A Scoping Analysis of the Performance Test Loop of PDRC with the MARS-LMR Code

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1. Introduction

A scoping analysis is conducted to verify the performance of PDRC (Passive Decay heat Removal Circuit) with MARS-LMR in the sodium thermal-hydraulic experimental facility, which is based on a scaled-down length of 1/5 for length, 1/125 for volume of the KALIMER-600. PDRC loops consist of decay heat exchanger (DHX), air heat exchanger (AHX), and pipes. When a transient accompanies a pump trip and occurs the increase of hot pool level and overflow of coolant into the shell side of DHX where the heat removal rate rapidly increases with beginning of the overflow, the heat removal through the PDRC loops is going to be balanced with the core heat generation rate to maintain the reactor condition within the safety limit.

2. Analysis Methods and Results

Fig. 1 shows the MARS-LMR [2] nodalization for the system. In the primary system two main pumps take sodium from the pool and discharge it into inlet pipes. Then the flow is entered into the inlet plenum. The sodium is heated through 4 core regions and mixed in an outlet plenum of the reactor. Then the sodium goes IHX(Intermediate Heat eXchanger) inlet through lower hot pool nodes. In the IHX, the sodium transfers its heat to the sodium of the intermediate loop. The primary sodium leaving the IHX dumps directly back into the cold pool.

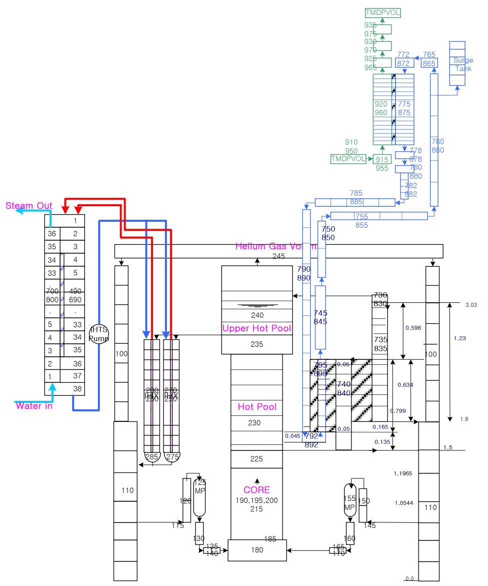

Fig.1. Decay heat removals by AHXs

Table 1 shows the analysis results of steady-state calculation. The result shows that the flow-rate through an AHX is smaller than an expected value. It is caused by the uncertainty to apply the thermal hydraulic correlation to the helical design of tubes so that some modifications are needed such as installations of flow-rate control valves.

Table 1. Analysis Results of Steady-state

Design parameter	Test loop	MARS-LMR
Power, MW # of PHTS pump # of IHX # of PDRC	27.25 2 4 2	27.25 2 4 2
Core in/out temperature, °C Flowrate, kg/s IHX in/out temperature, °C	390/545 138.3 320.7/526	390/545 138.3 320/529
PDRC flowrate, kg/s AHX flowrate, kg/s	0.58 0.49	0.54 0.3

The LOF means the loss of core cooling capability due to the pumping failure of primary pumps. In this simulation, all primary pumps are tripped at 10 seconds, thereby the reactor scram occurs by the low primary pumping flow rate. LOF event is assumed to be occurred at the full power condition. The additively considered assumptions are as follows: (1) The reactor scrammed by a high power trip of 111 %, high core outlet temperature of 555 ℃ or low pumping flow rate of 84 %. (2) The isolation time of SG (Steam Generator) feed water line is the same as the pump trip time. (3) Two independent PDRC's are available.

Table 2 shows the analysis results of LOF event.

Table 2. Analysis Results of LOF event

Accident	KALIMER-600	Test loop
Reactor trip	11.84 s	11.234 s
IHTS Tc~Th	~700 s	~1000 s
PDRC Overflow start	~6250 s	6218 s
Coolant temperature at outlet,	625 °C	574.7 °C

Fig. 2 shows the over-flow behavior from a hot pool to a DHX shell.

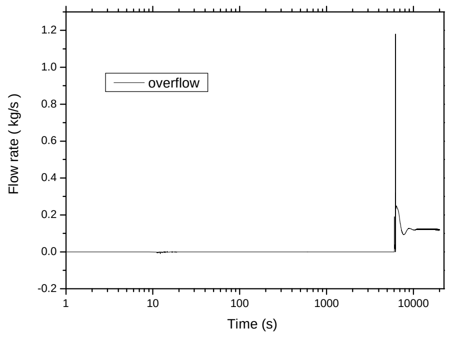

Fig.2. DHX over-flow to PDRC system at LOF

The heat removal through the PDRC loops is going to be balanced with the core heat generation rate to maintain the reactor condition shown in Fig.3. And coolant temperatures behaviors in core inlet/outlet are shown in Fig.4.

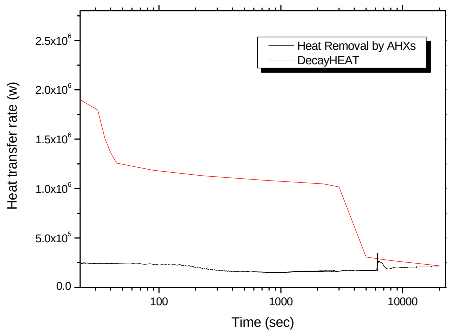

Fig.3. Decay heat removals by AHXs

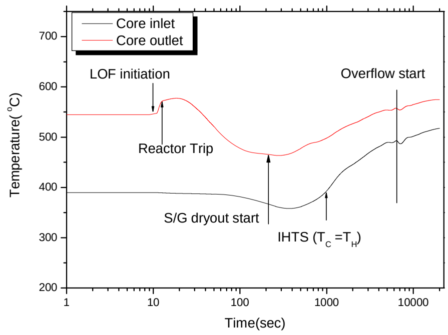

Fig.4. Coolant temperatures behaviors at Core inlet/outlet

3. Conclusions

A scoping analysis is performed for PDRC test loop with MARS-LMR. The result of steady-state calculations shows that the flow-rate through an AHX is smaller than an expected value. It is caused by the uncertainty to apply the thermal hydraulic correlation to the helical design of tubes. The result of LOF event shows that the design maintains its safety functions required for the mitigation of accidents. But the outlet coolant temperature is lower by 50 ^{o}C and an overflow from a hot pool is delayed than an expected value. It means that the PDRC systems are designed with an over margin.

REFERENCES

[1] T. H. Lee, Preliminary test requirements for the performance test of passive decay heat removal system of sodium-cooled fast reactor, Korea Atomic Energy Research and Institute, 2009.
[2] H. Y. Jeong, Conservative analysis of TOP and LOF for KALIMER-600 with the SSC-K code, Korea Atomic Energy Research and Institute, 2009.