Feasibility Study of Hybrid Heat Pipe Control Rod Application on Nuclear Power Plant using UNIST Reactor Innovation LOop(URI-LO)

Hyeon Ji Kim, Do Yeong Lim, In Cheol Bang*
Department of Nuclear Engineering
Ulsan National Institute of Science and Technology (UNIST).,
50 UNIST-gil, Ulju-gun, Ulsan, 44919, Republic of Korea
**Corresponding author: icbang@unist.ac.kr*

1. Introduction

After Fukushima nuclear power plant accidents, the demand for passive safety nuclear power plants is growing. Various safety systems have been proposed to deal with the accidental transient such as station blackout(SBO). During the SBO which accompanies the loss of all AC power, a passive safety system that operates without operator intervention is needed to cool the reactor core and to prevent fuel meltdown. In order to overcome the risk of loss of coolability, a hybrid heat pipe control rod(HPCR) was proposed, which is a combination of the heat pipe and the control rod [1]. After the reactor shut down due to the insertion of the control rod in the event of SBO, HPCR can play the role of both neutron absorber and passive residual heat removal device. In this study, the application feasibility of HPCR to nuclear power plants was evaluated with URI-LO(UNIST reactor innovation loop). The URI-LO is an APR-1400 1/8 scaled-down thermal-hydraulic integral effect test facility which is depicted in Fig. 1 [2].

The experimental evaluation of the heat removal rate of HPCR under the SBO scenario was conducted with the URI-LO facility. The computational fluid dynamics(CFD) simulation with ANSYS Fluent was conducted to analyze the thermal-hydraulic phenomena inside the HPCR operating conditions.

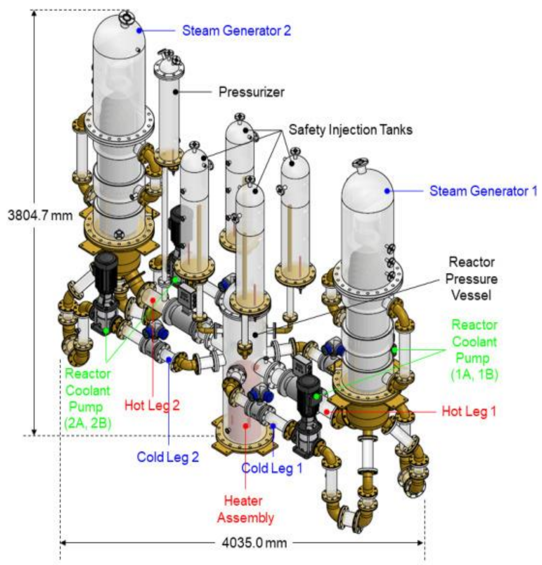

Fig. 1. Design features of URI-LO integral effect facility [2]

2. Experimental Methods

2.1 Design Feature of the HPCR for URI-LO Facility

The control rod is a key component with the function of controlling or shutting down the nuclear reactor power. The two types of control rod designs were considered in the current study. The first one is the conventional control rods design, and the other is the HPCR. Both control rod designs for URI-LO were made with stainless steel SS316L. The outer diameter, thickness, and length of the control rods are 25.4 mm, 1.24 mm, and 2,208 mm respectively. The B4C pellets, neutron absorber inside the control rod were simulated with alumina sleeves (outer diameter of 22 mm and a height of 725 mm).

In the case of the HPCR, the working fluid was injected inside the conventional control rod design to remove the heat by the phase change of the working fluid. The concepts of the HPCR can be found in Fig. 2. The internal pressure of the HPCR is reduced to the 0.1 bar with a vacuum pump to remove the non-condensable gases. The 200 mL of deionized water corresponding to 100 % fill ratio was injected. The length ratio of HPCR is Evaporator : Adiabatic : Condenser = 1.220 m : 0.491 m : 0.5 m. The phase change of the working fluid occurs inside the evaporator section of the HPCR. The generated steam flows upward to the condenser section. Vapor is condensed at the condenser section and makes downward flow to the evaporator. The HPCR can transfer the heat from the core to the heat sink with passive means.

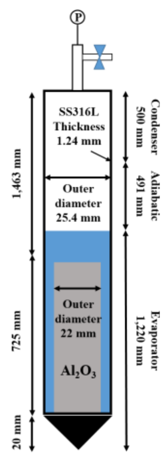

Fig. 2. Schematic of hybrid heat pipe control rod for URI-LO

2.2 URI-LO Experimental Condition

In order to investigate whether HPCRs can improve the safety margin of the nuclear power plant, the SBO accident scenario was simulated by the URI-LO facility with conventional control rods and HPCRs. The main feature of the HPCR is to shut down the reactor core while removing the decay heat from the core. The operation of the HPCR can delay or prevent the exposure of the nuclear fuel into the high-temperature steam environment by removing the decay heat from the core passively [1]. The experimental conditions of the URI-LO were set to simulate the SBO accident after the steam generator dry-out. The initial experimental condition of URI-LO is summarized in Table 1. The decay heat was simulated by manipulating the thermal power level of the embedded cartridge heater in the core region of the URI-LO. The experiment was terminated when the hot leg temperature exceeds 80℃ to ensure the integrity of the URI-LO facility (The main vessel and piping are made of acrylic). The coast-down of the primary coolant under the SBO scenario was simulated by controlling the rotational speed of the coolant pump. The five control rods were considered in the current study. The thermal-hydraulic response of the URI-LO with and without HPCR was experimentally evaluated until the core outlet temperature reach the preset temperature limit.

Table I: Summary of URI-LO SBO experimental conditions

RCS pressure [bar]	1.0 (Water)
Core inlet temperature [°C]	65
Core flow rate [kg/s]	0.8
Core power [kW]	38
Steam generator pressure [bar]	1 bar (FC-72)
Steam generator temperature [°C]	56 (FC-72)
Steam generator heat removal [MW]	0.098

2.3 HPCR CFD analysis conditions

The computational fluid dynamic simulation was conducted with ANSYS Fluent to analyze and characterize the internal two-phase flow phenomena. The computational model for the CFD calculation was made by preserving the geometric characteristics of the HPCR considered in the current study. A total of 222,522 nodes and 212,429 elements were calculated. The denser mesh was applied to the narrow annular gap region formed by the control rod container and B4C pellets, to capture the phase change phenomena in an evaporator section. The initial conditions and boundary conditions were applied identically to the experimental conditions.

3. Results and discussion

In the current study, the URI-LO experiment simulates the SBO accident sequence after the dry out of the steam generator. The comparison of the major thermal-hydraulic parameters including primary coolant flow rate and cladding temperature between experimentally measured data from URI-LO and the MARS-KS calculation was made. The MARS-KS simulation results of URI-LO with conventional control rod are depicted in Fig. 3. The experimental conditions are corresponding to the colored region in figure 3 when the steam generator water inventory is totally dried out (4,000 seconds) [2]. The comparison results of the MARS-KS simulation and the experiment are depicted in Fig. 4. As shown in the figure, the experimental data shows good agreement with MARS-KS simulation with the error of 2.8 - 7 % for the cladding temperature and 0 - 11 % for the core flow rate.

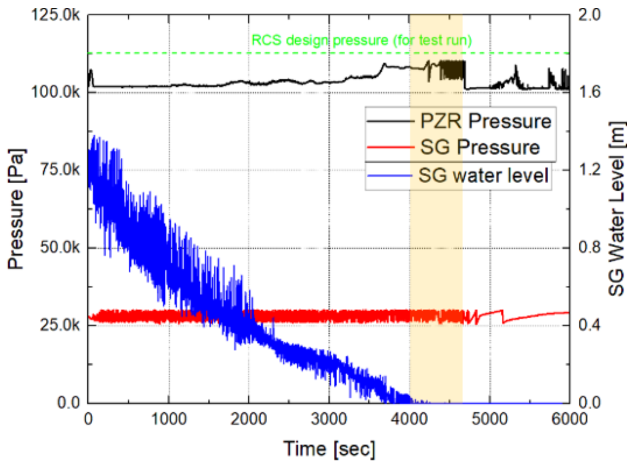

Fig. 3. Simulation of URI-LO SBO accident by MARS-KS [2]

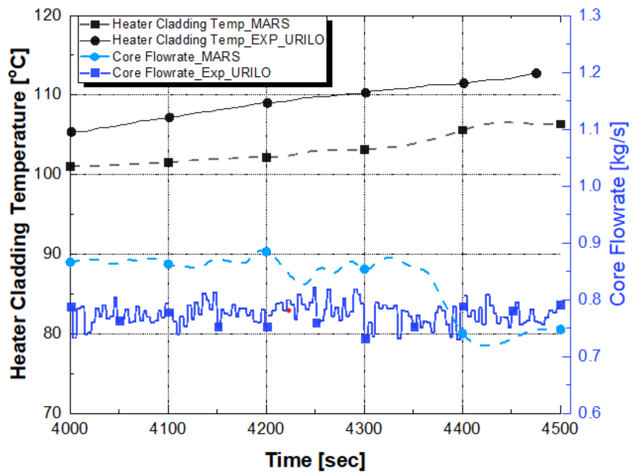

Fig. 4. Comparison results of URI-LO experiment and MARS-KS analysis in SBO accident

3.1 Residual heat removal rate of HPCR

The transient response of the URI-LO with and without HPCR under the SBO scenario was shown in Fig. 5. Both experiments were conducted until the hot leg temperature reached the preset temperature limit (80℃). The primary coolant temperature shows similar behavior for conventional control rod and HPCR cases until 4,200 seconds. However, the times to reach the preset temperature limit are different. The conventional control rods took 4,476 seconds to reach 80℃, while the HPCRs took 4,577 seconds. The HPCRs decrease the heating rate of the primary coolant and delayed the time period to reach the preset temperature limitation (101 seconds) compared to the conventional control rod. The decreasing heating rate of the primary coolant represents that the HPCR can remove the decay heat of the core during the SBO accidents.

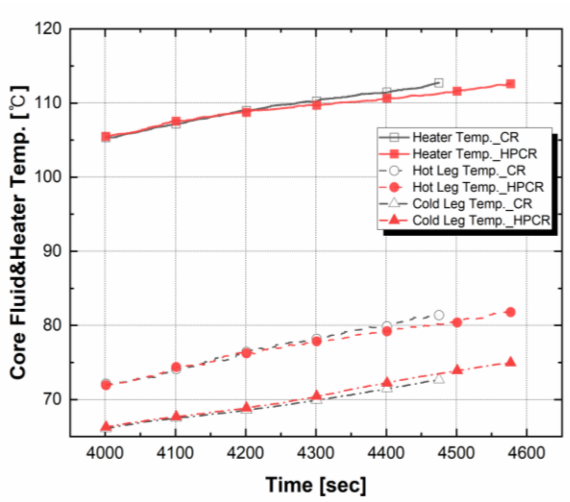

Fig. 5. Comparison results of primary coolant temperature and cladding temperature with and without HPCR.

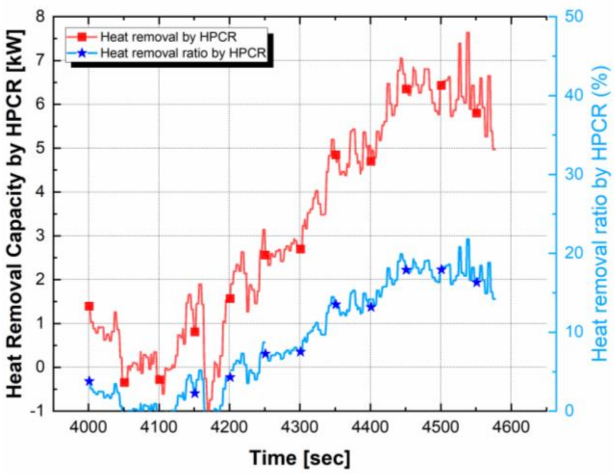

Fig. 6. HPCR core heat removal rate and ratio

The HPCRs heat removal rate and heat removal ratio (relative to the core decay heat) under the URI-LO SBO experiment were shown in Fig. 6. The maximum heat removal rate of the HPCR was 7.5 kW. After 4,450 seconds, the heat removal rate of the HPCR becomes saturated with slight oscillation (the average value of the heat removal rate was 6 kW). Since five HPCRs were considered in the current study, the heat removal rate of each HPCR was ~1.2 kW. The heat removal ratio means the ratio of the HPCRs heat removal rate to the residual heat. After 4,450 seconds, the heat removal ratio reaches approximately 18 %. The operation flooding limits of a single HPCR used in this experiment were calculated and shown in Fig. 7. As shown in the figure, the experimentally measured heat removal rate of the single HPCR (1.2 kW) shows good agreement with the flooding limit correlation for the annular heat pipe proposed by Kim and Bang (1.5 kW) [3].

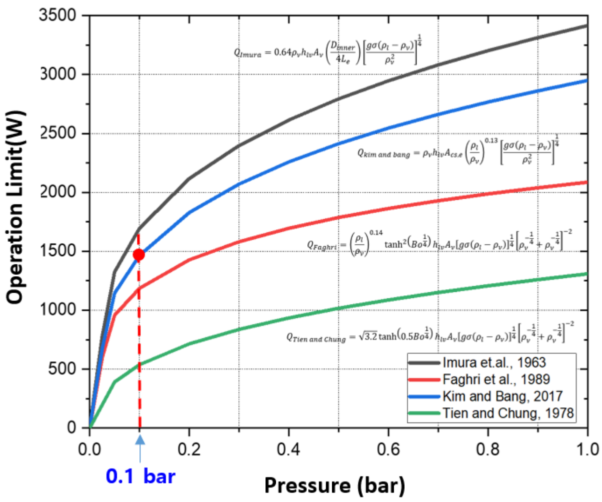

Fig. 7. Operation limits of HPCR [3-6]

3.2 CFD analysis result of a HPCR

In this section, the internal two-phase flow of a single HPCR was analyzed and the vaporization and boiling process of the working fluid was visualized over time 0~2 seconds. The maximum heat removal rate predicted by the CFD simulation was 1.4 kW which shows good agreement with the experimentally evaluated heat removal rate within 20 % error. The boiling of the working fluid was observed in the evaporator section of the HPCR in the current CFD analysis.

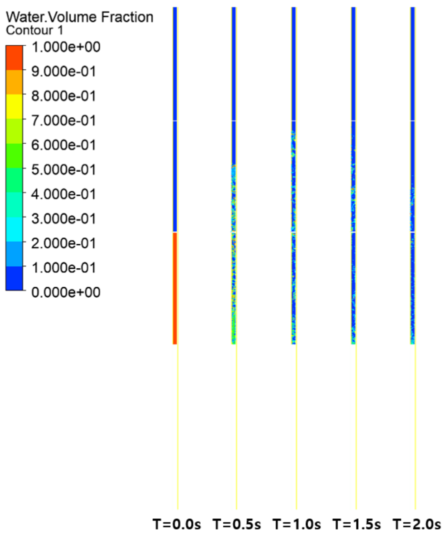

Fig. 8. CFD Results of HPCR

4. Conclusion

In this study, heat removal rate differences between conventional control rods and heat pipe-control rods were experimentally evaluated to analyze the feasibility of the HPCR application to the actual nuclear power plant. The SBO accident was simulated by URI-LO integral effect tests with and without the HPCR. The experimental results show that the HPCR can lower the heating rate of the primary coolant during the SBO accident with 101 seconds delay of the temperature rise for the hot leg temperature to reach the preset limit. The maximum heat removal rate of HPCRs was approximately 7.5 kW, and the average heat removal rate was 6 kW after reaching the steady-state after 4,500 seconds. The CFD analysis results show the presence of boiling of working fluid inside the evaporator section of the HPCR.

The HPCRs are expected to delay core exposure time by about 20% in the actual nuclear power plant SBO accident, contributing to secure golden time. This study is expected to be used as basic data to suggest the validity of the HPCR as a nuclear power plant passive cooling system.

ACKNOWLEDEMENT

This work was supported by the National Research Foundation of Korea(NRF) funded by the Ministry of Science and ICT(NRF-2020M2D6A2093776) and also funded by the Korea government(MSIT) (No.2021M2D2A1A03048950).

REFERENCES

[1] Y. S. Jeong, K. M. Kim, I. G. Kim, and I. C. Bang, Hybrid heat pipe based passive in-core cooling system for advanced nuclear power plant, Applied Thermal Engineering, Vol.90, pp.609–618, 2015.
[2] K. M. Kim and I. C. Bang, Design and operation of the transparent integral effect test facility, URI-LO for nuclear innovation platform, Nuclear Engineering and Technology, Vol.53, pp.776–792, 2021.
[3] K. M. Kim and I. C. Bang, Heat transfer characteristics and operation limit of pressurized hybrid heat pipe for small modular reactors, Applied Thermal Engineering, Vol.112, pp.560–571, 2017.
[4] H. Imura, K. Sasaguchi, H. Kozai, Critical heat flux in a closed two-phase thermosyphon, International Journal of Heat and Mass Transfer, Vol.26, pp.1181–1188, 1983.
[5] A. Faghri, M.M. Chen, M. Morgan, Heat transfer characteristics in two-phase closed conventional and concentric annular thermosyphons, Journal of Heat Transfer, Vol.111, pp. 611–618, 1989.
[6] C.L. Tien, K.S. Chung, Entrainment limits in heat pipes, Proceedings of the 3rd International Heat Pipe Conference, May.22-24, 1978, Palo Allo, California.