ASSESSMENT OF RELAP5 MODEL FOR THE UNIVERSITY OF MASSACHUSETTS LOWELL RESEARCH REACTOR

by

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Received on February 27, 2006; accepted on May 31, 2006

RELAP5 is a system code developed at the Idaho National Environmental and Engineering Laboratory for thermal hydraulic analysis of nuclear reactors. The code RELAP5 is widely used for safety analysis studies of commercial nuclear power plants. However, recent released version of RELAP5/3.2 and over present significant capabilities for analysis of nuclear reactor research systems. As a contribution to the assessment of RELAP5/3.3 for research reactor safety analysis, experimental data from the University of Massachusetts Lowell Research Reactor UMLRR are used. The UMLRR is a 1 MW, light water moderated and cooled, graphite-reflected, open-pool type research reactor.

This paper presents the development and the validation of a UMLRR-RELAP model using experimental data. For this purpose, a series of experiments were performed for benchmarking RELAP5 calculations for research reactor systems. As a result of this study, the UMLRR nodalization is shown to be representative of the experimental data reactor behavior.

Key words: research reactor, thermal hydraulic analysis, safety analysis, transient analysis, RELAP5

INTRODUCTION

The Reactor Excursion and Leak Analysis Program (RELAP) is an advanced thermal hydraulics system code used for the simulation of a wide range of transients and accident in power reactors such as loss-of-coolant accidents (LOCA), anticipated transient without scram, loss of feed water, loss of flow incidents, and reactivity transients. It is based on coupled equations that reflect thermal hydraulic reactor coolant system and neutron kinetics of the reactor core.

Scientific paper
UDC: 621.039.4:519.7:536.24
BIBLID: 1451-3994, *21* (2006), 1, pp. 3-12

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The RELAP5 code has been widely used around the world by governments and industry for nuclear reactor simulation and safety analysis for the past two to three decades. As a result, an extensive validation and assessment work has been done around power reactor operating conditions. Furthermore, even though the RELAP5 code system was first devoted for accident analysis in nuclear power plants, there are significant capabilities in recently released version of RELAP5 that could be extended to nuclear research reactor safety analysis providing some assessment and verification. Indeed, so far, research reactor safety analysis was performed using conservative computational tools [1-3].

However, nowadays, with extensive use of research reactor, there is a real need to get more realistic simulations of the phenomena involved during steady-state and transient conditions and eventually the identification of design/safety requirements that can be relaxed or enhanced [4]. So far, only restricted works were performed to assess the applicability of the code to research reactors operating conditions (low pressure, low mass flow rates, low power, *etc.*) [5, 6] and the present work represent a contribution to the assessment of the RELAP5 for research reactor operating conditions.

In this framework, measurement data retrieved from previous experiments performed in the University of Massachusetts Lowell Research Reactor (UMLRR) are used to illustrate many of the basic concepts needed in the modeling, validation, and application of RELAP5 for general research reactor systems.

Indeed, significant amounts of measured data from the UMLRR were available through the control and data acquisition system CDAS. In 2003, the UMLRR completed a major control room upgrade, which included process control, process monitoring, and data acquisition systems. The CDAS provides a mix of instrument readings and plot trends using real-time or historical data. Most of the data parameters within the UMLRR (power level indicators, various temperatures, flow rates, pressures, and conductivity measurements, control blade position indicators, on/off status of various motors, valves, fans, *etc*.) can be retrieved in several forms such as real-time display, data trending, and archival storage. The CDAS can easily record a variety of experimental data that are not generally available and easily accessed from typical research reactors.

Furthermore, the UMLRR provide another unique capability by making the research reactor data available to educational users via a standard web browser, located at www.nuclear101.com. The purpose of this website is to serve as an educational resource for students, instructors, and working professionals who are interested in the nuclear engineering field with primary focuses on nuclear reactor physics, reactor operations, and the modeling and analysis of nuclear systems. The unique aspect of this site is to provide a direct link for the real-time operating data from the UMLRR (power level, flow rate, various temperatures, *etc*.), which allows a user to view reactor operational data, in real time, as an experiment is being performed. This represents a great opportunity for use of the UMLRR as source for detailed measurement data for various code validations and system analysis.

Furthermore, in the near future, it is expected to combine this direct access to the UMLRR data, with the RELAP5 simulation capability in order to show online comparison between simulation and experimental data as a real-time capability for live demonstrations. RELAP5 will be used as a support tool for the UMLRR website. In addition to the comparisons with experiments, a set of severe hypothetical accident scenarios can also be performed with the RELAP5 model in order to provide also online RELAP5 analysis of much severe accident that could not be verified experimentally.

In this paper, a variety of experiments involving normal operating transients were considered for benchmarking the RELAP5 simulation of the UMLRR. Further assessments are also presented by direct comparisons of RELAP5 against previous computational thermal hydraulic analysis using conservative tools such as NATCON [7] and PLTEMP [8] when experimental data are not available.

The code PLTEMP calculates for steady-state forced convection conditions the flow velocity, fuel plate temperature, coolant temperature, as well as the margins to boiling crisis and flow instability. The core can be described either via a single hot channel associated with a single hot plate, or via the so-called subchannel description. In the latter case, the core is described by a cluster of parallel, non interacting channels connecting two plenums, which can have different characteristics. The subchannel description can accommodate up to 150 fuel elements with up to 30 channels each, and up to five different types of non-fuel flow paths (*i. e*. bypasses). With the subchannel modelling approach the flow distribution in the core can be determined. Instead, for safety calculations the single channel approach is usually implemented, by considering the limiting (hottest) channel in the core.

The code NATCON has been written to analyze the steady-state thermal-hydraulics of plate-type fuel in a research reactor cooled by natural convection. The reactor core is immersed within a pool of water that is assumed to be at a constant temperature. The flow is determined iteratively from the balance between buoyancy and friction. The code computes coolant flow rate, axial temperature distributions (in both coolant and fuel) and the margin to ONB (Onset of Nucleate Boiling). Hot channel factors may be introduced for determining safety margins.

REACTOR DESCRIPTION

The UMLRR is a 1 MW, light-water moderated and cooled, graphite-reflected, open-pool type research reactor that has been in operation since January 1975. The primary use of the reactor is to provide a neutron source for various nuclear related education and research activities. The reactor core consists of an array of fuel elements, reflector elements, control blades, the regulating rod, the startup source, nuclear instrumentation, radiation baskets, and centralized flux trap. Four control blades control core reactivity and the regulating rod controls fine power-level adjustments. The startup source generates neutrons to provide accurate nuclear instrumentation indications during reactor startup.

Various experimental facilities are available to subject samples to the neutron radiation generated by the reactor including the radiation baskets, the beam ports, the thermal column, and the fast neutron irradiator (FNI).

Core configuration

The reactor is housed in a 10 meters deep pool filled with about 280 m^3 of high-purity water. The UMLRR core contains a 9 7 grid of fuel assemblies, graphite reflector elements, radiation baskets, voided lead boxes, and corner posts. It is suspended about 8 m below the surface of the pool. An aluminum grid plate and thin aluminum core box are part of the core support structure. Four large control blade assemblies are used for gross reactivity control and for flux shape adjustments. A low-worth regulating rod for fine reactivity control and an external neutron source are also located in the core grid.

The reactor is surrounded by a large pool of demineralized water on the top and bottom and on two sides, and large graphite thermal column on the remaining sides. The core contains 19 full fuel assemblies and 2 partial assemblies arranged roughly in the center of the grid. Directly in the middle of the core is a central irradiation zone known as the flux trap. The flux trap is similar to a radiation basket. The three radiation baskets just to the left of the fuel are used as sample holders, and the remaining baskets simply act as water reflectors. Filling out most of the remaining positions is a series of 8 8 cm graphite reflector elements.

The fuel assembly is roughly 7.6 7.6 63.5 cm with 60 cm of active height. Each fuel assembly used low enriched uranium silicide fuel (LEU) in 16 plates, with two end plates containing pure aluminum. The meat of LEU fuel is an U_3Si_2-Al alloy. The U_3Si_2 contribution is about 67 w/o. The uranium in the LEU fuel is enriched to about 20 w/o ^{235}U. Each plate contains 12.5 g of ^{235}U. Figure 1 illustrates the dimension of a fuel assembly and a fuel plate respectively.

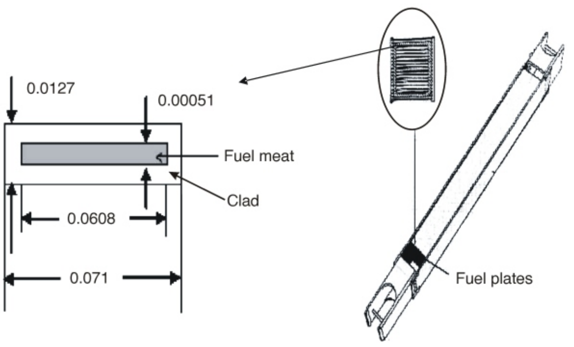

Figure 1. Fuel plate geometry (unit in meter)

Balance of the UMLRR system

The balance of plant system consists of two major systems: a primary coolant system and a secondary coolant system. Figure 2 shows a simplified flow chart of the UMLRR primary and secondary system.

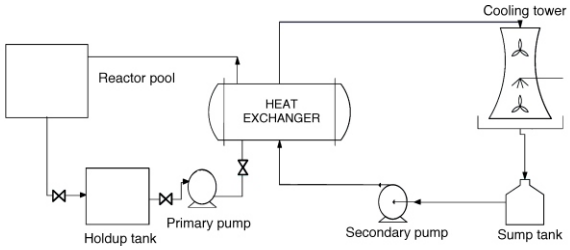

Figure 2. Schematic diagram of the cooling system of UMLRR

The UMLRR was designed so that the water will flow from the top to the bottom of the core during forced convection and *vice versa* under the natural convection mode. When the primary pump is on, there is a high coolant flow going through the primary piping, which will force the small natural convection gate on one side of the duct to close, generating a closed loop between the lower plenum and primary piping system. On the other hand, the natural convection gate was designed to be open due to the buoyancy of the gate when the pump is turned off and there is no suction force in rectangular duct to force coolant flow though the primary piping. When the gate is open, the coolant then can flow through the gate and into the core. In the forced convection mode, the coolant is forced to flow downward into the core using the pump suction force. But during the natural convection mode, core cooling is maintained by natural convection, flow moving upward through the core.

The primary coolant system transfers heat from the reactor to the secondary coolant system at the heat exchanger. It is used to remove heat and maintain core temperatures below a fixed limit. Transfer of heat by natural convection to the primary system water is allowed for power levels up to 100 kW. Above 100 kW, forced convection cooling is required. The primary system coolant is moved through the reactor core, pass through a hold up tank to decay ^{16}N and ^{19}O isotopes, and then through a heat exchanger and back to the core in a closed system. The secondary cooling loop system allows the system to transfer heat from the primary coolant through the heat exchanger to the atmosphere at the cooling tower. A substantial amount of makeup water is required to replenish the resulting loss to the atmosphere. The water is held in a sump

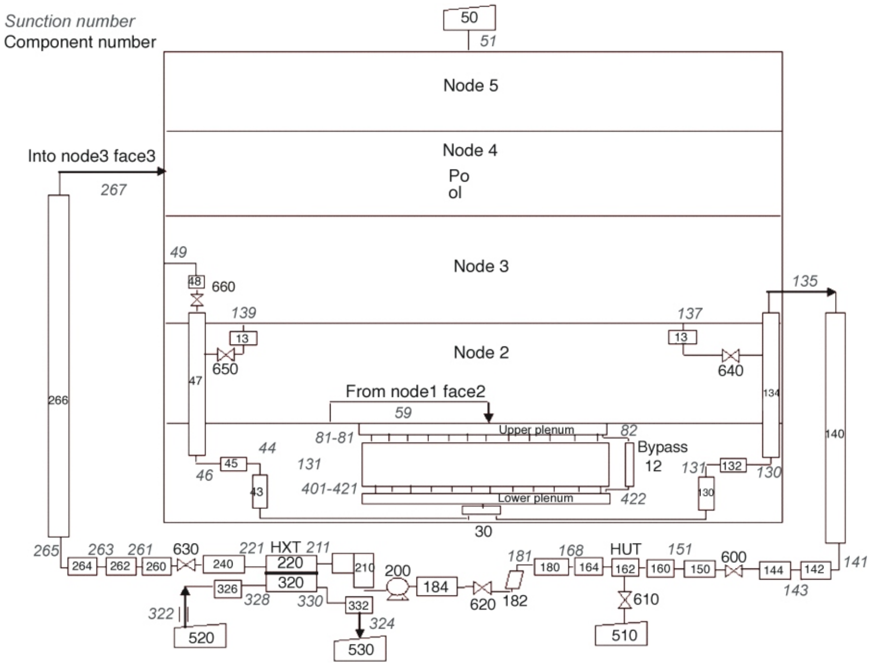

Figure 3. UMLRR nodalization

tank of 15 m^3 located under two cooling towers. The secondary cooling water is neither activated by direct contact with the reactor core, nor contaminated by mixture with primary coolant in the heat exchanger. The secondary cooling loop includes the heat exchanger and temperature control valve, the cooling tower with cooling tower basin, a secondary pump, and sump tank. The secondary water is drained from the sump tank, pass the secondary pump, then through the heat exchanger, pass the heat exchanger, and out to the cooling towers.

NODALIZATION AND MODELING

The UMLRR RELAP5 nodalization was developed to reflect great detail according to the nodalization methodology developed at the University of Pisa [9]. Therefore, to achieve a reliable nodalization the following items should be fulfilled [10]:

(1) The nodalization should have a geometrical fidelity with the involved plant,

(2) The nodalization should reproduce the nominal measured steady-state condition of that plant. In the steady-state level, the operational parameters of the simulated system should fulfill the acceptability criteria as outlined in [9], and

(3) The nodalization should get a satisfactory behavior in time-dependent conditions of any test or operational transients of the nuclear plant. In fact, the demonstration of the nodalization quality at the steady-state level does not ensure that the prediction of a transient scenario is "phenomenologically" correct or even that the nodalization (input deck) is free of errors. Errors can be part of an input deck that has been qualified at the "steady-state" level. The on-transient nodalization qualification process is demonstrated through the capability to correctly predict relevant phenomena and transient scenarios of the facility being simulated.

Putting together a computational model for the reactor system required accurate information about the geometry and material composition of the actual system. The information about the geometry of the reactor system was gathered from the UMLRR blueprints while thermal hydraulics and neutron kinetics data were obtained form various sources of information including the Final Safety Analysis Report or FSAR and from previous thermal hydraulics and reactor physic analysis. Figure 3 shows the developed layout nodalization of the UMLRR. This RELAP5 model consists of a several hydrodynamics components and heat structures representative of thermal hydraulic systems, reactor pool, core region, primary system, holdup tank, primary pump, and heat exchanger. Table 1 shows the name and type of hydrodynamics components associated with the components in the UMLRR. Main components of the UMLRR nodalization are illustrated in tab. 2.

In the RELAP UMLRR model, the reactor pool was modeled using a pipe component divided into five nodes. A time dependent volume was used to simulate the atmospheric pressure on pool surface. Upper plenum and lower plenum were modeled right above and below the reactor core, respectively. Each of the 21 fuel assemblies (19 fuel assemblies and 2 partial fuel assemblies) were modeled separately and 21 heat structure components were used and associated with their corresponding hydrodynamic pipe components (fuel assemblies).

Table 1. Main component for the nodalization

UMLRR component	RELAP component number	RELAP component name
Core	101 to 121	chan01 to chan21
Natural convection valve	640 and 650	V.NC1 and V.NC2
Primary side heat exchanger	220	HEX-1
Secondary side heat exchanger	320	HEX-2
Reactor pool	50	POOL
Bypass	122	bypass
Hold up tank	160, 162, and 164	HUT1, HUT_br and HUT2
Primary pump	200	Pr.Pump

The bypass area in the core takes into account the presence of the central flux trap and the radiation baskets of this particular core configuration.

The natural convection system consists of two different channels associated with the actual physical system. Both coolant gates are open during natural convection, allowing the water from the pool to flow in or from the reactor core. These gates are closed during forced convection.

The primary piping system was modeled with a series of pipe components as shown in fig. 3. The hold up tank was modeled in order to simulate the delay of the primary coolant for about 90 s before entering the primary pump.

The secondary system was modeled using two time dependent volumes and pipe components. These time dependent volumes were used to substitute the whole secondary loop circuit, where detailed modeling is not important for the current framework. The primary and secondary systems of the UMLRR are connected through the heat exchanger number 800.

The point kinetics model was used in the current model. A detailed representation of each assembly is, however, essential to properly take into account the radial power distribution associated with location of the fuel assemblies. The radial power distribution shown in fig.4 was computed with VENTURE code [11]. The axial power distribution, a bottom peak chopped cosine profile, was also given from previous calculation performed considering the blade at 38 cm withdrawn from the core [12]. Although the above modeling procedure is approximate, it is used here to maintain the actual axial and radial power distribution fixed.

MODEL VALIDATION

Once the nodalization settled, the first step concerns the assessment of the RELAP5 response to steady-state conditions through the comparison of the calculated parameters with experimental data. In a second step, transient conditions are simulated with the RELAP5 on the adjusted nodalization.

Table 2. UMLRR characteristics

CORE MATERIAL	
Nuclear fuel	U_3Si_2 Al alloy
Fuel element	MTR plate-type
Cladding	Al alloy
Coolant	light water
Moderator	light water
Reflector	graphite
CORE THERMAL-HYDRAULICS	
Core power [MW]	1
Core mass flow rate [kg/s]	71.4
Core mass flow rate per assembly [kg/s]	3.4
Axial peaking factor	1.39
Radial peaking factor	1.45
Operating pressure [bar]*	1.52
Inlet coolant temperature [C]	20-25
Fuel thermal conductivity [W/m C]	105
Cladding thermal conductivity [W/m C]	180
Pressure drop through the core [psi]	0.32
FUEL ELEMENT DATA	
Number fuel assembly	21
full fuel assembly	19
partial fuel assembly	2
Number of plate/fuel assembly	16
Plate width [cm]	7.14
Fuel mear width [cm]	6.085
Plate thickness [cm]	0.127
Fuel meat thickness [cm]	0.051
Total height [cm]	63.5
Active height [cm]	59.69
Fuel meat volume [cm³]	18.524
Total active surface/fuel assembly heated [m²]	1.1623
Water gap between plates [cm]	0.2963
Total core flow area [cm²]	644.0
Bypass (central flux trap) flow area [cm²]	2.77E-02**
CORE KINETICS	
Effective delayed neutron fraction	0.0078
Prompt neutron generation time [s]	6.45E-05
Void feedback coefficient [$/%void]	3.03E-01
Doppler feedback coefficient [$/ C]	1.92E-03
Coolant temperature feedback [$/ C]	6.15E-03

* 1 bar = 10^5 Pa
** 2.77E-02 read as 2.77 ·10

Steady-state

The validations of the RELAP5 nodalization pass through the demonstration that the RELAP model reproduces the measured steady state conditions of the UMLRR with acceptable margins.

	1	2	3	4	5	6	7	8	9
A									
B				0.039	0.044	0.042			
C			0.043	0.057	0.047	0.061	0.050		
D			0.045	0.060		0.065	0.052	0.011	
E			0.044	0.057	0.047	0.062	0.050		
F				0.039	0.043	0.042			
G									

Figure 4. Radial power distribution

For this purpose, a number of parameters are selected for comparison with measured data. Two steady-state experiments under forced and natural convective regimes were performed. The data were recorded and processed through the CDAS system. The information provided by CDAS included power, temperatures, flow rate, pump status, *etc.*

Forced convection steady-state

In this case, the reactor operated at steady-state of about 900 kW with primary pump turned on (forced convection mode) as recommended in the FSAR. The water is forced to flow downward in to the core with a pump flow rate of 360-380 m^3/h.

A RELAP simulation of the reactor was run for about 100 s in order to obtain stable calculation results. Table 3 shows the key parameters from the RELAP5 simulation and the measured data from the experiment. The calculated results show good agreement with the measured data and the deviation in the worst case did not exceed 6%.

Table 3. Comparison of steady-state forced convection mode

	REALP	Experiment	Error* [%]
Power [MW]	0.9	0.88-0.92	2.2
Pool inlet temperature [C]	23.91	23.66	1.0
Core inlet temperature [C]	23.01	22.70	1.3
Core outlet temperature [C]	25.12	24.70	1.7
Delta core temperature [C]	2.11	2.0	5.5
Primary flow rate [gpm]	1650	1700	2.9
Primary pump	ON	ON	-
Steady-state	yes	yes	-

* Error = (RELAP-Experiment)/Experiment

The pool inlet, core inlet, and core outlet temperature obtained from RELAP5 were slightly overestimated by almost 1 °C. These little discrepancies are negligible but could be explained by the fluctuations of the measured experimental power and to errors in the measurement of temperature, which is around 1 °C. Furthermore, using several positions for power detectors in the reactor, the range of the power measured varied from 880 to 920 kW whereas in case of RELAP5, the steady-state power was fixed to 900 kW corresponding to the average value with a deviation of 4.4%. The temperature increase in the core was about 2 °C. This latter fact demonstrated the suitability of the adopted pool volume model used in the current nodalization.

Steady-state natural convection

In a second step, the natural convection operation mode is calculated by RELAP5. In this case, the primary pump is turned off and the coolant flow though the reactor core depends only on the core temperature difference. The operating power under natural convection is fixed at 80 kW. Figure 5 shows the mass flow rate through the fuel assembly as calculated by REALP5. One can observe that the flow stabilize after 100 s with a negative value of the flow rate that denote an upward direction of the flow corresponding to the natural convection mode. Unfortunately, at this stage, it is not possible to compare the value of the mass flow rate since the UMLRR instrumentation does not have the capability for such purpose. This kind of instrumentation is expected to be installed in the future.

Therefore the code-to-code comparison was considered using the NATCON code. Figure 6 shows the results of the two codes for the axial clad tempera-

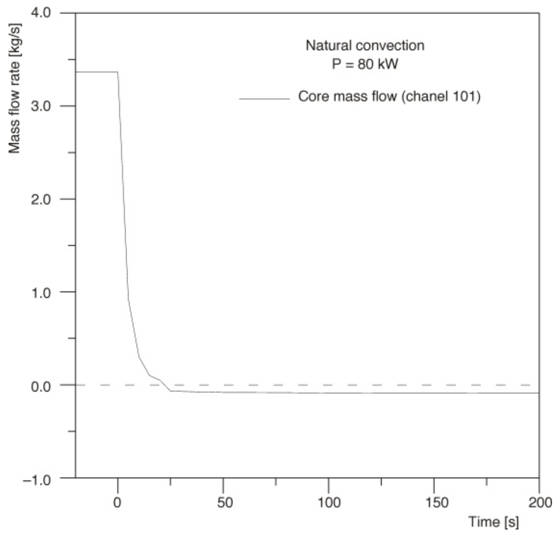

Figure 5. Fuel assembly mass flow rate

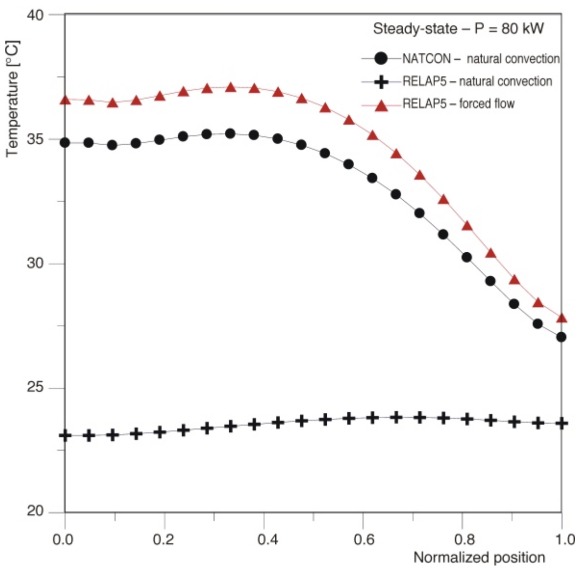

Figure 6. Clad temperatures for natural convection mode

ture profiles where some minor deviations are observed. The clad temperature as calculated by RELAP5 is underestimating the NATCON data by 3 °C whereas it overestimated at the top by 2 °C. These deviations are expected to be due to the heat transfer correlations used in each code for natural convection regime. As a consequence, the total temperature increase across the core computed by RELAP5 is 10 °C while it was 15 °C in case of NATCON.

Nevertheless, since the deviation of the RELAP5 from the experimental data were shown to be less than 6% and since the comparison with some referenced channel codes has shown acceptable agreement, we can consider that experimental uncertainties bounds completely the calculation discrepancies which are quite acceptable. Accordingly, at a first glance, the developed nodalization maybe considered representative of UMLRR operating modes (forced and natural convection) notwithstanding the fact that not all the nodalization assessment criteria, as reported in [9] and [10], are considered and fulfilled.

Transient analysis

In a second assessment step, a series of transient situations were managed experimentally and recalculated by RELAP5. The benchmarking analysis considered in this part are also collected database of the CDAS UMLRR system.

Analysis of pool heat-up

The purpose of this section is to assess the thermal-hydraulics model of the RELAP5 code by recalculating the whole system heat-up. For this case, the nodalization used includes only the primary piping system and the reactor pool.

The RELAP5 simulation was performed using data of the experiment performed on June 21, 2005, as part of the nuclear101.com demonstration website. The basic idea of that experiment was to operate the reactor at critical constant power when the secondary circuit is disabled. The primary pump was turned on whereas the secondary pump was turned off in order to induce an increase of the pool temperature. In this experiment, the four control blades were positioned to make the reactor critical. The regulating rod was set in auto mode to control and compensate any reactivity changes in the core in order to keep the reactor critical at the fixed power. Key parameters used for this case are summarized in tab. 4. The calculations were run with the primary and secondary systems turned on for about 400 s to establish the steady-state condition. Then the secondary system was shut-off by setting the flow rate through the secondary side to zero. The temperature coefficient data for point kinetics model were set to zero since the regulating rod was used to compensate the temperature feedback and to maintain core power level constant.

Table 4. System conditions

Core power [MW]	0.9
Convection mode	forced
Primary system	on
Secondary system	off
Inlet temperature for the pool (1st guess) [°C]	23
Inlet temperature HEX 1st side (1st guess) [°C]	26
Inlet temperature HEX 2nd side (1st guess) [°C]	25
Feedback coefficients	0

During the steady-state condition, acceptable agreement is observed between the calculated and the measured data (up to 400 s) as could be seen on fig. 9 and 10 for core plenums and the hole pool; the discrepancies are less than 1 °C. These discrepancies are completely bounded by the measurement uncertainties.

After the beginning of the transient, the temperatures increase as consequence of no energy removal from the primary since the secondary system was off. All the temperatures are shown to increase gradually at a mean rate of 2.7 °C per hour because of continuous power production. Figures 7 and 8 show temperature evolution at different positions in the pool (core plenums and otherwise in pool) as obtained by RELAP5 and by measurements. In general, the same trends are observed and a good agreement is achieved. Furthermore, we observe that the discrepancies are reduced further a long time after the beginning of the transient; such behavior could indi-

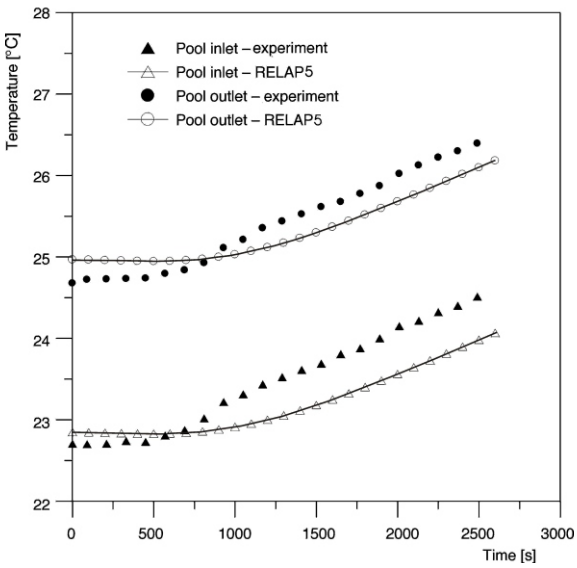

Figure 7. Pool heat-up during the heat-up experiment

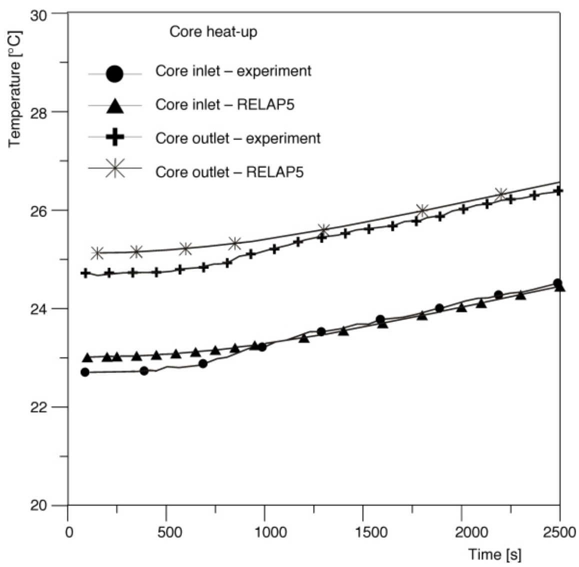

Figure 8. Core (plenums) inlet and outlet temperatures

cate that the calculations are well reproduced "macroscopically" by the RELAP5 model when neutronic interactions are not considered.

Reactivity insertion transient

The reactivity insertion transient (RIT) experiments were performed during normal operating condition of the reactor. However, they give meaningful information about the reactor behavior. In the current framework these data are used for the validation of the RELAP5 UMLRR model in case of both positive and negative reactivity addition to the core.

The experiment was conducted as follow: the reactor was brought critical at 200 kW. After stabilization of all core parameters, the regulating blade was inserted by almost 15.25 cm from 45 to 30 cm into the core to simulate a negative reactivity insertion of approximately 0.141 \$. After about five minutes, the reactor was bought back to critical at 20 kW and the regulating blade was then withdrawn about 7.5 cm out to simulate a positive reactivity insertion of 0.079 \$ from a critical power of 20 kW. For the last step of the experiment, the control blade#2 was dropped in the core in order to simulate a large negative reactivity insertion of 2.19\$.

Since the regulating rod has a maximum speed of 38 cm/mn, the first two experiments could be represented by ramp reactivity insertions (positive or negative). The third experiment could be represented by a step reactivity insertion, however due to the lack of experimental measurements, this case is not reported in the current work. The key parameters that govern the core behavior during these tests are outlined in tab. 5.

Table 5. Reactivity insertion key parameters

Conditions	Case 1	Case 2	Case 3
Initial power [kW]	200	20	200
Primary side	on	on	on
Secondary side	off	off	off
Feedback reactivity coefficients functions	on	on	on
Reactivity change [\$]	0.141	0.08	2.194
Control blade movement	Regulating	Regulating	Blade#2
Time of the movement [s]	21	11	1

The results obtained by the RELAP5 simulations are compared against experimental data in fig. 9, and fig. 10 for the transient described above. Generally, we can observe a good agreement between RELAP5 results and the experimental data even if some discrepancies are observed for positive reactivity insertion. These discrepancy are more pronounced after 200 s. These discrepancies could be explained as follow:

difference between the initial powers conditions (about 11%),

feedback model adopted in RELAP5, which may underestimate the feedback reactivity,

difference of the reactor period (experiment and calculated by point kinetic model). To illustrate this, the experimental and the RELAP5 data are fitted by a function of the form $e^{\omega t}$; the values are ω_{EXP} = 0.00685 and ω_{RELAP} = = 0.00679 and the corresponding periods are

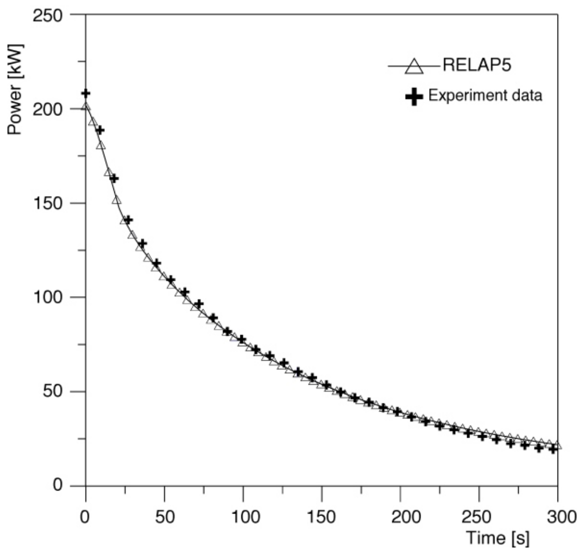

Figure 9. Power distribution of a small negative reactivity insertion

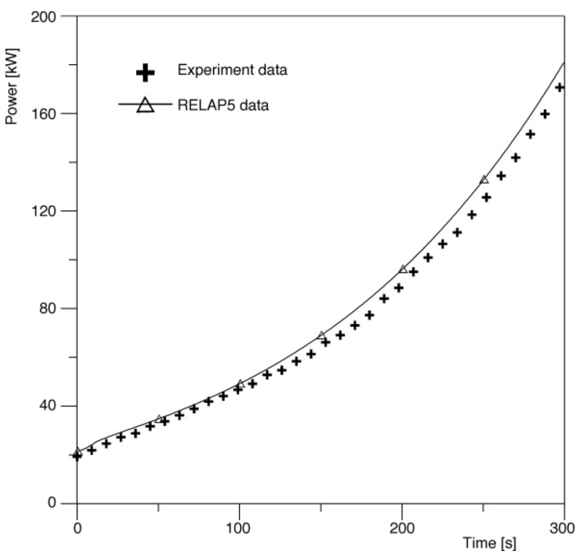

Figure 10. Power distribution of a small positive reactivity insertion

146 s, and 147 s respectively (the reactor period is the defined as $T\ = \omega\ ^{1}$).

Nevertheless the little deviation of the calculated period (about 1%), the deviation of the calculated power may go increasing in time, as observed in fig. 10 because the initial power is different for both cases (11%); however, since the period calculated by RELAP is slightly less than the measured, the discrepancy could be reduced later during the excursion as could be illustrated by the deviation of the two fit-equations as follows:

$$P_{\mathrm{EXP}} \quad 22.74\,\mathrm{e}^{\omega_{\mathrm{EXP}} t}$$
$$P_{\mathrm{RELAP}} \quad 24.45\,\mathrm{e}^{\omega_{\mathrm{RELAP}} t}$$

The mean deviation of these two fitted equations is approximately 6% during the first instant of the transient when the contribution of the feedback could be neglected.

As a conclusion for this part, we can state that the point kinetic model is sufficient to reproduce reactivity transient under normal scheduled experiments *i. e*., far away from the safety margins. For more severe transients scenarios a 3-D kinetics model will be more suitable since a strong effect of feedback is expected.

CONCLUSION

In this framework, a nodalization for the UMLRR research reactor for RELAP5 calculation has been developed. The nodalization was validated against experimental data from both steady state and transient conditions using experimental data retrieved from the CDAS system. The agreement between the RELAP5 results and the measured data was shown and the little discrepancies observed were explained.

The RELAP5 results have been also compared with some data obtained by specific channel codes used up to date for the safety analysis of the UMLRR reactor. This comparison has shown good agreements between the codes and the little discrepancies observed could be explained by the different empirical correlations embedded within each code.

According to this good agreement, the RELAP5 nodalization for UMLRR could be considered representative of steady-state operational conditions and for the range of the transient considered here. The good agreement of the results obtained in this study confirmed the initial idea to use RELAP5 as a supporting tool for the UMLRR website project since the range of experiments are comprised within the postulated events considered here.

The results from this phase combined with the results from the future work will provide both experimental and numerical information, as well as detailed information about normal and off-normal transient phenomena that could occur in research reactors.

However, the utilization of this model for more severe accidents could be suitable for giving at first glance trends information on the behavior of the reactor under postulated RIA, LOFA, and even LOCA. Also, the next step will consider the development of a 3-D kinetics model of UMLRR based on the developed RELAP model with more extensive experimental verification of the model.

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ОЦЕНА RELAP5 МОДЕЛА ЛОВЕЛ ИСРАЖИВАЧКОГ РЕАКТОРА УНИВЕРЗИТЕТА У МАСАЧУСЕТСУ

Релап5 је систем програма развијених у INEEL националној лабораторији у Ајдаху за термохидрауличке анализе нуклеарних реактора, који се веома користи за сигурносне анализе комерцијалних нуклеарних електрана. Међутим, скорије објављена верзија RELAP5/3.2, и наредне, указују на значајне могућности за анализу система истраживачких нуклеарних реактора. Као један допринос оцени RELAP5/3.2 за сигурносну анализу истраживачког реактора, коришћени су експериментални подаци Ловел истраживачког реактора (UMLRR) Универзитета у Масачусетсу. UMLRR је истраживачки реактор базенског типа, снаге 1 MW, модериран и хлађен лаком водом и са графитним рефлектором.

Коришћењем експерименталних података, у овом раду приказан је развој и ваљаност UMLRR-RELAP модела. У ту сврху, изведена је серија експеримената ради стандардизације RELAP5 прорачуна система истраживачког реактора. Као резултат овог проучавања, показано је да је UMLRR нодализација сагласна са експерименталним подацима о понашању реактора.

Кључне речи: истраживачки реактор, термохидрауличке анализе, сигурносне анализе, анализе прелазних стања, RELAP5