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# Chemical Process Design

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## Modeling of Forced and Natural Circulation Test Loop for a Pressurized Water Reactor using Thermal-Hydraulic Method

Iman Khonsha<sup>1</sup>

<sup>1</sup>Corresponding Author, Department of Chemical Engineering, Shi.C., Islamic Azad University, Shiraz, Postcode: 71993-1, Iran.  
Email: [iman.khonsha@iau.ac.ir](mailto:iman.khonsha@iau.ac.ir)

### ARTICLE INFO

**Article type:**  
*Research Article*

**Article history:**  
Received: 2025-07-14  
Received in revised form: 2025-12-07  
Accepted: 2025-12-11  
Available online: 2025-12-11

**Keywords:** Modeling; Forced and Natural Circulation; Test Loop; Pressurized Water Reactor; RELAP5/MOD3.2

### ABSTRACT

The present paper studies an Integral Test Facility (ITF) using the thermal-hydraulic method in order to determine its safety and reliability. The scaling and thermal-hydraulic design of the Islamic Azad University Test Loop (IAUTL) is conducted, accompanied by the energy scaling methodology for a typical pressurized water reactor (PWR). The scaling findings are authenticated by the experimental data resulting from the pressurized water reactor for pure water and forced circulation (FC) flow in 100% of core nominal power using the RELAP5/MOD3.2 code. From the comparison of the simulation results of core temperature in natural and forced flow, it can be concluded that the maximum outlet temperature of the reactor core in natural and forced flow is 169.4 °C and 174 °C, respectively. This temperature increase in natural flow is due to the increase in the residence time of the fluid in the natural flow regime around the core. Also, the temperature drop in the heat exchanger in natural and forced flow is 20 °C and 13 °C, respectively. This increase in heat loss in natural flow is due to the lower velocity of the natural flow and, as a result, the heat exchange also increases in natural flow. The pressure drop between the inlet and outlet of the reactor core, in both free and forced flow, is approximately 2 bar. The simulation results indicate that the IAUTL core can remove the residual heat from the core using NC flow.

**Cite this article:** Khonsha, I., (2026), Modeling of Forced and Natural Circulation Test Loop for a Pressurized Water Reactor using Thermal-Hydraulic Method, *Chemical Process Design*, 5(1), 96-110. <http://doi.org/10.22111/cpd.2025.52615.1067>



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DOI: <http://doi.org/10.22111/cpd.2025.52615.1067>

Publisher: University of Sistan and Baluchestan

### 1. Introduction

Natural circulation (NC) is regarded as an outstanding mechanism in several industrial systems and it is important to get acquainted with its behavior for design, operation and safety of the nuclear reactor [1, 2]. For designing various nuclear reactors and passive safety systems, the natural circulation flow-driven systems have been applied. System thermal-hydraulics analysis codes are commonly utilized for analyzing the normal operations and safety of the nuclear

reactor [3]. Single-Phase Natural Circulation (SPNC) regime denotes no void occurrence in the upper plenum of the system. Consequently, the coolant at the core outlet will be sub cooled up to nearly saturated. Core flow rate has its origin in the balance between driving and resistant forces. The driving forces are created by fluid density variations occurring between {descending side of U-tubes and vessel down comer} and {core and ascending side of U-tubes}. While irreversible friction pressure drops along the entire loop gives rise to the resistant forces. Resulting fluid velocities are adequate for removing the core power in (sub cooled) nucleate boiling or forced convection heat transfer regimes: no film boiling condition is experienced in the core. SPNC may come to pass at any primary system pressure, congruently with SG pressure. Nevertheless, typical primary system pressures range between 8 and 16 MPa while the secondary pressure being near to the nominal operating condition. From the Nuclear Power Plant (NPP) design, it is expected that the nuclear decay heat can be removed from the core by SPNC, provided that SG cooling is available [4].

In the nuclear technology, this fact that the novel concepts largely exploit the gravity forces for the heat removal capability is especially true [5]. At this moment, the removal capability of the NC core power is only utilized for accident situations, principally for illustrating the inherent safety features of the plants [6]. Burgazzi et al. propounded the reliability evaluation for passive thermal-hydraulic systems [7]. NC phenomenology, including applications, stability studies, available databases and investigation of predictive capabilities of the computational tools was put forward by D'Auria et al. [8]. Mousavian et al. demonstrated the decay heat removing from the reactor in the event of station blackout with the intact primary and secondary circuits, and also proved the depressurization of Reactor Coolant System (RCS) under a small break LOCA [9]. Cherubini, et al. inspected a natural circulation test in the PSB-VVER test facility with RELAP5 code [10]. Gou et al. assessed the thermal hydraulic analysis of a Passive Residual Heat Removal System (PRHRS) for an integral pressurized water reactor [11].

Shanbin et al. got the measure of RELAP5/MOD3.2 for startup transients in a natural circulation test facility [3]. Plenty of researches have been carried out concerning scaling and thermal-hydraulic design. For instance, Liu et al. developed a power-to-mass scaling methodology for the reduced height and reduced pressure conditions, and they studied station blackout accident conditions for understanding reliability of this method [12]. Liu and Lee expanded upon a complete scheme of scaling procedures of designing the integral system test facilities with the reduced height and reduced pressure [13]. Jafari et al. conducted the code-based design and stability analysis of a test loop (TTL-1) [14]. Bae et al. developed the energy scaling methodology for the reduced height and reduced pressure conditions, and it including the effect of a density difference in the reduced pressure for calculating the thermal power in the test facility [15].

D'Auria and Galassi inspected the relevance of scaling in the water-cooled nuclear reactor technology. They distinguished the dimensionless design factors for ITF from the scaling factors [16]. In a new study, the behavior of Emergency Core Cooling System (ECCS) in an Advanced Pressurized Reactor 1400 (APR) during the early stage of Loss of Coolant Accident (LOCA), was analyzed by using Flownex software [17]. In another study, the modified one-dimensional thermal hydraulic program RELAP5/MOD3.3 was developed to simulate the natural circulation characteristics of an integrated reactor [18]. In another research, CFD calculations are presented for the three types of water-based nanofluids:  $Al_2O_3$ /water, CuO/water, and  $TiO_2$ /water with 0.1% volume fraction. These calculations are performed using ANSYS-CFX, and as the geometry, the scaled-down SRBTL test loop for a VVER-1000 reactor core design is utilized [19].

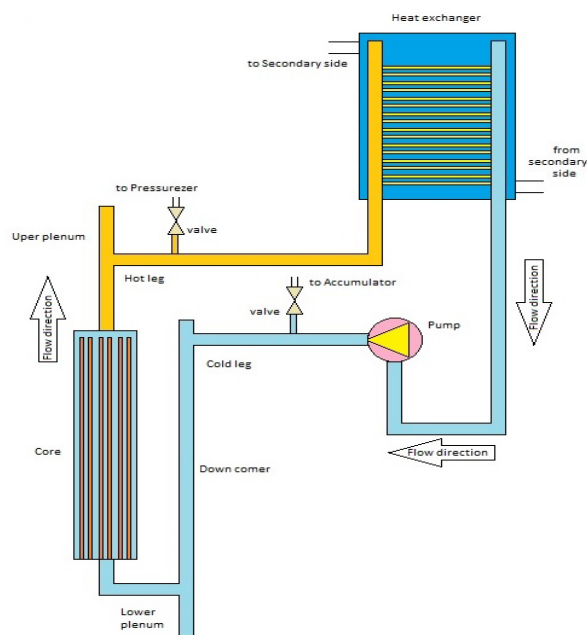
Also, various researches have been conducted recently, including on the factors affecting the selection of heat exchanger tubes [20], the causes of damage due to scale formation in boiler tubes [21], statistical analysis of the effects of aluminum oxide nanoparticles on the storage capacity of carbon dioxide hydrate formation [22], the use of nanoparticles as additives in lubricants [23], and the removal of copper (II) ions from wastewater by carbon nanotubes modified with tetrahydrofuran [24].

The present research aims at scaling down, performing the thermal-hydraulic design of a test loop, and evaluating the test loop core heat removal for the FC and NC fluid flow. The innovation of the present research is the use of an advanced methodology called energy scaling methodology for the conceptual design of a new innovative test loop. The design and validation with experimental data of a real-scale pressurized reactor based on forced convection flow is performed using the Reloop 5 code. In addition, in this research, modeling of natural convection flow in emergency conditions was investigated as an innovation. Because in events leading to reactor shutdown, which occurs with the loss of off-site power supply, other internal electrical power sources, or pump failure, the natural convection flow of the coolant can remove the residual heat caused by the decay of fission products and prevent core meltdown. Therefore, scaling of natural convection flow in test loop heat exchangers that have the ability to remove residual heat in the reactor core is mandatory. Also, the results of natural convection modeling and its validation will be another emphasis on correct scaling.

## 2. Materials and methods

### 2.1. IAUTL test loop description

IAUTL is designed for simulating the thermal-hydraulic behaviors of PWR, which involves all elements in the primary circuit of PWR. IAUTL has one circuit with the most important components in the PWR and the primary circuit, including core, downcomer, upper plenum, lower plenum, heat exchanger, pump, accumulator and pressurizer (Fig. 1).



**Fig.1.** Schematic view of IAUTL test loop

The reactor core, which is the main component of the IAUTL test facility, is shown in Fig. 2. As a new design, the IAUTL core consists of 24 heating elements (nuclear fuel rods) and a central guide element, which have the same diameter and pin pitch as the elements in the PWR core.

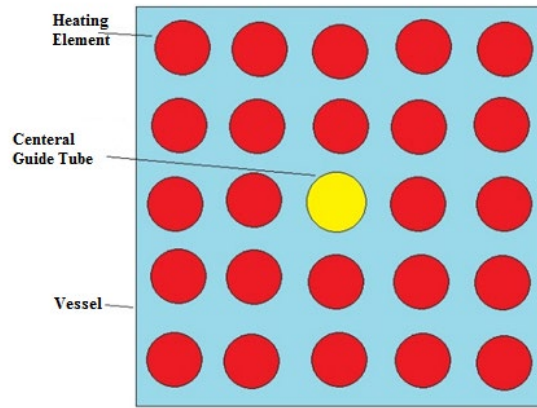


Fig. 2. IAUTL core cross-section

## 2.2. Scaling methodology

All sizes and dimensions of equipment and pipelines including height, length, diameter, area, volume, as well as the thermal-hydraulic properties excluding the diameter and pin pitch of heating elements are reduced using the energy scaling methodology. IAUTL is designed as a reduced-height, reduced-pressure and reduced-temperature ITF, and the energy scaling methodology is exploited for scaling down the coolant mass inventory and the thermal power.

The modeling assumptions are: The modeling for fluid flow is fully developed and single-phase with forced fluid circulation using a pump. The coolant for the reactor core is desalinated light water, and the energy scaling methodology has been employed for scaling calculations. It is assumed that energy loss from the walls of the test loop pipes is zero, and the walls are completely insulated. To create heat and simulate the test loop core, heating elements are used, and heat is extracted using a heat exchanger. In addition to the fluid circulation pump in this test loop, equipment such as a pressurizer and a tank for emergency injection in case of an accident, and an Emergency Core Cooling System (ECCS) are also included.

On account of the significant difference in the scaling factor for the system pressure, the enthalpy difference in accordance with a reduced pressure condition should not be disregarded. In the energy scaling methodology,  $(l_{OR})$  and  $(a_{OR})$  are defined as the scaling factors of length and area; hence, the scaling factor of volume  $(V_{OR})$  is multiplication of these two quantities. The initial value of mass inventory  $(M_0)$  is defined as follows [25, 26]:

$$M_0 \cong \rho_{avg} V = \rho(P_P, T_{avg}) V \quad (1)$$

and  $(T_{avg})$  is:

$$T_{avg} = \frac{T_{HL} + T_{CL}}{2} \quad (2)$$

where  $(T_{HL})$  is the hot-leg fluid temperature and  $(T_{CL})$  is the initial cold-leg fluid temperature.

$$M^* = \frac{M}{M_0} \quad (3)$$

$$h^* = \frac{h}{h_0} \quad (4)$$

$$t^* = \frac{t}{t_0} \quad (5)$$

The enthalpy difference between the cold leg and hot leg ( $\Delta h_0$ ) is defined:

$$\Delta h_0 = h_{HL} - h_{CL} \quad (6)$$

$h_{HL}$  and  $h_{CL}$  are the local enthalpy of the hot leg and cold leg.

$$\tau = \frac{M_0 \Delta h_0}{Q_{c0}} \quad (7)$$

$$\frac{dM^*}{dt} = \frac{\tau \dot{m}_{in}}{M_0} + \frac{\tau \dot{m}_{out}}{M_0} \quad (8)$$

$$\frac{dM^* h^*}{dt^*} = \frac{\tau Q_c}{M_0 \Delta h_0} + \frac{\tau \dot{m}_{in} h_{in}}{M_0 \Delta h_0} + \frac{\tau \dot{m}_{out} h_{out}}{M_0 \Delta h_0} \quad (9)$$

In the above equations, there is flexibility in determining a time scale,  $\tau$ . For a real-time simulation, the prototype and the facility employ the same time scale, *i.e.*,  $\tau_R = 1$ . As a consequence, from in line with Eq. (11) and Eq. (12), the scaling criteria for conserving the non-dimensional mass inventory ( $M^*$ ) and enthalpy ( $h^*$ ) are defined as follows:

$$1 = \left( \frac{\dot{m}_{in}}{M_0} \right)_R \quad (10)$$

$$1 = \left( \frac{\dot{m}_{out}}{M_0} \right)_R \quad (11)$$

$$1 = \left( \frac{Q_c}{M_0 \Delta h_0} \right)_R \quad (12)$$

$$1 = \left( \frac{h_{in}}{\Delta h_0} \right)_R \quad (13)$$

$$1 = \left( \frac{h_{out}}{\Delta h_0} \right)_R \quad (14)$$

The thermal power in the core ( $Q_c$ ) should be scaled by the ratio of the total energy ( $M_0 \Delta h_0$ ). For a real-time simulation, the prototype and the facility exploit the same time scale, *i.e.*,  $\tau_R = 1$ .

$$Q_{c0} = \frac{M_0 \Delta h_0}{\tau} \quad (15)$$

$$Q_{c0R} = \frac{(M_0 \Delta h_0)_R}{\tau_R} \quad (16)$$

The initial cold-leg fluid temperature ( $T_{CL}$ ) was determined with by consideration taking into account the heat transfer between the primary system and the secondary system in the steam generators as follows:

$$(T_{CL} - T_{sat}(P_s))_R = 1 \quad (17)$$

where  $\tau_R = 1$ , then thermal power scaling factor in the core is defined as follows:

$$Q_{c0R} = M_{0R} \Delta h_{0R} = \rho_R l_{0R} a_{0R} \Delta h_{0R} \quad (18)$$

Other components of the energy scaling factors are clarified as follows:

$$l_{0R} = \frac{(l_{comp})_{Model}}{(l_{comp})_{Prototype}} \quad (19)$$

$$a_{0R} = \frac{(a_{comp})_{Model}}{(a_{comp})_{Prototype}} \quad (20)$$

$$V_{0R} = \frac{(V_{comp})_{Model}}{(V_{comp})_{Prototype}} \quad (21)$$

So, the density scaling factor and the mass inventory scaling factor are as follows:

$$\rho_R = \frac{(\rho_{avg})_{Model}}{(\rho_{avg})_{Prototype}} = \frac{\rho(P_P, T_{avg})_{Model}}{\rho(P_P, T_{avg})_{Prototype}} \quad (22)$$

$$M_{0R} = \frac{(M_0)_{Model}}{(M_0)_{Prototype}} \quad (23)$$

The enthalpy scaling factor is:

$$\Delta h_{0R} = \frac{(\Delta h_0)_{Model}}{(\Delta h_0)_{Prototype}} \quad (24)$$

The last dimensionless number is pressure scaling factor:

$$P_{0R} = \frac{(P_P)_{Model}}{(P_P)_{Prototype}} \quad (25)$$

### 2.3. Scaling and thermal hydraulic design computations

IAUTL scaling factors are calculated using the foretasted equations. In Table 1, the energy scaling factors are obtained by considering calculations and relations 1 to 25, calculating each effective parameter in the thermal power of the reactor core, and multiplying the resulting dimensionless parameters with each other. This methodology employs both the conservation of fluid mass and the conservation of fluid energy principles. The IAUTL Scaling factors, PWR experimental data and the results of IAUTL scaling for FC fluid flow, are stated in Table 2 and Table 3. The effective parameters of each component and fluid are inserted in Pressure Drop Online-Calculator software for calculating the pressure drop of IAUTL [27]. The calculation results of IAUTL pressure drop are listed in Table 4, and they are compared with the same ones in PWR.

**Table 1.** Energy scaling factors and IAUTL scaling factors for FC fluid flow

Parameter	Energy scaling factor [15]	IAUTL scaling factor
Length	$l_{0R}$	1:5.87
area	$a_{0R}$	1:2137
density	$\rho_R$	1:0.785
Temperature	$T_{0R}$	1:1.886
Pressure	$P_{0R}$	1:15.9
Enthalpy	$\Delta h_{0R}$	1:2.42
Volume	$a_{0R}l_{0R}$	1:12569
Mass inventory	$\rho_R a_{0R} l_{0R}$	1:9929
Core thermal power	$\rho_R a_{0R} l_{0R} \Delta h_{0R}$	1:23831
Flow rate	$\rho_R a_{0R} l_{0R}$	1:9929
Time scale	1	1:1

### 2.4. Scaling validation

For the scaling validation method, a steady state condition of 100% nominal reactor power is considered for PWR. For comparing IAUTL scaling results and PWR experimental data, four Dimensionless Parameters ( $DP$ ) are defined according to the following formula:

**Table 2.** The scaling results of IAUTL core for FC fluid flow

Core parameters	IAUTL Scaling factors	PWR [25,26]	IAUTL (Scaling results)
Core height (m)	1:5.87	3.53	0.600
Volume of core (m <sup>3</sup> )	1:12569	15.3	12.17e-4
Generated heat in the core (kw)	1:23831	3.0e+6	126
Average generated power of each heating element/FE (kw)	-----	59	7
Flow rate in the core (m <sup>3</sup> /h)	1:9929	84800	8.54

Pressure in the core inlet (bar)	1:15.9	160.81	10
Pressure in the core outlet (bar)	1:15.9	157	9.44
Fluid density in the inlet and outlet of the core (kg/m <sup>3</sup> )	1:0.785	743-675	914- 896
Average fluid density in the core (kg/m <sup>3</sup> )	1:0.785	709	905
Average temperature in the core (°C)	1:1.886	306	161.9
Enthalpy in the inlet and outlet of the core (kJ/kg)	-----	1284-1452	649-719
Enthalpy difference between inlet and outlet of the core (kJ/kg)	1:2.42	168	70
Temperature in the inlet and outlet of the core (°C)	1:1.886	291-321	153.96-169.84
Temperature difference between inlet and outlet of the core (°C)	1:1.886	30	15.88
Height of heating element/FE (m)	1:5.87	3.53	0.60
Saturated temperature in the outlet of the core (°C)	1:1.886	346	183

**Table 3.** The scaling results of IAUTL primary circuit for FC fluid flow

Primary circuit parameter	IAUTL Scaling factor	PWR [25,26]	IAUTL (Scaling results)
Height of test loop until U-tubes of HE/SG (m)	1:5.87	15.80	2.69
Cold leg volume (m <sup>3</sup> )	1:12569	4×(15.8)	50.27e-4
Hot leg volume (m <sup>3</sup> )	1:12569	4×(7.4)	23.55e-4
Volume of U-tubes of HE/SG (m <sup>3</sup> )	1:12569	4×(16.2)	51.54e-4
Height of U-tubes of HE/SG (m)	1: 5.87	2.670	0.454
Area of U-tubes of HE/ SG (m <sup>2</sup> )	1:2137	6115	2.86
Volume of downcomer (m <sup>3</sup> )	1:12571	19.2	15.27e-4
Height of downcomer (m)	1: 5.87	7.08	1.204
Diameter of downcomer (m)	1: 5.87	0.263	0.0447
Volume of upper plenum (m <sup>3</sup> )	1:12569	59.9	47.65e-4
Volume in lower plenum (m <sup>3</sup> )	1:12569	13.3	10.58e-4
Volume of pump/RCP (m <sup>3</sup> )	1:12569	4×(3)	9.55e-4
Flow rate of pump/RCP (m <sup>3</sup> /h)	1:9929	4×(21200)	8.54
Height of PRZ (m)	1: 5.87	13.235	2.25
Diameter of PRZ (m)	1: 5.87	3.330	0.57
Volume of ACC (m <sup>3</sup> )	1:12569	240	19.0
Water volume in the ACC (m <sup>3</sup> )	1:12569	200	15.7
Inlet temperature of the HE in secondary loop (°C)	1:1.886	220	116.0
Pressure of the HE in secondary loop (bar)	1:15.9	61	3.81

**Table 4.** Pressure drop comparison between PWR and IAUTL elements

Parameter	PWR[25,26]	IAUTL(Scaling results)
Core pressure drop (bar)	3.81	0.56
HE/SG pressure drop (bar)	1.35	0.33
Pipelines pressure drop (bar)	1.08	0.47
Sum of the all components pressure drop (bar)	6.24	1.36
Head of Pump/RCP(bar)	6.24	1.36

$$X_{DP} = \frac{X_{FC}}{(X_{FC})_{avg}} \quad (26)$$

where  $X$  can be pressure ( $P$ ), enthalpy ( $h$ ), dynamic viscosity ( $\mu$ ) or density ( $\rho$ ). The thermal-hydraulic parameters of PWR reactor core and IAUTL core such as inlet, outlet and average value are listed in Table 5.

**Table 5.** Four thermal-hydraulic parameters of PWR core and IAUTL core for FC fluid flow

Core parameters	PWR [25,26]	IAUTL (scaling results)
Pressure in the inlet (bar)	160.81	10

Pressure in the outlet (bar)	157	9.44
Average pressure (bar)	158.905	9.72
Fluid density in the inlet (kg/m <sup>3</sup> )	743	914
Fluid density in the outlet (kg/m <sup>3</sup> )	675	896
Average fluid density (kg/m <sup>3</sup> )	716	906
Enthalpy in the inlet (kJ/kg)	1284	649
Enthalpy in the outlet (kJ/kg)	1452	719
Average enthalpy (kJ/kg)	1368	684
Dynamic viscosity in the inlet (Pa.s)	9.2220e-5	17.7540e-5
Dynamic viscosity in the outlet (Pa.s)	8.0182e-5	15.9750e-5
Average Dynamic viscosity (Pa.s)	8.6201e-5	16.8600e-5

Table 6 presented the comparison of dimensionless thermal-hydraulic parameters between PWR reactor core and IAUTL core for FC fluid flow. Due to the acceptable variations and small errors of the comparison, it can be come to the conclusion that the scaling down of IAUTL test loop is valid and comparing the results of RELAP 5 simulation with the scaling down of IAUTL test loop is acceptable.

**Table 6.** Comparison of dimensionless thermal-hydraulic parameters between PWR reactor core and IAUTL core for FC fluid flow

Dimensionless parameter	PWR [25,26]	IAUTL (Scaling results)	Error%
$P_{DP}$ for core inlet	1.0120	1.0290	1.68
$P_{DP}$ for core outlet	0.9880	0.9710	1.72
$h_{DP}$ for core inlet	0.9383	0.9496	1.08
$h_{DP}$ for core outlet	1.0620	1.0500	0.96
$\mu_{DP}$ for core inlet	1.0698	1.0530	1.68
$\mu_{DP}$ for core outlet	0.9302	0.9475	1.86
$\rho_{DP}$ for core inlet	1.0479	1.0099	3.79
$\rho_{DP}$ for core outlet	0.9521	0.9900	3.98

### 2.5. Code input model and Nodalization of IAUTL

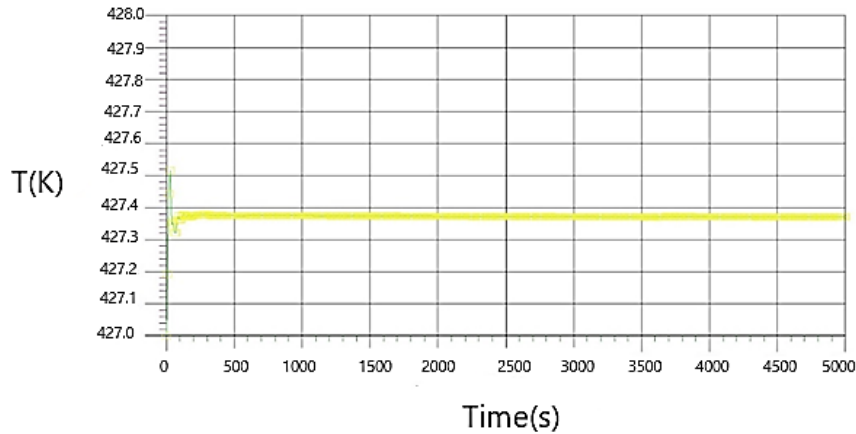
Thermal-hydraulic analysis of IAUTL is performed using the RELAP5/MOD3.2 code. RELAP5 was developed at the Idaho National Engineering Laboratory (INEL) for the US Nuclear Regulatory Commission (NRC). A generic modeling approach is utilized, which permits simulating a variety of thermal hydraulic systems. The control system and secondary system components are included for enabling the modeling of plant controls, turbines, condensers, and secondary feedwater systems. The general solution procedure subdivides the system into a number of control volumes connected by flow paths. The code includes many generic component models from which general systems can be simulated. The component models comprise pumps, valves, pipes, heat structures, reactor point kinetics, separators, control system components, etc. Heat structures are supposed to be represented in rectangular or cylindrical geometry by one-dimensional heat conduction.

For converting the unit surface of one-dimensional calculation to the actual surface of heat structure Surface multipliers are employed. A finite difference method is used to achieve the heat conduction solutions. Each mesh interval includes different mesh spacing, various materials or both [28]. In this modeling, all major components of the primary side including core and primary circuit and that of the secondary side are nodalized and simulated using RELAP5 code. In this modeling, the scaled down parameters and dimensions are utilized to clarify the flow areas, volumes, hydraulic diameters, elevations, heat transfer area and heat structure masses. On the primary side, the reactor core, downcomer, upper plenum and lower plenum, pumps, main circulation pipes, and the pressurizer have been modelled. On the primary and secondary sides, a scaled down vertical Heat Exchanger (HE) is also nodalized and simulated. In the primary side, HE includes six equivalent branch pipes.

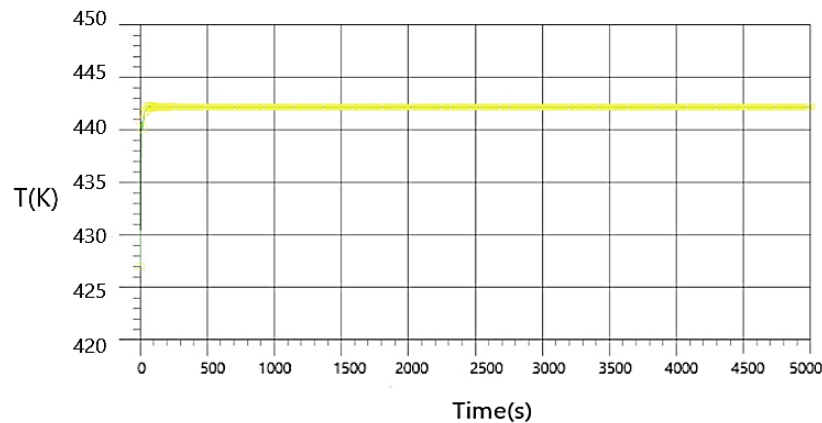
### 3. Results and discussions

#### 3.1. Core evaluation with RELAP5 code for FC fluid flow

The temperature variation results using RELAP5 code for FC fluid flow in the inlet and outlet of the IAUTL core are shown in Figs. 3 and 4.



**Fig. 3.** Temperature variation diagram vs. time in the inlet of the IAUTL core for FC fluid flow



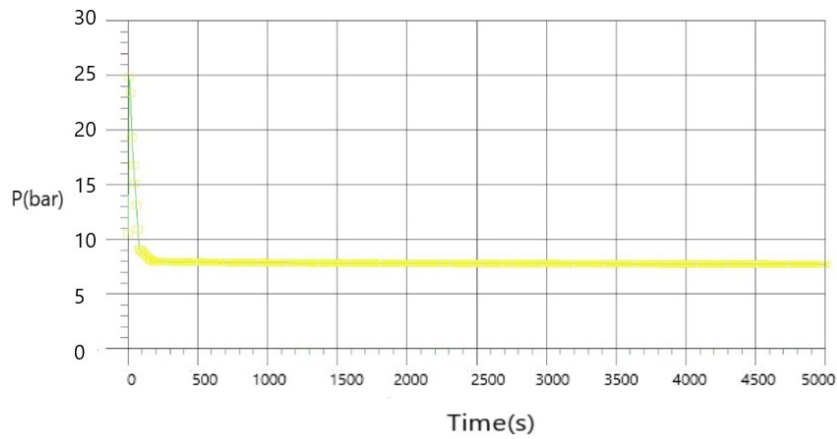
**Fig. 4.** Temperature variation diagram vs. time in the outlet of the IAUTL core for FC fluid flow

Table 7 represents the comparison of IAUTL core temperature variation between scaling and simulation results for FC fluid flow. The maximum temperature in the core inlet is 154.28°C, the maximum temperature in the core outlet is 169.4°C, the maximum temperature difference between the core inlet and outlet is 15.12°C, and the average temperature in the core is 161.84°C from the RELAP5/MOD3.2 code. The core pressure variations for the FC fluid flow are shown in the Fig. 5 and the results are given in Table 10.

As it is evident from Figs. 3 and 5, the simulation diagrams are almost constant or with little changes after about the first 200 seconds of the simulation, for this reason that the steady state condition is happened after about 200 seconds.

**Table 7.** Comparison of IAUTL core temperature variation between scaling and simulation results for FC fluid flow

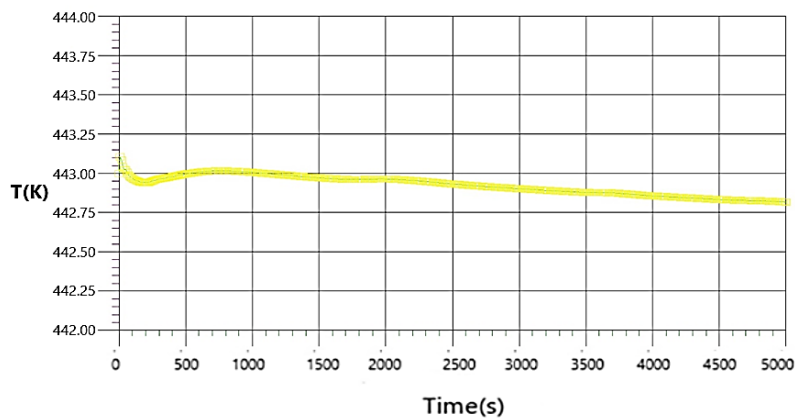
Core temperature(°C)	IAUTL (Scaling results)	IAUTL (RELAP5), Value	IAUTL (RELAP5), Error%
Inlet maximum temperature	153.96	154.28	2.08
Outlet Maximum temperature	169.84	169.40	2.59
Maximum temperature difference	15.88	15.12	4.79
Average temperature	161.90	161.84	0.04



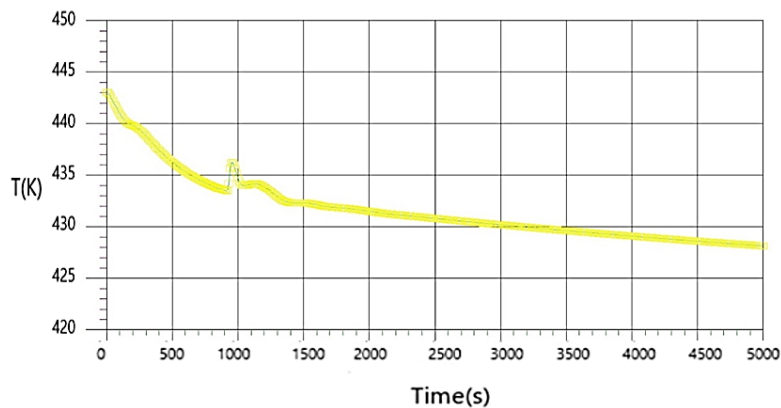
**Fig. 5.** Pressure variation diagram vs. time in the outlet of the IAUTL core for FC fluid flow

### 3.2. HE evaluation using RELAP5 code for FC fluid flow

The simulation results in the inlet and outlet of HE for FC fluid flow are revealed in Figs. 6 and 7. It is possible to utilize the Fig. 6 diagram for calculating the average value, since it changes almost linearly after about the first 500 seconds of the simulation. This is attributable to the steady state condition after beginning of the simulation.



**Fig. 6.** IAUTL temperature variation diagram vs. time in the inlet of the HE (Pipe#190) for FC fluid flow



**Fig. 7.** IAUTL temperature variation diagram vs. time in the outlet of the HE (Pipe#190) for FC fluid flow  
For HE temperature outlet in Fig. 7, after about the first 1500 seconds of the simulation, variation of the diagram is almost linearly or with a little change, and the average value can be calculated by employing this part. This is due to the steady state condition after starting the simulation and initial input temperatures in RELAP5 cards. Likewise, HE output temperature simulation results from RELAP5 are very close to the scaling results. The HE temperature comparison between scaling and simulation results for FC fluid flow is shown in Table 8. As it is conspicuous from

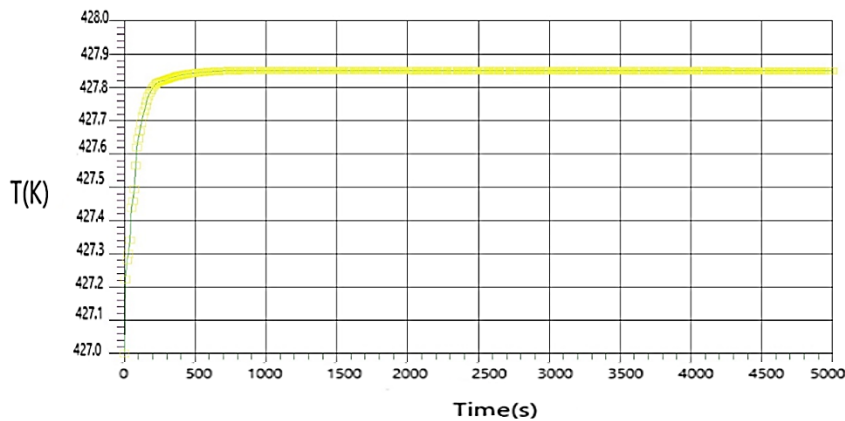
Table 8, the HE average inlet and outlet temperature in the RELAP5 simulation results is very close to the scaling results.

**Table 8.** Comparison of HE temperature between scaling and simulation results for FC fluid flow

HE Parameter	IAUTL (Scaling results)	IAUTL (RELAP5), Value	IAUTL (RELAP5), Error%
Average outlet temperature after the 1500th second(°C)			
For Pipe#160	153.96	155.00	0.676
For Pipe#170	153.96	155.00	0.676
For Pipe#180	153.96	157.00	1.974
For Pipe#190	153.96	157.00	1.974
Average inlet temperature after the 500th second (°C)			
For Pipe#160	169.84	169.00	0.494
For Pipe#170	169.84	169.30	0.318
For Pipe#180	169.84	169.85	0.006
For Pipe#190	169.84	169.80	0.024

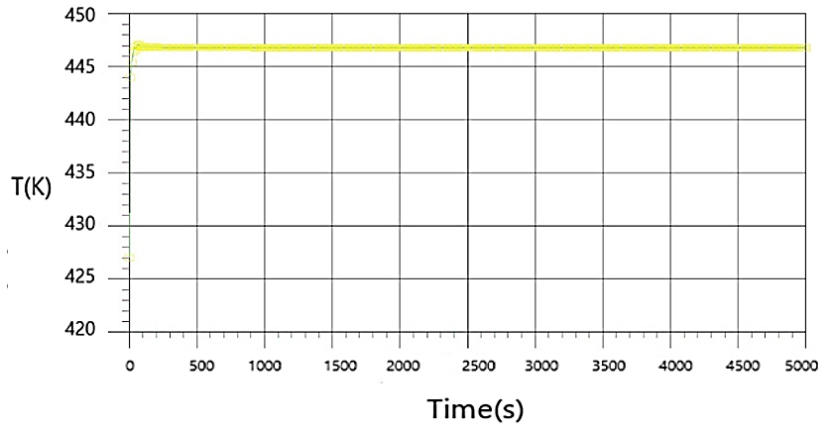
### 3.3. Core evaluation with RELAP5 code for NC fluid flow

Thermal-hydraulic analysis is carried out using the RELAP5/MOD3.2 code so as to conduct an investigation into the NC heat removal capability of IAUTL core. The effective major features including hot leg, Pressurizer, heat exchanger, cold leg, downcomer, lower plenum, core and upper plenum are nodalized and simulated using RELAP5/Mod3.2, and the pump is considered as a branch. Temperature variations in the IAUTL core for NC fluid flow are represented in Figs. 8 and 9. The core pressure variations for the NC fluid flow are shown in the Fig. 10 and the results are given in Table 10.

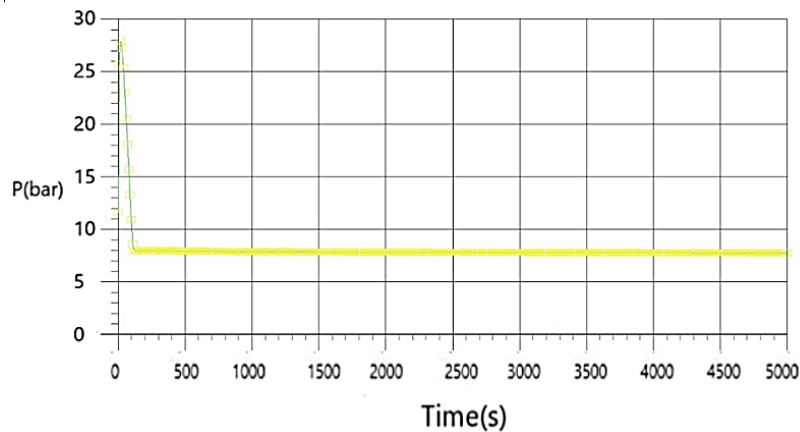


**Fig. 8.** Temperature variation diagram vs. time in the inlet of the IAUTL core for NC fluid flow

As can be seen from Figs. 8 and 9, the simulation diagrams are almost constant after about the first 200 seconds of the simulation, and in such a case, it is similar to the FC fluid flow. The Comparison of IAUTL core and HE temperature from RELAP5 simulation results for NC fluid flow are presented in Table 9.



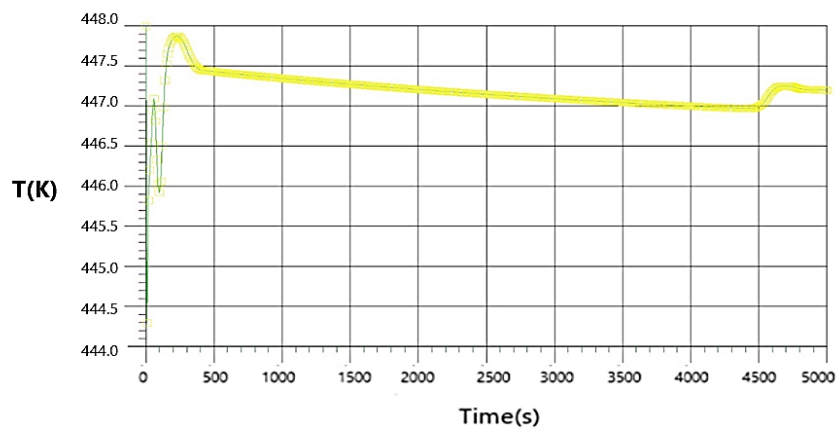
**Fig. 9.** Temperature variation diagram vs. time in the outlet of the IAUTL core for NC fluid flow



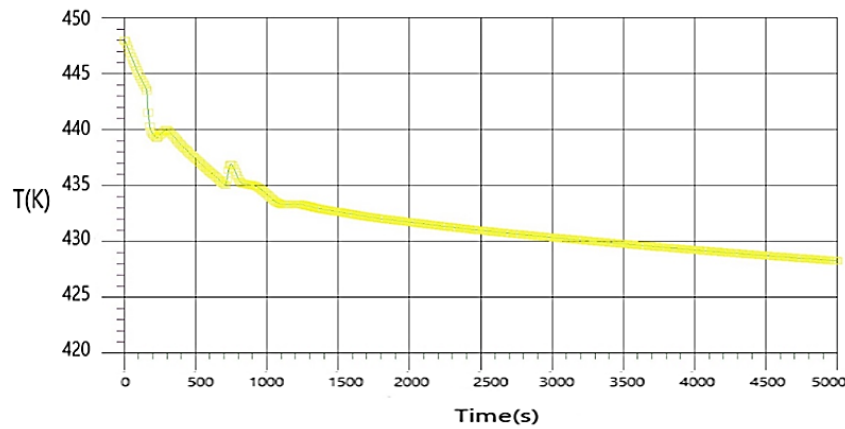
**Fig. 10.** Pressure variation diagram vs. time in the outlet of the IAUTL core for NC fluid flow

#### 3.4. HE evaluation with RELAP5 code for NC fluid flow

HE heat removal is assessed by means of RELAP5/MOD3.2 code for the NC fluid flow. The examples of temperature variations in the inlet and outlet of HE for the NC fluid flow are revealed in Figs. 11 and 12. In Fig. 11, after 500 of a second, the diagram variation is linear or constant. As it is apparent from Fig. 12, after the first 1500 seconds of simulation, the diagram changes linearly, and it is proper to calculate the average value. From Figs. 10 and 12, the following data are listed in Tables 9. Table 9 shows the results of core and heat exchanger temperature changes using RELAP5 simulation results for forced and natural fluid flows.



**Fig. 11.** Temperature variation diagram vs. time in the inlet of HE for NC fluid flow



**Fig. 12.** Temperature variation diagram vs. time in the outlet of HE for NC fluid flow

**Table 9.** Comparison of IAUTL core and HE temperatures from RELAP5 simulation results for FC and NC fluid flow

Parameter	Core, NC	Core, FC	HE, NC	HE, FC
Average inlet temperature (°C)	154.85	154.28	175.00	169
Average outlet temperature (°C)	174.00	169.4	155.00	156
Average difference temperature (°C)	19.15	15.12	20.00	13

From the comparison of the simulation results of the core temperature in the natural and forced modes in Table 9, it can be concluded that the maximum outlet temperature of the reactor core in the natural and forced flow is 169.4°C and 174°C, respectively. This increase in temperature in the natural flow compared to the forced flow is consistent with the experimental condition and is due to the increase in residence time due to the lower fluid velocity in the natural flow regime around the reactor core. Also according to Table 9, the temperature drop in the heat exchanger in the natural and forced flow is 20°C and 13°C, respectively. This increase in heat loss in the natural flow compared to the forced flow is due to the lower natural flow velocity in the heat exchanger, and as a result, the heat exchange also increases in the natural flow. Core pressure variations in Fig. 5 and Fig. 10 for the NC and FC fluid flows are represented in Table 10.

**Table 10.** Comparison of IAUTL core pressure RELAP5 simulation results for NC and FC fluid flow

Parameter	IAUTL (scaling results)	IAUTL (RELAP5), FC	IAUTL (RELAP5), NC
Average inlet pressure (bar)	10.00	10.00	10.00
Average outlet pressure (bar)	8.10	8.00	9.44
Difference in pressure (bar)	1.90	2.00	0.56

From the comparison of simulation and scaling results in Table 10, it can be concluded that the pressure drop between the inlet and outlet of the reactor core, in both free and forced flow, is approximately 2 bar, and there is a good agreement between the modeling and scaling data. The simulation results demonstrate that RELAP5/MOD3.2 code can simulate a test loop core fairly in cases of the FC and NC fluid flows. Hence, IAUTL is adequately able to remove the core heat by means of the FC and NC fluid flows.

#### 4. Conclusion

By using the energy scaling methodology, a typical PWR and its primary circuit are employed to scale down IAUTL for the FC fluid flow. IAUTL is designed as a reduced-height, reduced-temperature, and reduced-pressure test loop. The comparison between the significant parameters of scaling results and the experimental data is carried out. The good agreement between IAUTL scaling results and PWR experimental data signifies the validation of the scaling

computations. Meanwhile, IAUTL core heat removal is investigated by using the RELAP5/MOD3.2 code for the NC and FC fluid flows. The FC and NC simulations with the RELAP5 codes are valid on account of the validation of IAUTL scaling. From the simulation results, it is proven that IAUTL is adequately able to remove the core heat using the FC and NC fluid flows. Thus, IAUTL can be utilized for analyzing the normal and transient conditions. The maximum temperature in the core inlet is 154.28°C, the maximum temperature in the core outlet is 169.4°C, the maximum temperature difference between the core inlet and outlet is 15.12°C, and the average temperature in the core is 161.84°C from the RELAP5/MOD3.2 code for FC fluid flow.

The HE temperature comparison between scaling and simulation results for FC fluid flow shows that the HE average inlet and outlet temperature in the RELAP5 simulation results is very close to the scaling results. From the comparison of the simulation results of the core temperature in the natural and forced flow, it can be concluded that the maximum outlet temperature of the reactor core in the natural and forced flow modes is 169.4°C and 174°C, respectively. This increase in temperature in the natural flow compared to the forced flow is consistent with the experimental condition and is due to the increase in residence time due to the lower fluid velocity in the natural flow regime around the reactor core. Also, the temperature drop in the heat exchanger in the natural and forced flow is 20°C and 13°C, respectively. This increase in heat loss in the natural flow compared to the forced flow is due to the lower natural flow velocity in the heat exchanger, and as a result, the heat exchange also increases in the natural flow. From the comparison of simulation and scaling results, it can be concluded that the pressure drop between the inlet and outlet of the reactor core, in both free and forced flow, is approximately 2 bar, and there is a good agreement between the modeling and scaling data. The simulation results demonstrate that RELAP5/MOD3.2 code can simulate a test loop core fairly in cases of the FC and NC fluid flows. Hence, IAUTL is adequately able to remove the core heat by means of the FC and NC fluid flows.

## Nomenclature

### Symbols

$a_{OR}$	Global area ratio
$h$	Specific enthalpy (Jkg <sup>-1</sup> )
$h_{fg}$	Latent heat (Jkg <sup>-1</sup> )
$l_{OR}$	Global length ratio
$M$	Primary mass coolant inventory (kg)
$P$	Pressure (Pa)
$Q$	Thermal power (W)
$t$	Time (s)
$T$	Temperature (K)
$V$	Volume of primary system (m <sup>3</sup> )

### Greek symbols

$\rho$	Density (kgm <sup>-3</sup> )
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### Subscripts

0	Initial parameters
avg	Average
c	Core
CL	Cold Leg
comp	Components
HL	Hot Leg
p	Primary loop
R	Ratio of model to prototype

Ratio	Ratio of value to average value
s	Secondary loop
sat	Saturated water

### Abbreviation

ACC	Accumulator
DP	Dimensionless Parameter
FA	Fuel Assemblies
FC	Forced Circulation
FE	Fuel Elements
HE	Heat Exchanger
IAUTL	Islamic Azad University Test Loop
INEL	Idaho National Engineering Laboratory
ITF	Integral Test Facility
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
NC	Natural Circulation
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
LWR	Light Water Reactor
PRZ	Pressurizer
PWR	Pressurized Water Reactor
SG	Steam Generator

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