysis is performed for the conceptual design of the advanced sodium-cooled fast reactor using MARS-LMR. The LOHS accident is analyzed in order to verify the functionality of safety systems with the effect of the air inlet temperature in the AHX damper and the AHX damper open time. Through the analyses, it is shown that the design of the advanced sodium-cooled fast reactor is appropriately performed and the advanced sodium-cooled fast reactor can carry out its safety functions required for preventing the accidents with an appropriate margin.

REFERENCES

[1] K. S. Ha, Safety Evaluation for Transient of Demonstration reactor, Korea Atomic Energy Research and Institute, 2011.