Design Analysis of UNIST Reactor Innovation LOop as Thermal-hydraulic Integral Effect Test Facility for Nuclear Innovation

K. M. Kim[a] and I. C. Bang[a]*
[a]*Department of Nuclear Engineering,*
Ulsan National Institute of Science and Technology (UNIST),
50 UNIST-gil, Ulsan, Republic of Korea, 44919
[*]*Corresponding author: icbang@unist.ac.kr*

1. Introduction

Conventional integral effect test facilities have been constructed to observe thermal-hydraulic phenomena and reactor behaviors regarding reactor safety under postulated accident conditions. The height scaling ratios of IET facilities were designed from 1/4 to 1/1 with reference power plants due to the efforts to simulate the system behaviors precisely. Although the facilities contributed to improvement of understanding on thermal-hydraulic phenomena and reactor safety through integral effect tests, application of new technologies and their performance tests have been limited due to cost and large scale of the facilities.

Recently, various nuclear technologies applying 4th industrial revolution technologies such as artificial intelligence [1-3], drone [4], and 3D printing have been studied for improved operation efficiency of reactors. Furthermore, new conceptual passive safety systems are under design for enhanced reactor safety [5-7]. To improve quality and feasibility of the technologies, performance tests with appropriate test facility is necessary. Therefore, a new integral effect test facility, having noticeable scaling ratio, URI-LO (UNIST Reactor Innovation LOop), was designed and constructed. In addition, URI-LO has visualization function by constructing with transparent material. The specialized function will extend the physical insight on two-phase flow inside the reactor system and contribute to public acceptance issue. Hence, URI-LO is expected to be a nuclear power plant innovation platform contributing convergence of various technologies with nuclear technology and public perception. In this paper, the design feature of URI-LO was introduced with preliminary analysis results for design validation.

2. Design feature of URI-LO

URI-LO was designed based on three-level scaling method suggested by Ishii and Kataoka [8] to conserve the system behavior of reference power plant, APR-1400 (1/8 reduced-height and 1/12 reduced-diameter). Richardson number, Time ratio number, and heat source numbers were matched by controlling the pressure loss in pipeline system with orifices, the flow velocity, and solid thickness. Stanton number and Biot number, those are important in terms of conservation of energy, are closely related with heat transfer coefficients. In case of URI-LO, Stanton number and Biot number was distorted due to material compatibility between structural material and coolant. To simulate the system behavior under the high pressure and high temperature condition of reference system in reduced pressure level, the simulant fluid must be utilized as the transparent materials have low mechanical strength. Therefore, flow parameters and degrees of scaling distortions of several refrigerants, FC-72, R-123, R-600a, and R-134a under matched density ratio condition (similar two-phase flow condition) were analyzed. Difference of Prandtl numbers between water and refrigerants induced noticeable distortions of heat transfer coefficients, while flow velocity could be simulated appropriately. Among the refrigerants, FC-72 showed appropriate boiling temperature, compatibility with transparent material, and low degree of heat transfer coefficient distortion. Therefore, FC-72 was selected as simulant fluid of URI-LO.

Table I: Major scaling parameters and scaling ratios for single natural circulation of URI-LO (operate with water) [9]

Parameters	Scaling ratio	URI-LO
Length (height)	l_{oR}	1/8
Diameter	d_{oR}	1/12
Temperature rise	dT_{oR}	1/2
Velocity	$l_{oR}^{1/2}$	1/2.828
Time	$l_{oR}^{1/2}$	1/2.828
Richardson #	1.0	
Friction #	1.0	
Time ratio #	0.84	
Stanton # (laminar)	0.66	
Stanton # (turbulent)	0.50	
Biot # (laminar)	0.80	
Biot # (turbulent)	0.64	
Heat source #	0.72	

The detail of scaling ratios and nondimensional numbers of URI-LO is summarized in TABLE I. Several specific phenomena such as discharge flow through safety valves, flashing, flow instability, countercurrent flow, flow pattern transition, enthalpy change with elevations, and so on are expected to be distorted due to different property changes between water and simulant fluid, and pressure difference between system and environment. The distortions will be analyzed separately with data from reference power plant and large integral effect test facilities, and the analysis results on scaling distortion will contribute to improvement of scaling analysis methodology for highly reduced facilities.

The URI-LO designed by abovementioned process is depicted in Fig. 1. Reactor coolant system includes reactor pressure vessel with downcomer, heater simulating fuel assemblies (maximum 200 kW), pressurizer, 2 hot legs, 4 cold legs, and 4 reactor coolant pumps. Heater diameter was preserved with that of reference model, and height was scaled by 1/8. Number of heaters was further reduced to attenuate the distortion effect of turbulent heat transfer coefficient. Four safety injection tanks (pressurized by nitrogen gas) having direct vessel injection line were installed as fundamental passive safety system. Two steam generators having u-tubes, steam dryer, steam separator, main feedwater to economizer and downcomer with recirculation pipeline were designed.

Excluding the metal parts, such as SG lower plenum, curved pipe lines, u-tubes, all components will be manufactured by transparent materials for the visualizing function as shown in Fig. 2.

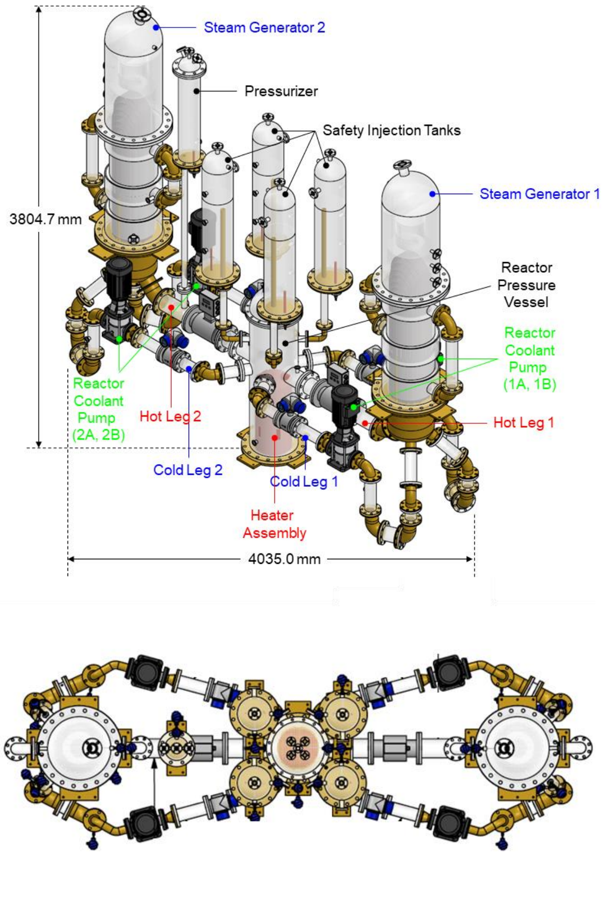

Fig. 1. Drawing of design feature of URI-LO [9]

Instrumentation system consists of 40 thermocouples, 16 pressure transducers, and 12 mass flowmeters to measure temperature, pressure, and flow rate of coolant inside the system. The visualization data regarding thermal-hydraulic phenomena inside the system could provide physical insight on multi-dimensional behaviors related to multiphase such as flow regimes, countercurrent flow, and RCP loop seal clearing.

Fig. 2. Photo of constructed URI-LO [9].

3. Design validation of URI-LO

3.1 Steady state

To confirm the simulatability of the URI-LO regarding the system behavior under transient condition, preliminary analysis with MARS-KS code was conducted. MARS nodalization of URI-LO for preliminary analysis is shown in Fig. 3. Although the facility was designed with operated with simulant fluid, test run condition with water was simulated in this study. Based on scaling analysis results with water, normal operating condition (test run) of URI-LO was determined as presented in TABLE II. The operating condition corresponds to 2% full power operation condition of reference power plant (2% full power considers the decay heat at initial phase of accident scenarios).

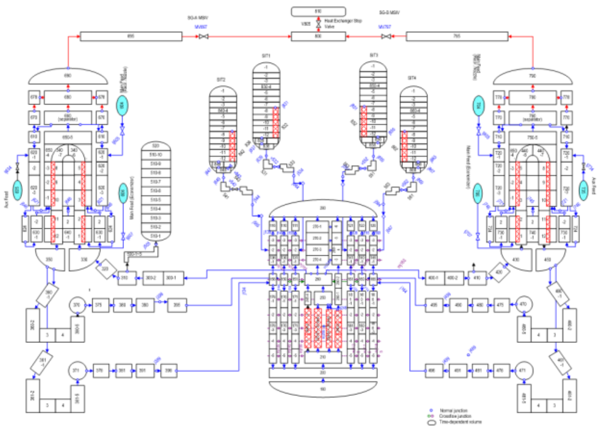

Fig. 3. MARS nodalization of URI-LO [9].

Target normal operating condition was successfully achieved by MARS-KS calculation, therefore, station blackout condition was simulated using the normal operating condition as initial condition to observe the simulatability regarding transient behaviors.

Table II: Determined steady states of URI-LO (test run) and normal operation condition of ARP-1400

Parameters	APR-1400	URI-LO
Primary system pressure [bar]	155.13	1.0
Core inlet temperature [K]	563.7	347.2
Core outlet temperature [K]	598.1	364.4
Core temperature difference [K]	34.4	17.2
Core flow rate [kg/s]	550.6	2.7
Core power [MW]	79.83	0.196
SG pressure [bar]	70.0	0.24
SG steam temperature [K]	564.1	339.2
SG Heat removal rate [MW]	39.92	0.098

3.2 Station blackout

ATLAS SBO-01 test [10] was selected as simulation condition, because ATLAS facility was designed to scale down the APR-1400 and many integral effect tests validated the simulatability of the facility. In this experiment, RCPs and main feedwater pumps were stopped at initiation of accident with simultaneous reactor trip and main steam isolation. The analyzed system behaviors of URI-LO under SBO condition were plotted in Figs. 4~6.

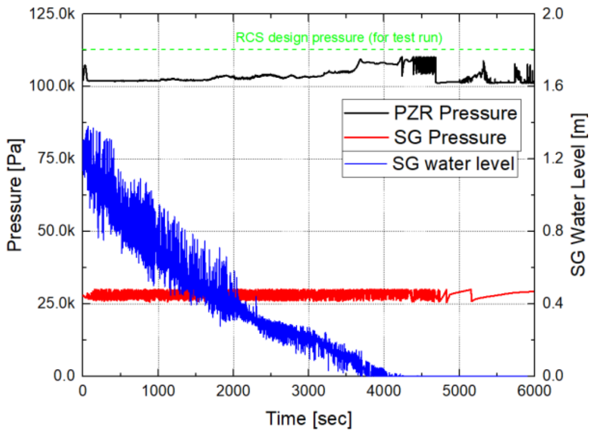

Fig. 4. Variations of system pressures and steam generator water level of URI-LO during simulated SBO condition.

As shown in Fig. 4, isolations of main feedwater and steam induced the operation of MSSVs. Discharge flow through MSSVs induced decrease of steam generator collapsed water level, consequently, inventory of steam generator was depleted. The open/close of MSSVs induced oscillation of SG water level. The reduction of decay heat removal through natural circulation between core and steam generators resulted in increase of RCS pressure. When the RCS pressure reached the POSRV opening setpoint, POSRV operated and pressurizer water level decreased as shown in Fig. 5. During the SBO condition, upper head of reactor pressure vessel (RPV) was saturated and core level was decreased. Then, RPV become a pressurizer, while PZR become a buffer tank. This phenomenon induced time delay between operation of POSRV and PZR water level. At initial phase of accident condition, reactor trip and decay heat removal through steam generators decreased RCS coolant temperature and pressurizer water level. As the steam generator water level decreases significantly, the reactor coolant temperature and pressurizer water level increased as shown in Figs. 5 and 6.

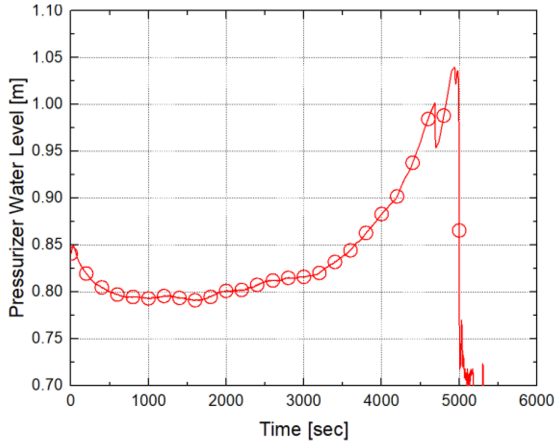

Fig. 5. Variations of pressurizer water level of URI-LO during simulated SBO condition.

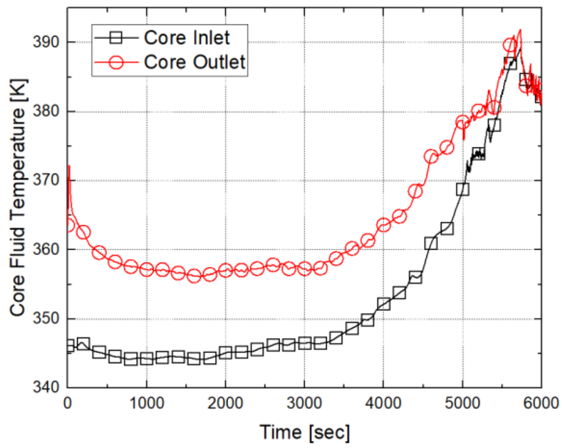

Fig. 6. Change of coolant temperature at core inlet and outlet during simulated SBO condition.

Other system behaviors, such as core water level, RCS flow rate, peak cladding temperature, showed similar behavior with experimental data. Therefore, it was confirmed that transient system behaviors could be simulated appropriately with the constructed facility. The quantitative design validation will be conducted after integral effect tests with the facility.

4. Conclusions

UNIST reactor innovation loop, URI-LO, which is a compact integral effect test facility was designed and constructed as a test bed for newly suggested technologies and education tool. The facility was designed by three level scaling method to have scaling ratios of 1/8 height ratio and 1/12 diameter ratio of APR-1400. The performance of URI-LO regarding transient condition (ATLAS SBO-01 test) was analyzed by MARS-KS code simulation. Analysis results indicated that URI-LO can appropriately simulate the system behaviors under station blackout condition. The integral effect tests will be conducted in the future, and quantitative design validation will be carried out with utilization of refrigerant. The research activities of URI-LO is expected to contribute to provision of visualization data related to thermal-hydraulic phenomena, performance test for newly suggested technologies, and improvement of public perception on nuclear power plant.

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