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Research Note

Experimental Study of Small and Medium Break LOCA in the TTL-2 Thermo-Hydraulic Test Loop and Its Modeling with RELAP5/MOD3.2 Code

M. Taherzadeh[1,*], J. Jafari[2], N. Vosoughi[1] and H. Arabnezhad[1]

Abstract. *Small and medium break LOCA accidents at low pressure and under low velocity conditions have been studied in the TTL-2 Thermo-hydraulic Test Loop, experimentally. TTL-2 is a thermal hydraulic test facility which is designed and constructed in NSTRI to study thermal hydraulic parameters under normal operational and accident conditions of nuclear research reactors. A nodalization has been developed for the TTL-2 and experimental results have been compared with RELAP5/MOD3.2 results. The considered accidents are a 25% and 50% cold leg break without emergency core cooling systems. Results show good agreement between experiments and RELAP5/MOD3.2 results. This research provides experimental data for evaluation of thermo hydraulic codes for nuclear research reactors, and verifies that RELAP5/MOD3.2 has a good capability to estimate the thermal hydraulic behavior of low pressure and low velocity thermal hydraulic systems, such as research reactors under steady state and transient conditions.*

Keywords: *TTL-2 thermal hydraulic test loop; LOCA; RELAP5/MOD3.2; Experimental results; Code validation; Research reactor.*

INTRODUCTION

One of the main objectives of nuclear reactor safety is to maintain the reactor core in such a condition that there is no release of radio nuclides into the environment. In order to ensure this, many computer codes and thermo-hydraulic test facilities have been developed to estimate the behavior of the nuclear reactors under steady state operational and transient conditions. Extensive theoretical and empirical studies have been performed in this area [1-4], but most of them are performed in high pressure and high velocity thermal hydraulic systems. Because of the different behavior of water at low pressure, neither their results nor the provided codes could be applied for research reactors without careful and precise investigation. A comparison between the RELAP5/MOD3 and PARET/ANL codes for the analysis of IAEA benchmark transients has been performed and results show good agreement between the two [5]. Another comparison between the STHRIP-1 and RELAP5 codes is done by Antonella et al. under a steady state condition that showed good agreement between the two codes [6]. The Onset of Flow Instability (OFI) in research reactors operating conditions was studied by Hamidouche et al. [7] and results indicate the use of RELAP5 is not straightforward and needs to be complemented by additional experimental work. The Loss of Coolant Accident (LOCA) in the Tehran Research Reactor (TRR) was simulated by Jafari et al. [8]. Their results showed that although RELAP5/MOD3.2 code can simulate research reactors actually under operating conditions, this code simulates large transient phases less accurately.

One of the purposes of this research is to verify the capability of RELAP5/MOD3.2 under steady state and

1. *Department of Energy Engineering, Sharif University of Technology, Terhan, P.O. Box 11155-11155, Iran.*
2. *Reactor & Accelerators Research and Development School, Nuclear Science and Technology Research Institute (NSTRI), Tehran, P.O. Box 14395-836, Iran.*
*. *Corresponding author. E-mail: taherzadeh@energy.sharif.ir*

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transient conditions for application of nuclear research reactors. Furthermore, this research provides experimental data for the evaluation of thermo hydraulic codes for the research reactors. RELAP5 is a thermo-hydraulic system code for modeling of nuclear power plants (high pressure and high velocity condition), but its new version has the capability to also model research reactors (low pressure and low velocity condition) [9].

One important event initiating events in the design basis accident category is the Loss of Coolant Accident which has been studied experimentally in the Thermal-hydraulic Test Loop 2 (TTL-2) and has been analyzed with the RELAP5/MOD3.2 code. In the present work, at first, a brief description of the TTL-2 and some correlations that have been used at the design stage are given. Application of the RELAP5 and developing the TTL-2 nodalization has been explained in the next sections. The LOCA with 25% and 50% cold leg break in the TTL-2 is simulated by the code and compared with obtained experimental data at the final stage.

DESCRIPTION OF THE TTL-2

TTL-2 apparatus is the modified version of the TTL-1, which was previously designed and constructed in NSTRI [10-12]. This test facility is typical, which contains the main components of a nuclear reactor and is capable of doing the various thermal-hydraulic experiments under steady state, transient and accident conditions. Table 1 compares some characteristics of the TTL-2 thermo hydraulic test loop with other well-known test loops. Moreover, by using the experimental data of this test facility, it is possible to verify the safety and thermo hydraulic parameters of research reactors. Another application of the TTL-2 is verification of thermo hydraulic codes. Regarding the previous configuration of the test facility, some modification has been made, in order to comply with a need for accurate representation of nuclear reactor components. The schematic diagram of the TTL-2 is depicted in Figure 1. This test facility is divided into two sections, primary loop and secondary loop. The secondary loop is designed to remove the generated heat in the primary loop. Hydro-thermal characteristics of the test loop are given in Table 2. Figure 2 shows a picture of the Test Section (TS) of the loop and the locations of thermocouples T1, T3 and T5. Figures 3 and 4 show the sealing of TS heaters and other detailed information of TTL-2, respectively.

Test Section

The main element of the test facility is the test section which contains 31 heater rods in a stainless steel duct. A transparent window is created at the upper part of the duct so that fluid regimes can easily be tracked. In this activity, heating elements represent the hot

Table 1. A comparison between the TTL-2 and other well-known test loops.

	Loop Name	Reference Reactor	Year	Scale	Power (MW)	Tests	Country
1	LOBI,M1,2	W-PWR	1991-1972	1:700	5.3	L,M,SBLOCA	Italy
2	PKL-1,2,3	KWU-PWR	1986-1977	1:145	2.5	L,M,SBLOCA	Germany
3	BETHSY	FRA-PWR	1982	1:132	3	LOCA-NC	France
4	SPES-1,2	AP-600	1993-1989	1:611	4.9	SLOCA-NC	Italy
5	PACTEL	VVER-440	1991	1:305	1	LOCA	Finland
6	FARO	LWR	1999-1987	-	-	Core melt	Italy
7	KROTOS	LWR	1995	-	-	Core melt	France
9	THTL	LWR	1988-1987	-	0.1	FC-two phase	Japan
10	LPITF	SBWR	1995	-	10	NC	USA
11	LSTF	W-PWR	1995	1:48	10	SGTR	Japan
12	PSB-VVER	VVER-1000	2002-1998	1:300	0.038	Accident management	Russia
13	REWET-1,2	VVER-440	1989	-	0.035	ECCS-NC	Russia
14	MTT-1	-	2003	-	0.005	NC	Italy
15	Piper-one	SBWR		1:2200		SB-LOCA,NC	Italy
16	TTL-1	-	2003	-	0.022	NC	Iran
17	TTL-2	-	2009	-	0.025	M,SBLOCA-NC-LOHA-LOFA	Iran

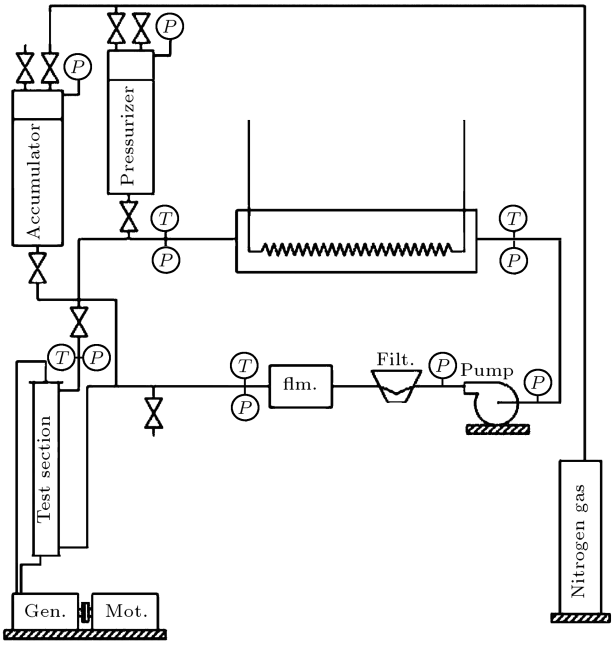


Figure 1. Sketch of TTL-2 loop.

Table 2. Thermal hydraulic characteristics of TTL-2.

Test Facility Parameter	Value
Heater power (W)	25000
Primary circuit pressure (bar)	5
Test section length (m)	1.1
Core length (m)	0.8
Number of fuel rods (heaters)	31
Down comer volume (m^3)	4.91E-04
Test section volume (m^3)	2.97E-03
Hot leg volume (m^3)	1.80E-03
Cold leg volume (m^3)	2.75E-03
Heat exchanging surface (m^2)	1.006
Water volume in primary circuit (m^3)	6.19E-02
Water volume in pressurizer (m^3)	4.87E-02
Nominal flow rate (kg/s)	0.36

assembly fuel rods in the reactor core. Heater rods with diameter of 9.5 mm are so arranged that the equivalent hydraulic diameter of the test section is equal to the hydraulic diameter of the Hot Channel (HC) in the reactor core.

Heat Exchanger

In order to remove the heat absorbed by the coolant, a Heat Exchanger (HE) is used. This HE is a shell and tube one that plays the role of the cooler when the fluid is in a single phase, and a sub cooled state and

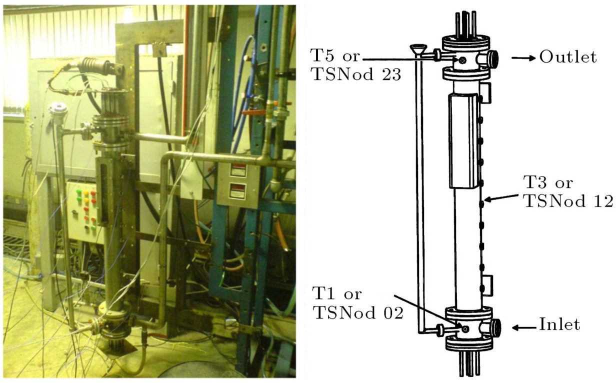


Figure 2. Test section of the loop and location of the thermocouples.

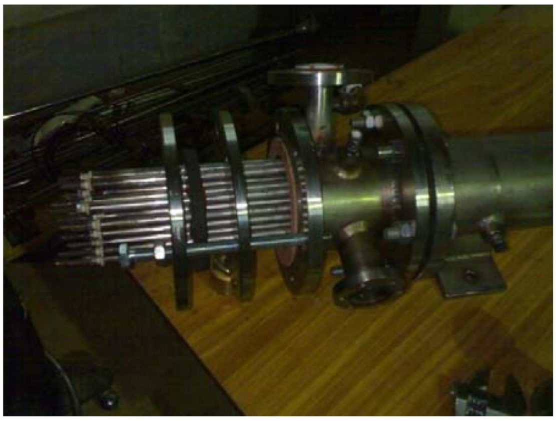


Figure 3. Sealing of test section's heaters.

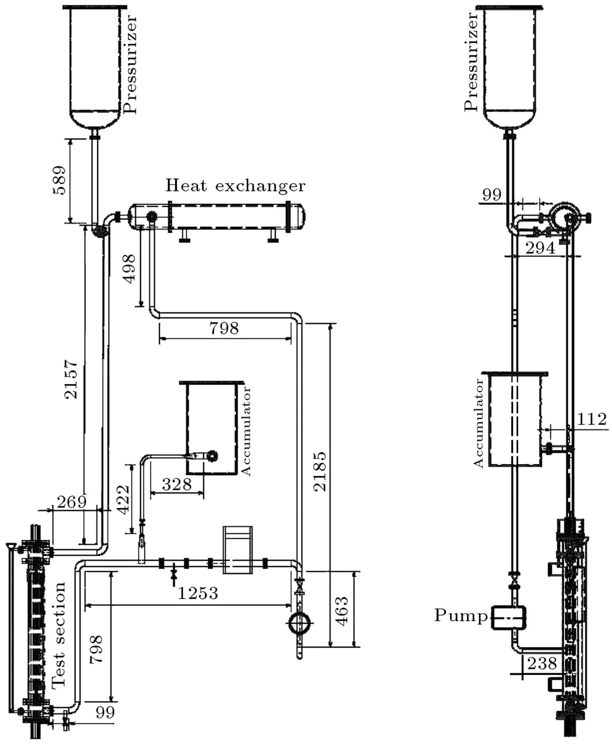


Figure 4. A drawing of the TTL-2 for detailed information.

condenser when the phase is changed into the primary side. Moreover, the HE can be placed horizontally or vertically which affects the study of natural circulation. The heat transfer area of the HE is 1.006 m^2 and seven U-shaped tubes with a length of 70 cm are included. So the total length of the tubes in HE is 14 m, surrounded by a shell with an inner diameter of 13.0 cm.

Pressurizer

The operating pressure of the system is 5 bars. This pressure is provided and sustained by a pressurizer (PRZ). Regulating pressure in the pressurizer tank is done by the injection of nitrogen gas from a nitrogen vessel with constant pressure and not by a heater and spray system like conventional PWRs. Thus the temperature of water in the PRZ in the experiments (about 40°C) will be lower than the water temperature of the loop.

Power Source

Power in the heater rods is generated by electric energy. Because of the noise that may be induced by alternative current in the measuring tools, a direct current should be supplied to the heaters. Conversion of AC to DC is done by coupling an electrical motor to a DC generator, so the motor drives the generator. The required voltage and current are adjusted by two rheostats; one with fine gain and another with coarse gain. Here, the reactor scram is modeled by the heater rods power cut-off under accident conditions, which can be done by breaking the excitation circuit of the generator. The nominal power of the motor-generator is 25 kW.

Data Acquisition System

The data acquisition system is of great importance in experimental studies. The precision of instruments, data processing and control of conditions during the test play an important role in the reliability of the obtained results. Pressure (P), temperature (T), flow meter ($flm.$) and level meters have been used to measure physical quantities. Table 3 shows the data acquisition system accuracy. Data from sensors are transferred to a computer and are saved by Labview software.

Table 3. Data acquisition system accuracy.

Accuracy	Sensor
Temperature sensor	$\pm 2.5 \pm 0.0075\ \lvert T(°\mathrm{C})\rvert$
Flow meter	± 0.001 ml/sec
Water level meter	± 1 mm
Pressure sensor	± 0.005 bar

DESCRIPTION OF THE RELAP5/MOD3.2 SYSTEM CODE

RELAP5 is a highly generic code that in addition to calculating the behavior of a reactor coolant system during a transient can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of steam, water and non-condensable gases [9]. The results obtained by the code have been evaluated by experiments on the scaled down facilities. The former versions were only applicable to power reactor conditions, but the later ones are so designed to also take in the research reactor operating conditions.

The Light Water Reactor (LWR) transient analysis code, RELAP5, was developed at the Idaho National Laboratory (INL) for the U.S. Nuclear Regulatory Commission (NRC). Code uses include analyses required to support rulemaking, licensing audit calculations, evaluation of accident mitigation strategies, evaluation of operator guidelines and experiment planning analysis. RELAP5 has also been used as the basis for a nuclear plant analyzer. Specific applications have included simulations of transients in LWR systems, such as loss of coolant [13], Anticipated Transients Without Scram (ATWS), and operational transients, such as loss of feedwater [14], loss of offsite power [15], station blackout [16] and turbine trip.

MODELING AND NODALIZATION OF THE TTL-2 USING RELAP5/MOD3.2 CODE

The test section region is modeled with a pipe and by 24 volumes. Sixteen of these volumes refer to the active length of heater elements that produce heat electrically. All heating elements with uniformly dispersed power source are considered to be accumulated in a single rod. So, it is assumed that all rods are identical in the temperature profile. This assumption seems to be reasonable according to the similar resistance of the rods. Some thermal-hydraulic parameters in steady state operation have been calculated analytically to see if the design values are accessible and, also, to verify the simulation of the code. First, the convective Heat Transfer Coefficient (HTC) is to be determined. Numerous investigations have been carried out to determine the HTC in channels [17]. But for large temperature drops across the film, the physical property most affected is the viscosity. For such a case, the Sieder-Tate correlation is used [18]:

$$\mathrm{Nu} = 0.023\ \mathrm{Re}^{0.8}\mathrm{Pr}^{0.4}\left(\frac{\mu}{\mu_w}\right)^{0.14}. \quad (1)$$

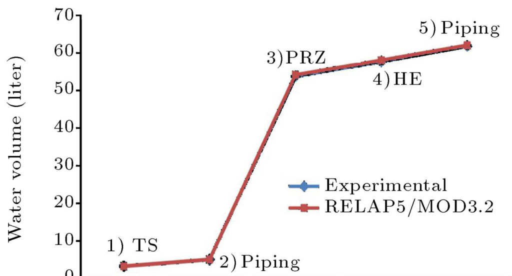
The boundary conditions were set in the input. The steady state has lasted sufficiently and all initial conditions for the start of a LOCA accident were

Table 4. Comparison of design values.

Design Value	Code Results	Correlations	Experimental
HTC in hot channel (W/K.m^2)	1689	1718	-
Fuel centerline temperature (°C)	215	217.9	215
Clad surface temperature (°C)	118	117.6	-
Bulk coolant temperature (°C)	98	98	98
Temperature difference between input and output of test section (°C)	16	16.47	16.45

encountered with sufficient accuracy. Table 4 shows test loop design parameters that were calculated by the RELAP5 code and correlations, and verified by some experimental data.

In order to ensure proper modeling of TTL-2 by RELAP5, some parameters in a steady state condition were compared by experimental data. The results of this comparison show that the input file properly models the test facility. For instance, the water inventory of the test loop has been compared with the experimental data in Figure 5. Figure 6 shows RELAP5 nodalization for TTL-2. For simplification, the secondary loop has been modeled as an open loop with proper boundary conditions representing the heat sink parameters to the atmosphere. Therefore, water flows from a time dependent volume (TMDPVOL 900) through a time dependent junction (TMDPJUN 910) and then goes to another volume (TMDPVOL 940). The pressurizer is modeled with a pipe by 10 volumes. In order to adjust the loop pressure by nitrogen gas, a time dependent volume (TMDPVOL 560) is connected to the pressurizer. Moreover, two time dependent volumes (TMDPVOL 41 & 43) are used for the adjustment of the pressurizer water level at a steady state condition. Also, the down comer is modeled by a pipe with 12 volumes (Pipe 800). Furthermore, for modeling the LOCA, an on-off valve (component 102) connected to the time dependent volume 101, representing the atmosphere pressure and



	1	2	3	4	5
Experimental	3.24	5.04	53.79	57.76	61.86
RELAP5/MOD3.2	3.25	5.05	54.15	58.03	62.03

Figure 5. Comparison between experimental and RELAP5 input water inventory of TTL-2.

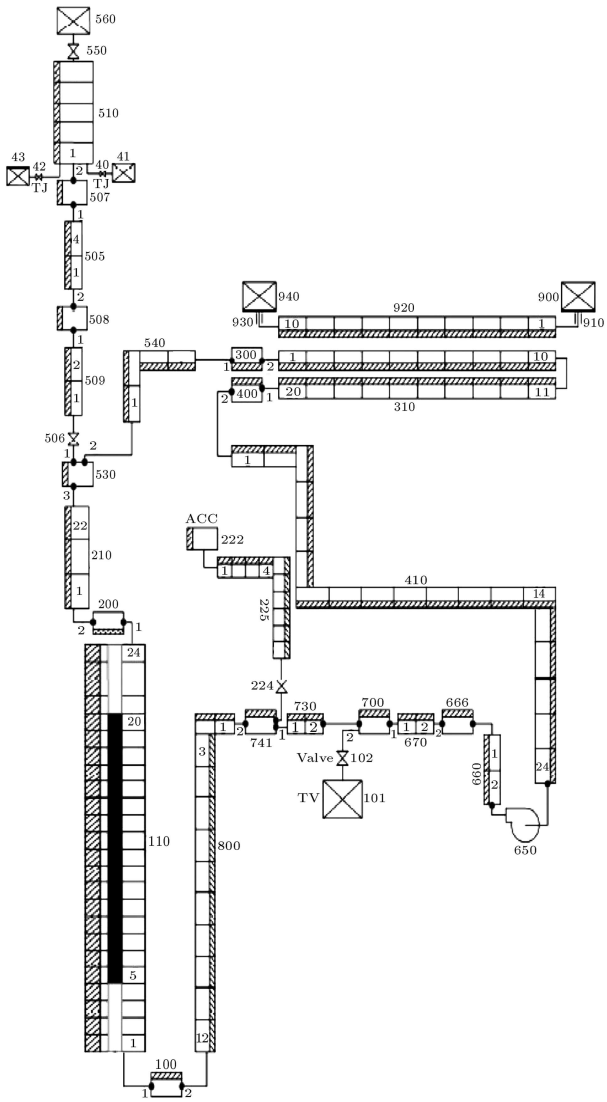


Figure 6. RELAP5 nodalization for TTL-2.

temperature, has been used. For LOCA simulation, this valve is opened and the pressurizer valves 40, 42 and 550 are closed by a time logical trip. After the break occurrence, the main circulating pump is turned off and power is decreased to ten percent of nominal power, or 2.5 kW, as conservative fuel rods decay the heat value. During the LOCA test in the experiment and RELAP5 modeling, nitrogen gas is allowed to enter the system, and the effect of non-condensable gas has been considered in the RELAP5/MOD3.2 input.

RESULTS AND DISCUSSIONS

Obtained results (experimental and theoretical) and their comparison are shown in Figures 7 to 21.

Figure 7 compares temperature distribution in the TS at a steady state condition. The coolant enters the TS with 88°C and exits with 104°C. The coolant temperature rise is 16°C, which is coincident with the TS heat balance equation, and there is a good agreement between the code results and experimental data.

Figures 8 to 11 show distributions of fluid temperatures in LOCA with 25% and 50% cold leg breaks, respectively. The accident occurs at 300 seconds; at the beginning of the accident, there is a spike in the temperature because of changing the coolant flow direction and a delay in scramming of the heater power. After that, the temperatures decrease rapidly on account of the blow down phase and the downward flow of the pressurizer water to the TS. It is clear that there is a higher decreasing rate of temperature in 50% of cold leg break cases, which can be seen in Figures 10 and 11. In the next phase of the accident, the fluid temperature remains constant due to the high water inventory of the loop. The coolant temperature remains constant at the last phase of the accident at the TS inlet (Figure 8), where there is no heat generation. Figure 9 shows the coolant temperature at the middle of the TS. It shows that the coolant temperature increases gradually in the last phase of

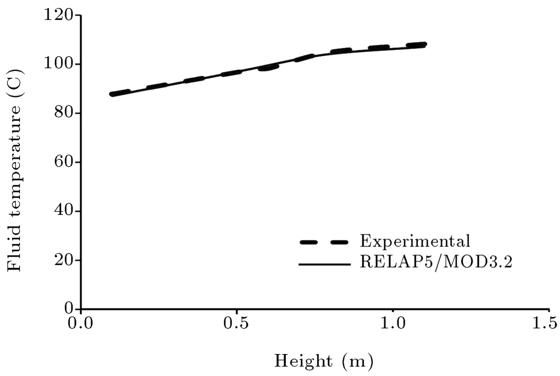


Figure 7. Experimental and analytical results for temperature distribution in the T.S. at steady state condition.

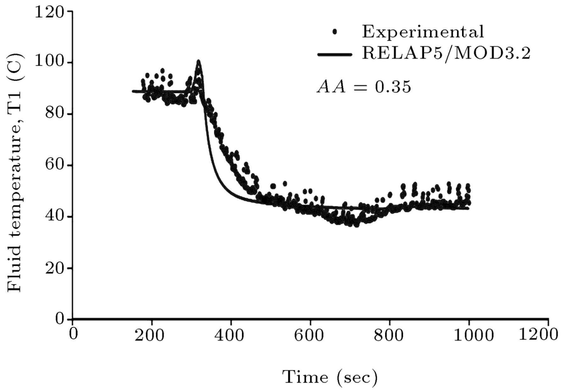


Figure 8. Experimental and analytical results for temperature of TSNOD02–LOCA 25%.

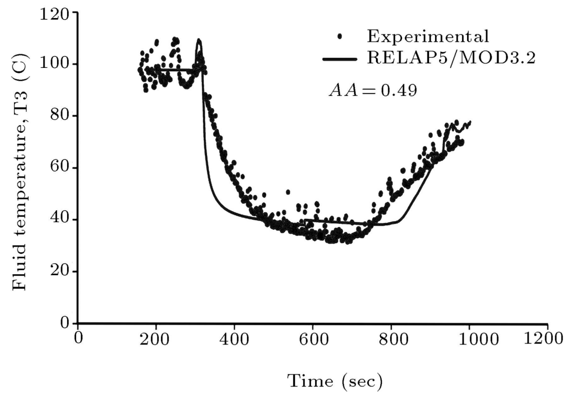


Figure 9. Experimental and analytical results for temperature of TSNOD12–LOCA 25%.

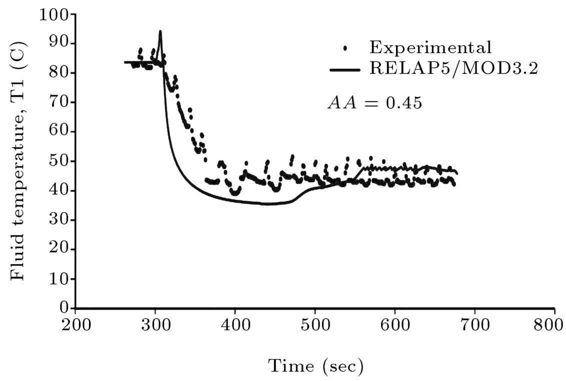


Figure 10. Experimental and analytical results for temperature of TSNOD02-LOCA 50%.

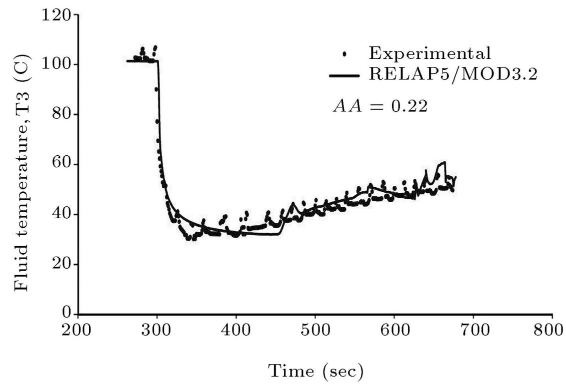


Figure 11. Experimental and analytical results for temperature of TSNOD23–LOCA 50%.

the accident, where the coolant flow is almost zero and decay heat is generated. It should be mentioned that the experimental apparatus only showed single phase thermal hydraulics. There is no boiling or phase change taking place during all phases of the accident.

Figures 12 and 13 show distributions of heater center line temperatures in LOCA with 25% and 50% cold leg breaks, respectively, and the chronology of the accidents is similar to Figure 9.

Figures 14 to 19 show the primary loop pressure, the water level of the pressurizer and the integral break flow during 25% and 50% cold leg breaks, respectively. As expected, the pressure and water level of the pressurizer at LOCA with a 50% cold break decreases more rapidly and the break flow rate is more than 25%. There are good agreements between experimental and analytical results.

Figures 20 and 21 show water levels in the TS during 25% and 50% cold leg breaks, respectively. The accident occurs at 300 seconds, but the TS remains full of water, with a constant water level up to 593 sec and 360 sec for 25% and 50% breaks, respectively. After these times, the injection of nitrogen gas into the loop causes the TS water level to decrease. There is a difference (less than 25%) between the experimental and

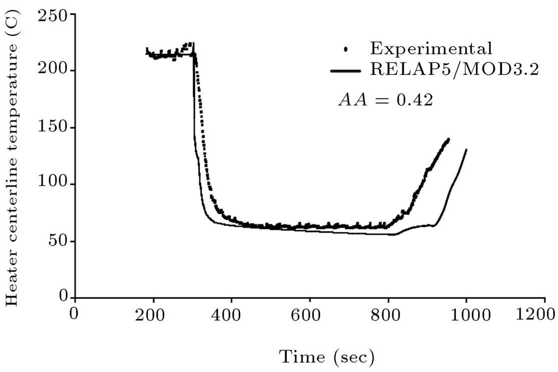


Figure 12. Experimental and analytical results for heater center line temperature of TSNOD21-LOCA 25%.

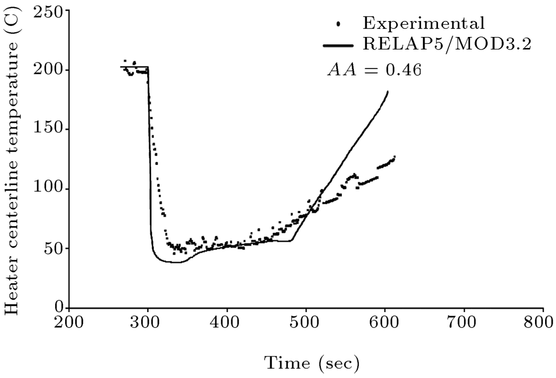


Figure 13. Experimental and analytical results for heater center line temperature of TSNOD21-LOCA 50%.

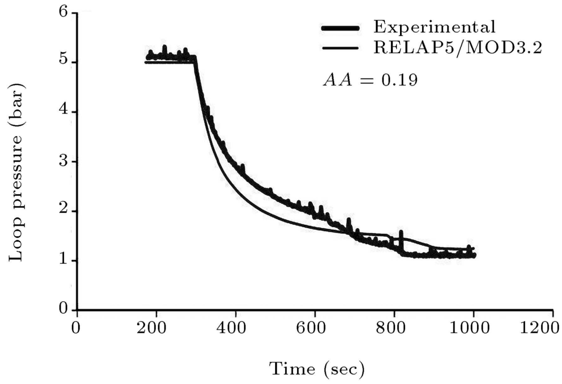


Figure 14. Experimental and analytical results for pressure of loop at LOCA 25%.

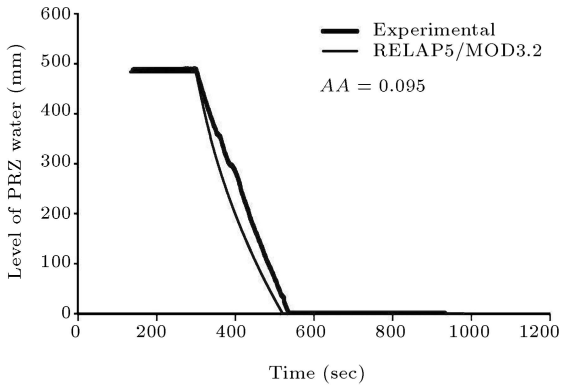


Figure 15. Experimental and analytical results for pressurizer water level of loop at LOCA 25%.

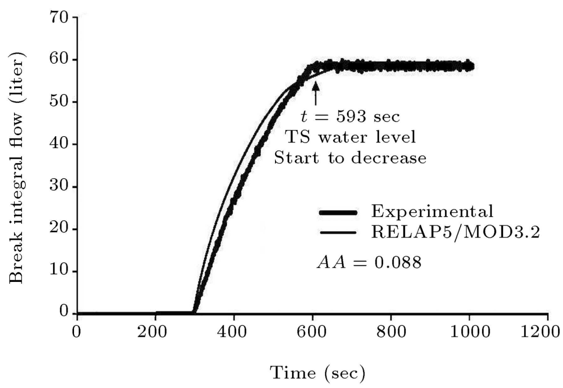


Figure 16. Experimental and analytical results for break integral cold leg flow at LOCA 25%.

analytical results in the last phase of both accidents. This is because of a small difference between the break flow rates in the analytical results and the experimental data, which is related to the type of break flow model and its uncertainty in the prediction of a break flow rate. Furthermore, due to the small ratio of hydraulic diameter to height of the TS, a small difference between break flow rates results in a big difference in the water level of the TS.

A quantitative error analysis regarding RE-

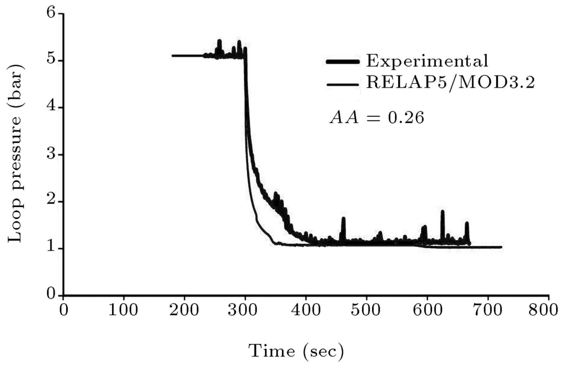


Figure 17. Experimental and analytical results for pressure of loop at LOCA 50%.

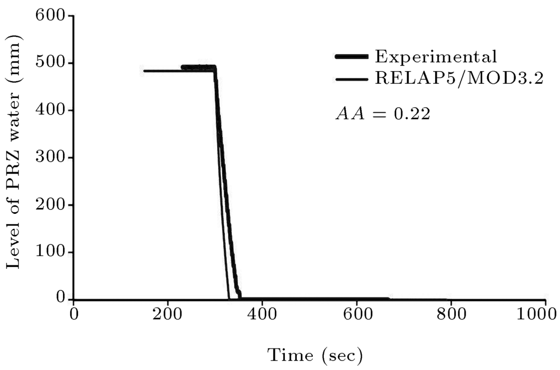


Figure 18. Experimental and analytical results for pressurizer water level of loop at LOCA 50%.

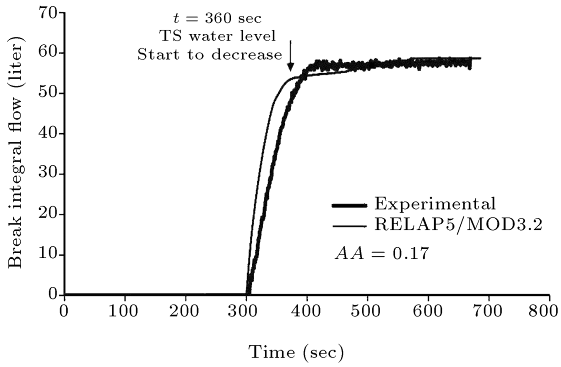


Figure 19. Experimental and analytical results for break integral cold leg flow at LOCA 50%.

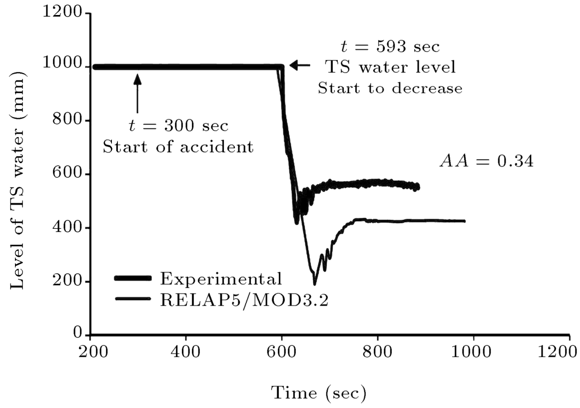


Figure 20. Experimental and analytical results for test section level at LOCA 25%.

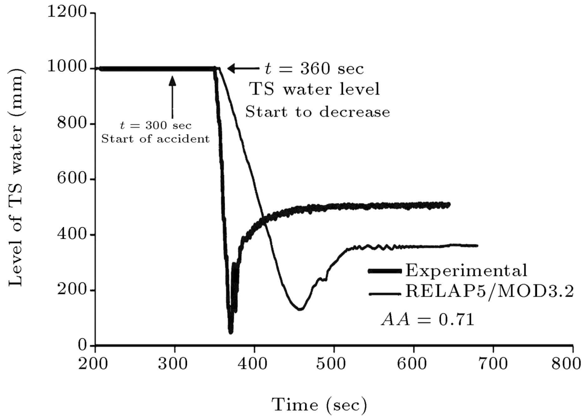


Figure 21. Experimental and analytical results for test section level at LOCA 50%.

LAP5/MOD3.2 in a small and medium break LOCA with a Fast Fourier Transform Based Method (FFTBM) [19] was performed to show the accuracy of the code and TTL-2 nodalization. In this method, the AA factor can be considered as "average fractional error". The most significant information given by AA is the relative magnitude of the discrepancy coming from the comparison between the calculation and the corresponding experimental variable time history. When the calculated and the experimental data are equal, then the error function is zero (AA is also equal to zero), characterizing perfect agreement. But the most suitable factor for definition of an acceptability criterion and overall code accuracy is the total AA_{tot}, with the following characterizations:

$$(AA)_{\text{tot}} = \sum_{n=0}^{N} (AA)_i.(w_f)_i, \tag{2}$$

$$\sum_{n=0}^{N} (w_f)_i = 1. \tag{3}$$

With:

$AA_{\text{tot}} = 0.3$	characterize very good code predictions,
$0.3 < AA_{\text{tot}} \leq 0.5$	characterize good code predictions,
$0.5 < AA_{\text{tot}} \leq 0.7$	characterize poor code predictions,
$AA_{\text{tot}} > 0.7$	characterize very poor code predictions.

where N is the number of parameters, and $(w_f)_i$ are weighting factors introduced for each parameter.

In this research, for LOCA 25%, AA_{tot} equals 0.3660, and for LOCA 50%, equals 0.3560, which show good code predictions.

CONCLUSIONS

An experimental and analytical study was performed using a TTL-2 thermo hydraulic test loop to provide experimental and analytical data with the RELAP5/MOD3.2 system code. Experimental data for two accidents, LOCA 25% and LOCA 50%, have been obtained from TTL-2 and compared with RELAP5/MOD3.2 outputs. Experimental data include the test section temperatures, loop pressure, pressurizer and test section water levels and integral cold leg break flow. The results show good agreement between the experimental data and the results provided by the code. This research verifies that RELAP5/MOD3.2 has a good capability to estimate the thermal hydraulic behavior of the low pressure and low velocity thermal hydraulic systems, such as research reactors at steady state and transient conditions.

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BIOGRAPHIES

Morteza Taherzadeh Fard received his B.S. degree in Mechanical Engineering (heat and flow) from the University of Kashan in 2007, and his M.S. degree in Nuclear Engineering from Sharif University of Technology in 2009, where he is now a Ph.D. student in that field. His research interests include: Safety Analysis of

Nuclear Reactors and Study of the Thermal Hydraulic Parameters of Nuclear Reactors and Two Phase Flow.

Jalil Jafari received his B.S. degree in Mechanical Engineering (heat and flow) from the University of Tabriz in 1990, his M.S. degree in Nuclear Engineering from Sharif University of Technology in 1994, and his Ph.D. in Nuclear Engineering from Amir Kabir University of Technology in 2002. He is now an Assistant Professor with the Atomic Energy Organization of Iran. His research interests include: Safety Analysis of Nuclear Reactors and Study of the Thermal Hydraulic Parameters of Nuclear Reactors.

Naser Vosoughi received his B.S. degree in Physics from the University of Tehran in 1993, and his M.S. and Ph.D. degrees in Nuclear Engineering from Sharif University of Technology in 1997 and 2002, respectively. He is now an Assistant Professor at Sharif University of Technology. His research interests include: Neutronic Design and Analysis of Nuclear Reactors.

Hadi Arabnejad received his B.S. degree in Mechanical Engineering (heat and flow) from the University of Kerman in 2007, and his M.S. degree in Nuclear Engineering from Sharif University of Technology in 2009. His research interests include: Safety Analysis of Nuclear Reactors and Study of the Thermal Hydraulic Parameters of Nuclear Reactors.