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USE OF THE THERMO-HYDRAULIC SYSTEM CODE RELAP5/MOD3.2 TO TRANSIENT ANALYSIS EVENTS OF THE VR-1 REACTOR

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Abstract. The article explores the application of the RELAP5 thermo-hydraulic code for transient simulations on the VR-1 research reactor, focusing on reactor vessel nodalisation and coolant stratification above the core. The study investigates Reactivity-Initiated Accidents (RIAs) at the VR-1 reactor, triggered by various initiating events such as inadvertent control rod withdrawal or experimental channel flooding. By comparing several nodalisation models for protected and unprotected RIAs, the study discusses the impact of the nodalisation approach. Key findings highlight the significant influence of user effects and reactor vessel nodalisation on the accuracy and reliability of these predictions. The article also examines two-phase flow flashing instability and the reactor SCRAM delay effect. The conclusions drawn from different nodalisation approaches may offer insights into optimising research reactor safety analysis and the simulation of natural circulation systems.

Keywords: RELAP5, natural circulation, VR-1 research reactor, transient simulation, RIA.

1. Introduction

Computer codes are crucial for the design, operation, licensing, and safety assessment of nuclear reactors. The development of new computational codes, along with the availability of experimental data and model validation, has enabled the use of best-estimate (BE) system codes instead of conservative ones. Conservative codes, while historically used, cannot simulate complex scenarios and often yield overly simplistic results. In contrast, BE codes provide more realistic simulations, with international trends in safety analysis and offer better evaluation of reactor operation. However, system codes' limited ability to simulate spatially dependent natural circulation leads to significant simplifications and assumptions, potentially resulting in non-physical solutions. The structure of user-defined models (e.g., nodalisation) can significantly affect results (often referred to as the user effect). While BE codes improve the adaptability and accuracy of safety analysis, managing challenges such as user effect and natural circulation is essential. This work aims to simulate various transients on the VR-1 training reactor using the RELAP5 system thermal-hydraulic (SYS-TH) code and analyse the impact of reactor vessel nodalisation [1].

2. Description of VR-1 research reactor

Located at the Czech Technical University in Prague, the VR-1 reactor is a pool-type, light water-cooled, and moderated research reactor. This "zero-power" reactor has a nominal power of 100 W and is cooled via natural circulation. The pool type arrangement assures quick and simple access to the reactor core, easy insertion and extraction of various experimental samples and detectors, and simple and safe manipulation of fuel elements. The height of the water column from the fuel inlet to the water level is 3.757 m (2.675 m from the core outlet). With an overall pool coolant volume of 17 m^3, the VR-1 reactor's geometry is depicted in Figure 1. The reactor utilizes IRT-4M tubular fuel (see Figure 2) composed of UO_2 enriched to 19.7 %. Within the IRT-4M cell, the fuel layer comprises a dispersion of Al and UO_2, with a thickness of 0.7 mm, encased in Al cladding made of SAV-1 alloy. Fuel assemblies have a total height of 0.88 m, with an active height of 0.68 m. Each fuel cell can consist of 8, 6, or 4 concentric tubes, forming separate flow channels [2].

Since the SYS-TH uses control volumes with limited geometry, several assumptions and simplifications must be made. The simplified scheme of the VR-1 reactor is depicted in Figure 3.

Certain structural components and experimental devices, such as radial and vertical experimental channels, power measurement detectors, and fuel dummies, are excluded from the model. The reactor core is simplified using several parallel channels. The reactor vessel's geometry is divided into three axial sections, while the overall coolant volume and water column height are preserved. In contrast to CFD simulations, RELAP5 solves 1D-flow equations; therefore, the model cannot accurately capture spatial recirculation and coolant stratification above the core. This limitation also impacts parameters such as the volumetric flow through the core and the cladding temperature during transients. A significant challenge is to

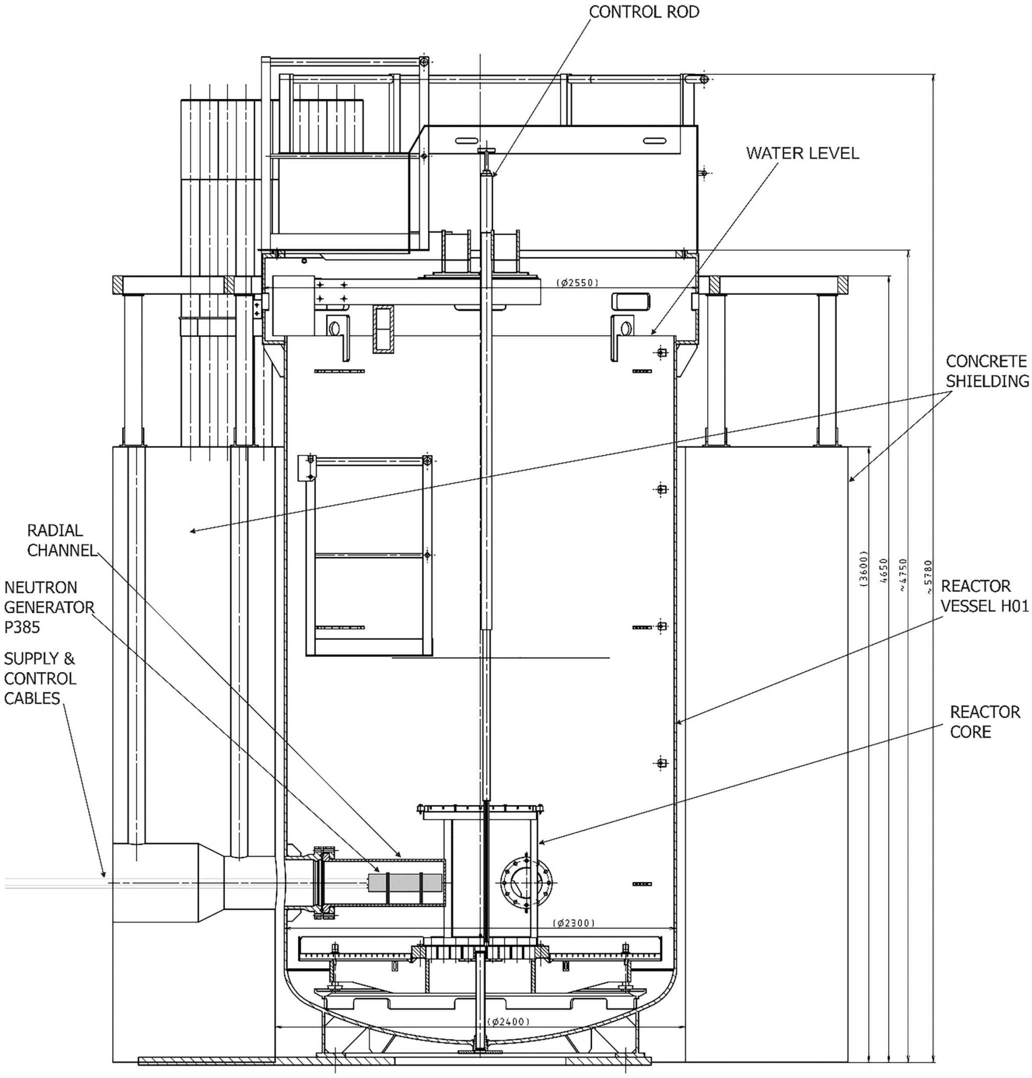

FIGURE 1. VR-1 reactor geometry [3].

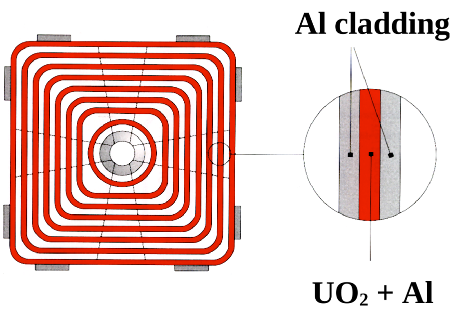

FIGURE 2. Top view of 8-tube IRT-4M fuel assembly [4].

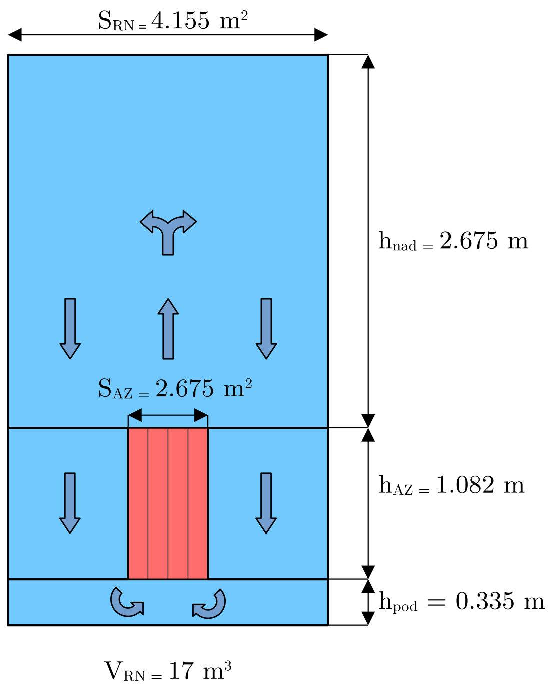

FIGURE 3. Scheme of the VR-1 simplified geometry.

create nodalisation and artificial recirculation loops to achieve satisfactory results. Various approaches are presented and analysed in this paper.

3. DESCRIPTION OF CODE RELAP5/MOD3.2

The Reactor Excursion and Leak Analysis Programme (RELAP) is a thermo-hydraulic code widely used for the analysis of LOCA, LOFA, or RIA accidents of

Parameter	Value
Reactor nominal power	100 W
FA Average power	6.25 W
Heated channels	16
Axial power distribution	sinusoid
Initial coolant temperature	20 °C
Maximum channel power ratio	1.4
Inner/outter FA radius	4.0168 / 4.0328 mm
Doppler coeff.	−2.05 pcm K^{-1}
Coolant temp. coeff.	0.13 pcm K^{-1}
Coolant dens. coeff. (20–50 °C)	−12.16 pcm K^{-1}
Coolant dens. coeff. (50–100 °C)	−22.86 pcm K^{-1}

Table 1. VR-1 reactor model parameters.

energetic reactors. It is based on the two-fluid, non-equilibrium, non-homogenous, hydrodynamic model for transient simulations of the 1D-flow. Heat generation in the system is defined by point kinetics equations with constant power spatial distribution, making it solely time-dependent. Implemented correlations and models are used to determine the flow regime, heat removal, sub-cooled boiling, and pressure drop in the heated channel [5].

4. RELAP5 model of VR-1 reactor

The RELAP5 model of the VR-1 reactor comprises 16 parallel separated channels without radial cross-flow: one hot channel with maximum power and the remaining channels with average power. The tubular IRT-4M fuel with 8 tubes has been simplified into a single heat structure while maintaining the heated diameter, thickness of the fuel tubes, and heat exchange surface. A time-dependent volume simulates the boundary condition of the reactor pool water level. General input parameters for the VR-1 reactor model are provided in Table 1. The SCRAM of the reactor is initiated when power change rate of 6 % is reached. Apart from the RELAP5 simulations, neutronic calculations using Serpent 2 and the ENDF/B-VIII.0 library were performed to determine the input parameters for the thermo-hydraulic model. Kinetic parameters, reactivity feedback coefficients, and initiating events were evaluated and averaged for several core configurations. The values of the average reactivity coefficients are shown in Table 1. Using the envelope method, initiating events for the VR-1 reactor were postulated (two of them will be presented in this work).

Several models with different reactor vessel nodalisations are presented in this work. Model A, the simplest one, is illustrated in Figure 4. The reactor core design remains the same for each model. Pipes 40 and 166 are collectively denoted as Free Coolant Volume (FCV), with pipe 40 specifically as the Stratified Volume (SV). For model A, the reactor vessel is represented by a single loop (see Figure 5). Models B, C, and D nodalise the FCV using 2, 3, and 4 vertical channels, respectively, while maintaining the overall

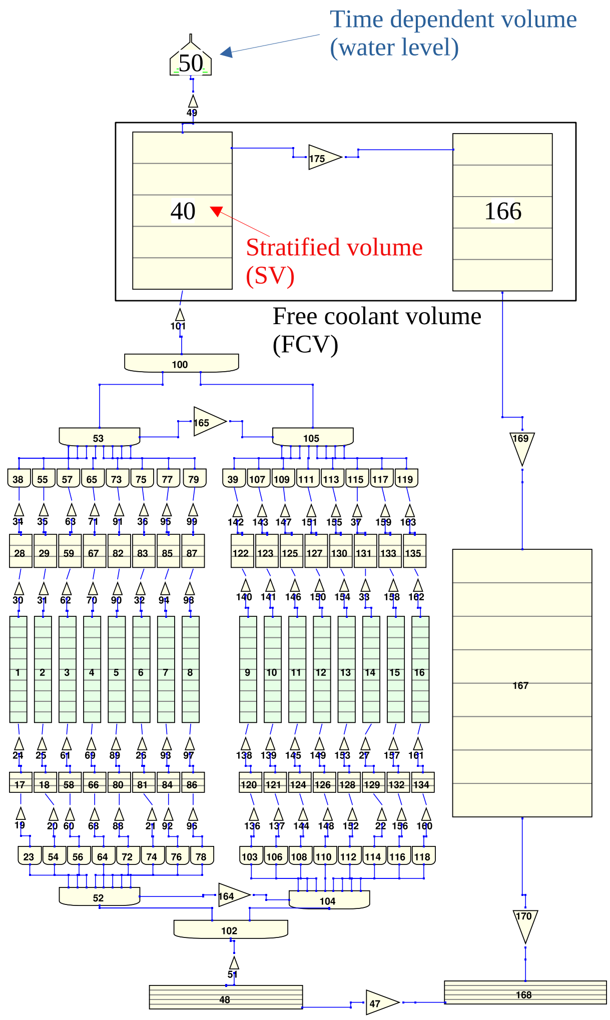

Figure 4. Reactor VR-1 model A - RELAP5.

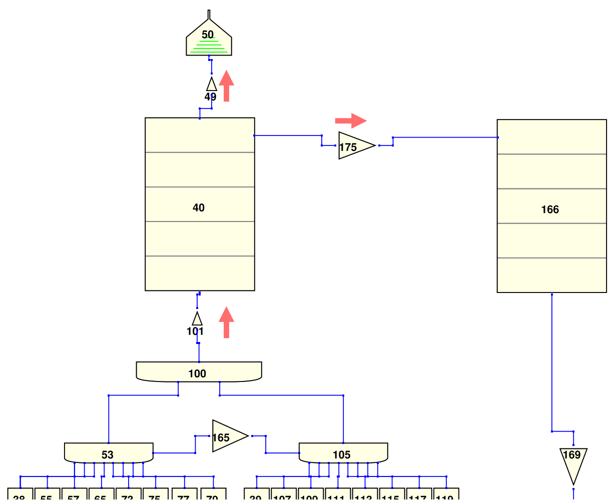

Figure 5. VR-1 reactor model A – recirculation scheme – RELAP5.

volume (see Figure 6, 7 and 8). The main difference between model A and models B to D is the lack of possible recirculation in the FCV. Model E also does not allow the recirculation and aims to minimize the stack effect in the FCV by merging both components

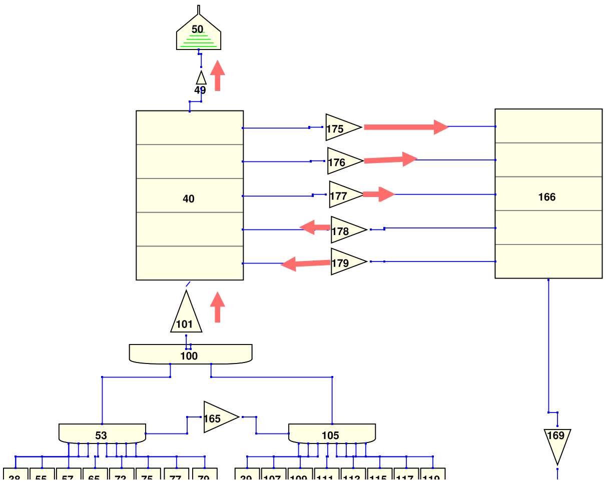

Figure 6. VR-1 reactor model B – recirculation scheme – RELAP5.

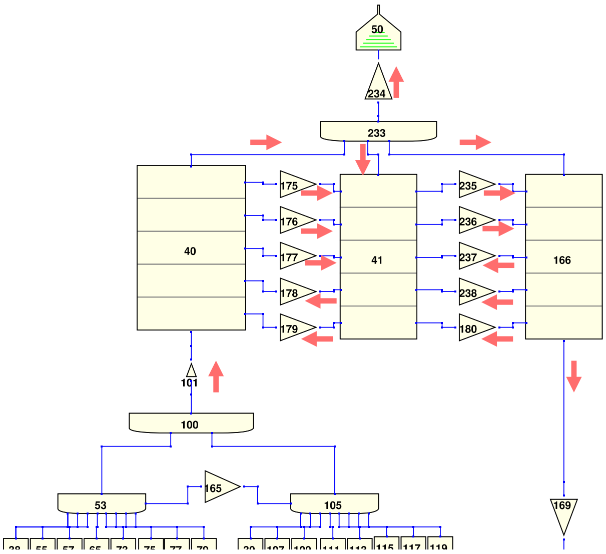

Figure 7. VR-1 reactor model C – recirculation scheme – RELAP5.

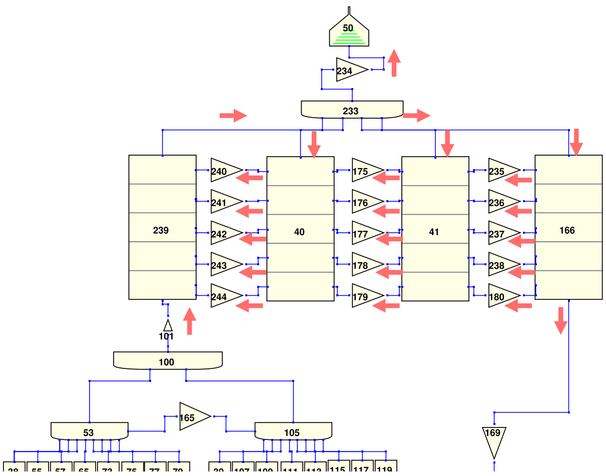

Figure 8. VR-1 reactor model D – recirculation scheme – RELAP5.

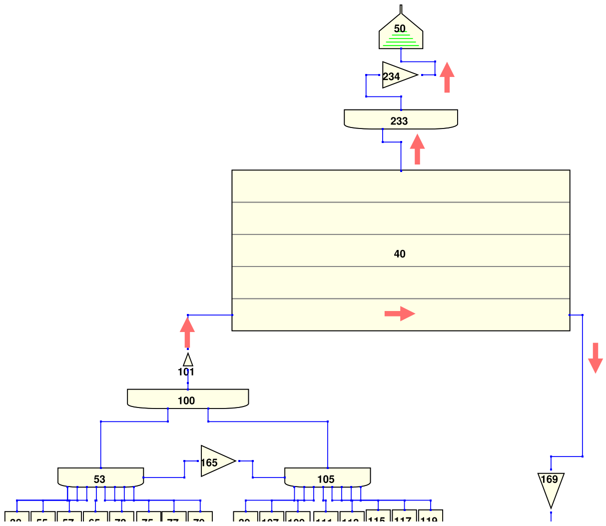

Figure 9. VR-1 reactor model E – recirculation scheme – RELAP5.

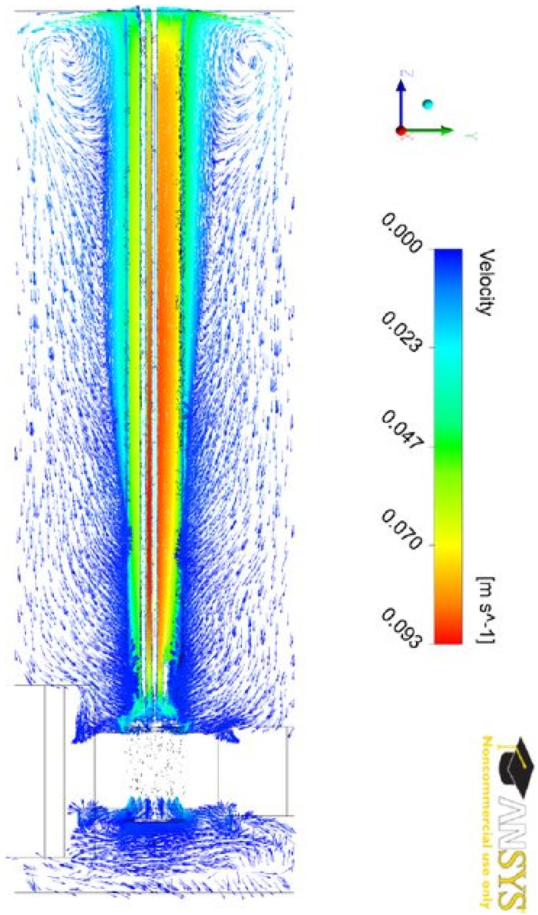

Figure 10. Velocity field of JSI TRIGA Mark-II pool reactor – CFD calculation [6].

into one pipe and redirecting the core outlet flow while maintaining the hydrostatic pressure, see Figure 9.

To illustrate the complexity of natural circulation in a pool-type research reactor, the velocity field for the natural circulation regime in the TRIGA Mark-II reactor is presented in Figure 10. The similarity between the TRIGA and VR-1 reactor vessel geome-

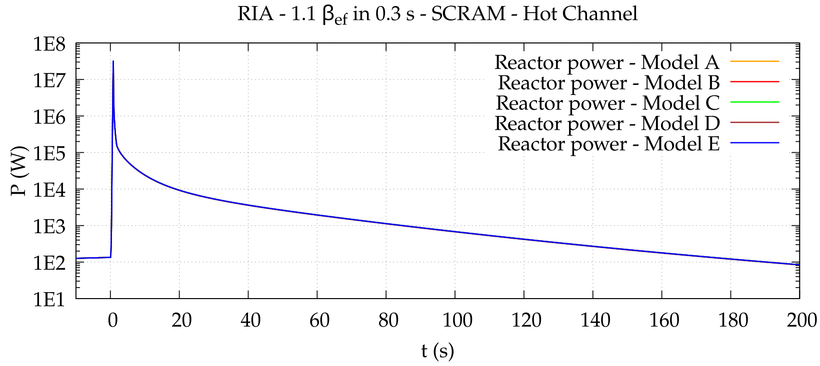

Figure 11. Reactor power during RIA accident (1.1 β_{ef} in 0.3 s with SCRAM) for different models.

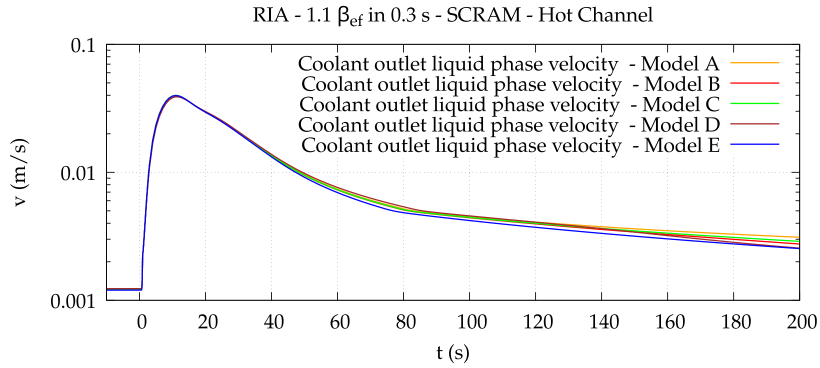

Figure 12. Coolant outlet velocity during RIA accident (1.1 β_{ef} in 0.3 s with SCRAM) for different models.

tries justifies assuming similar flow patterns. None of the RELAP5 models can recreate the 3D flow patterns and recirculation loops in as much detail as the CFD calculations. As the coolant leaves the core outlet, it rises in a conical direction with radial stratification. The aim of this study is to evaluate how different nodalisations affect system behaviour during transients and how this might impact safety analyses performed on research reactors.

5. Transient calculation

Due to the design and low power of the VR-1 reactor, both LOFA and LOCA accidents are not significant and do not call into question the nuclear safety of the facility. The absence of pumps and negligible decay power prevent any potential damage to the nuclear fuel that the transients might cause. The only relevant transient for the VR-1 reactor is an RIA accident. There are several initiating events postulated and analysed in the following text.

5.1. Flooding of the experimental channel

The VR-1 reactor design includes vertical experimental channels inserted at different positions in the reactor core. These channels are used for irradiation and measurement, and allow the introduction of different samples or detectors. In the event of a wall rupture at the bottom of the channel, the tube will fill with water, increasing the reactivity through improved moderation. Using the envelope method, an initiating event for RIA accident of 1.1 β_{ef} in 0.3 s was determined. The SCRAM of the reactor is initiated when the power change rate of 6 % is reached.

As the transient begins, the power rises, reaching its maximum at 0.85 s (see Figure 11). The SCRAM leads to an immediate power decrease, with delayed fission power returning to nominal level after 180 s. The coolant velocity at the outlet of the maximum channel reaches its peak at about 10 s. As natural circulation develops for each model, the results start to differ slightly after about a minute (see Figure 12). In the initial seconds of the transient, all models give similar results in terms of acceptance criteria parameters. This applies to all transients with SCRAM, including events such as rod withdrawal and other analysed accidents.

Effect of SCRAM delay Simulation of research reactors using SYS-TH codes requires numerous assumptions and estimates of key parameters. The effect of assumed input parameters causing a variety of results is called an user effect. One of the key parameters is the SCRAM delay, the time interval between the detection of the SCRAM signal and the start of the control rod movement. This time interval can have significant impact on the outcome of the initiating accidents. Figures 13 and 14 show the reactor power and cladding temperature at the core centre during a PIE of vertical channel flooding with different delay assumptions for the control rods' insertion. For the delay of 0.8 s, the cladding temperature acceptance

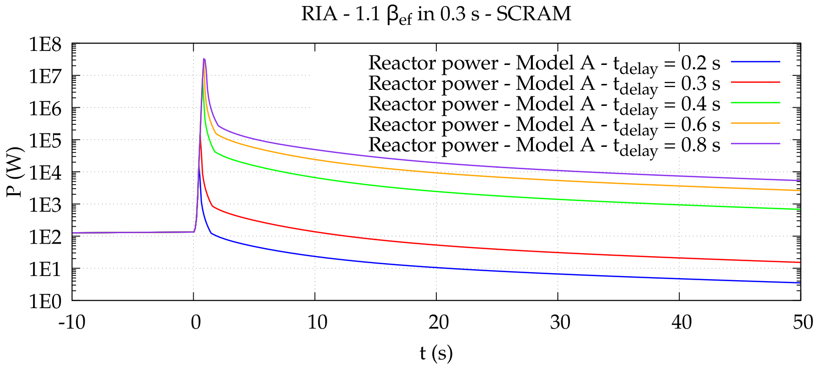

Figure 13. Reactor power for different SCRAM delay (RIA 1.1 β_{ef} in 0.3 s) – Model A.

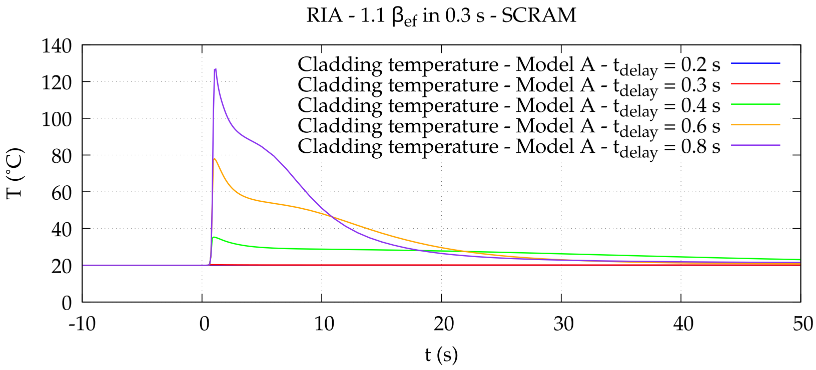

Figure 14. Cladding temperature for different SCRAM delay (RIA 1.1 β_{ef} in 0.3 s) – Model A.

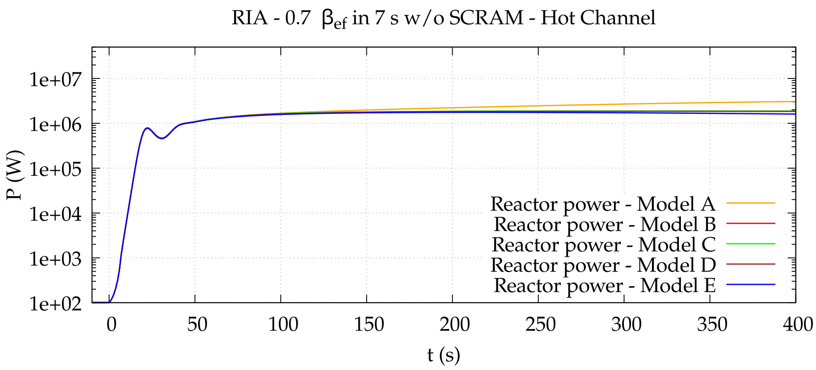

Figure 15. Reactor power during RIA accident (0.7 β_{ef} in 7 s without SCRAM) for different models.

criteria ($t_{clad} < 98\,°C$) are broken. Models with the 0.6 s delay are used for all design basis events with reactor shutdown, as it allows for a clearer comparison of transient states without breaching the acceptance criteria. The real delay is supposed to be much less than 0.1 s, therefore, the value of 0.6 s is very conservative.

The graphs indicate that a 0.8 s delay could lead to surface boiling, while the delay of 0.3 s might not cause any significant heat up or damage to the cladding. This shows a significant user-effect due to fast transient behaviour of research reactors.

5.2. Inadvertent control rod withdrawal during operation

Postulated initiating event of inadvertent control rod withdrawal is based on the VR-1 operational documentation, namely the VR-1 reactor limits and conditions [7]. For each core configuration, no control rod worth can excess 0.7 β_{ef} and reactivity introduction velocity of 0.1 $\beta_{ef}\,s^{-1}$. This implies a possible slow RIA accident 0.7 β_{ef} in 7 s. For the design basis accidents when SCRAM is assumed, the power excursion remains negligible. In the following text, it is assumed that the reactor shift will take such steps that it will not be possible to shut down the reactor, i.e. to insert control rods into the reactor core. The reactor power, cladding temperature, and coolant velocity is depicted in Figure 15, 16, and 17.

The reactor behaviour during an unprotected rod withdrawal accident can be divided into two distinct phases: *independent* and *interdependent*. During the independent phase, which lasts for approximately one minute, the reactor's behaviour is solely defined by its

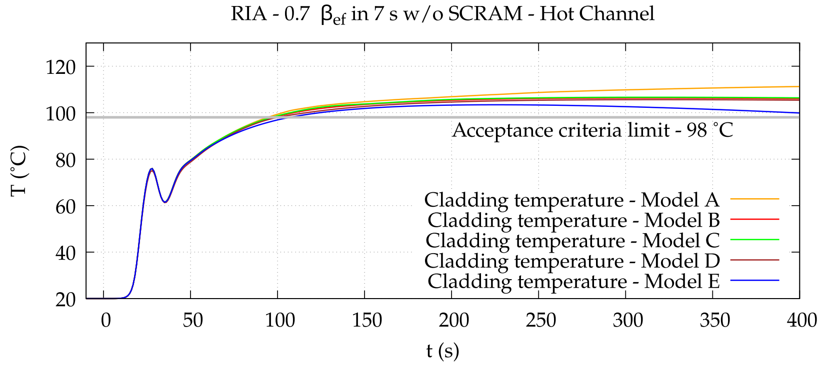

Figure 16. Cladding temperature in the centre of the fuel assembly during RIA accident (0.7 β_{ef} in 7 s without SCRAM) for different models.

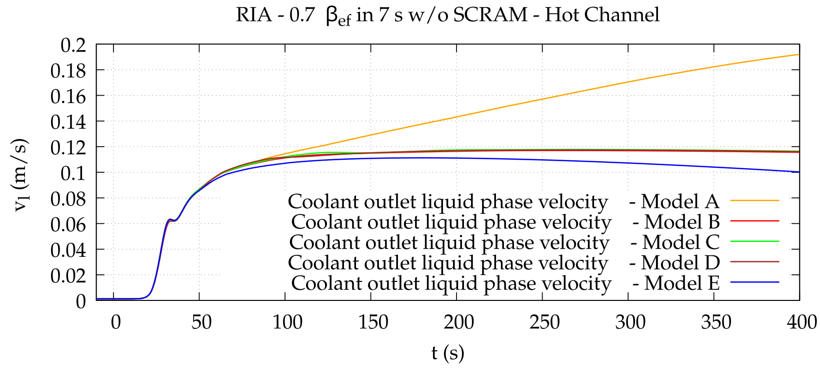

Figure 17. Coolant outlet velocity during RIA accident (0.7 β_{ef} in 7 s without SCRAM) for different models.

kinetics and feedback mechanisms, which are identical across all models. The nodalisation of the free coolant volume above the core starts to play a crucial role in shaping the model behaviour when natural circulation fully develops. This phenomenon applies universally to all transients and models.

For model A, the coolant flows through the stratified volume (SV) – pipe 40 – which serves as a 2.675 m long riser. The lack of mixing of the heated coolant with the cold water in the SV creates a high temperature & density difference at the SV inlet and outlet increasing the buoyancy forces. As can be seen in Figure 10, with the CFD calculation of the TRIGA reactor, the main flow direction remains vertical. This justifies neglecting the radial mixing and using the stratified volume (pipe 40) as a riser. The nodalisation of the model A leads to the higher flow, which, due to the negative coolant feedback, results in the higher power, and therefore higher cladding temperature (see Figures 15 and 16). It is important to notice the strong interconnection of the power and the flow for natural circulation systems.

The behaviour of models B, C, and D shows similarities. All three models allow for recirculation and cross flow in FCV to occur with varying stages of complexity (2, 3, or 4 vertical pipes nodalisation). The horizontal cross flow is depicted in Figures 6, 7, and 8. In comparison with model A, the temperature gradient in the stratified volume (SV) shows to be lower, leading to a lower circulation velocity. The horizontal flow in the FCV junctions appears unrealistic and does not replicate the behaviour observed for CFD calculations. The momentum vector created by heating up the coolant in the core can be broken down into the several oposite vectors in the FCV junctions – this causes RELAP5 to converge to unrealistic values. The flow through the core remains constant, but the velocity in the FCV junctions increases.

For model E, the hydrostatic pressure at the core outlet remains constant, while the momentum vector from core is primarily directed into the downcomer (pipe 167). This results in heated liquid being pushed through the downcomer, causing a reduction in the flow through the core, as depicted in Figure 17. Additionally, a reduction in cladding temperature is observed (see Figure 16) due to decreased power level. The power drop is caused by reactivity feedback, triggered by hot coolant entering the core inlet at 35 seconds. Notably, the heated coolant enters the core inlet only for model E.

Flashing instability When simulating unprotected RIA accidents for natural circulation systems, the models might exhibit flow instabilities. When saturation parameters are reached at the maximum power heated channel outlet (pipe 1), saturated boiling commences and steam formation occurs. As the void fraction increases, the hydrostatic pressure decreases, leading to a reduction in saturation parameters within the channel, which causes additional saturation boil-

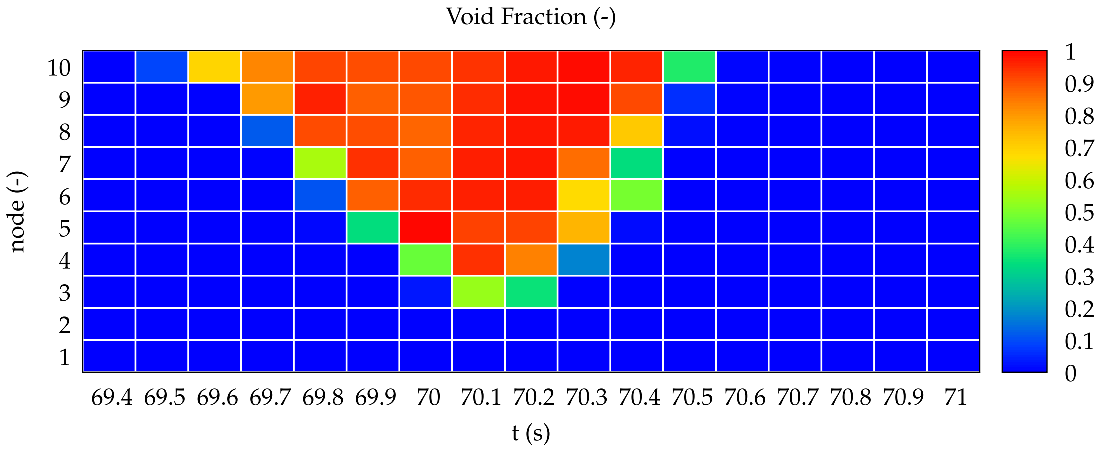

Figure 18. Void fraction oscillations in the heated channel with maximal power (pipe 1) – (RIA 1.2 β_{ef} in 0.3 s without SCRAM) – Model A.

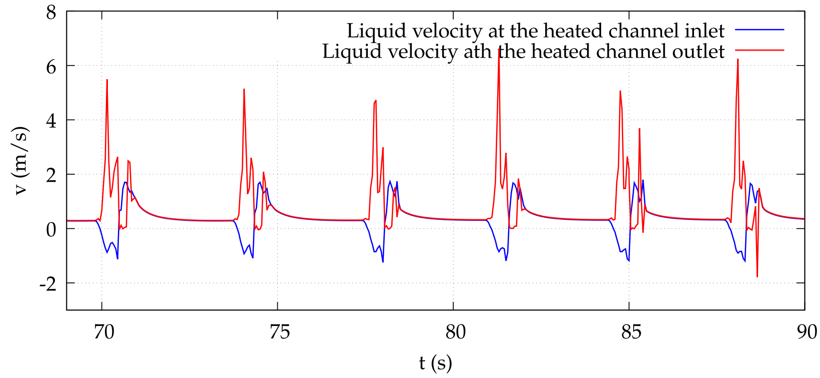

Figure 19. Liquid velocity at the channel outlet (RIA 1.2 β_{ef} in 0.3 s without SCRAM) for different models.

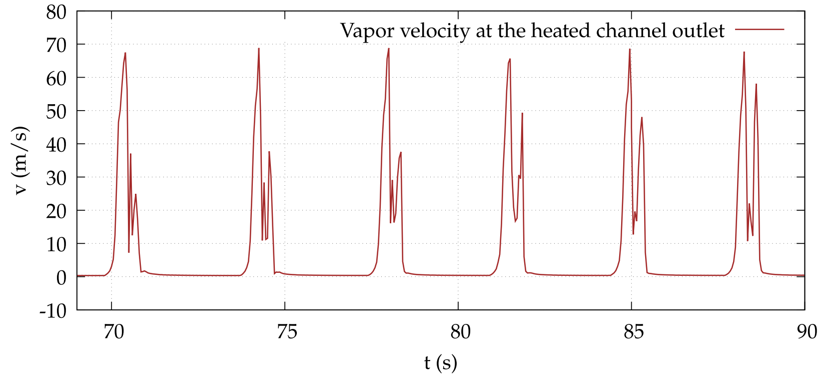

Figure 20. Vapour velocity at the channel outlet (RIA 1.2 β_{ef} in 0.3 s without SCRAM) for different models.

ing. The heated channel will rapidly fill with steam (see Figure 18). The rapid void increase causes the coolant to be pushed out from both the outlet and inlet (see negative inlet velocity in Figure 19) and increase the vapour phase void fraction in unheated upper and lower parts of the fuel assembly (pipe 17, 28, 38) and the branch 53. Subsequently, the subcooled water flows back into the heated channel and natural circulation begins to establish. The oscillation ends when initial parameters are reached. Each of the models exhibited oscillations with the same period (about 3 s) and amplitude. The liquid and vapour velocity at the hot channel inlet/outlet is illustrated in Figures 19 and 20. Several studies show that RELAP5 is able to accurately simulate natural circulation two-phase flow instabilities for low pressure conditions [8]. In case of the VR-1 reactor, the instability occurrence in the heated channel for long-term natural circulation without SCRAM is expected.

6. Conclusion

The aim of this work was to contribute to the standardisation of the application of SYS-TH codes to research reactors, to analyse the impact of nodalisation on the transient states in the VR-1 reactor using the RELAP5 code, and to propose the most suitable approach. Several models with different approach of reactor vessel nodalisation were introduced. The capacity to validate the RELAP5 model through experimental measurements is significantly constrained by the nominal power of the research reactor, which is approximately 100 W (500 W for 70 hours per year). This limitation poses challenges in accurately measuring variations in coolant temperature and flow velocity.

It was found that in the case of model A, without possible mixing of the coolant above the reactor core, there is a significant increase in the flow due to the chimney effect and a substantial temperature gradient. For natural circulation, power and flow are interdependent, leading to a different system behaviour. Models allowing the recirculation show less pronounced temperature differences in the volume above the core and the lower flow. On the contrary, the study also presents a limiting case that is not physically justified and represents the boundary behaviour of the system in which the chimney effect is significantly reduced. The effect of choosing the reactor shutdown delay time and the instability of the two-phase flow in the hot channel was also presented.

The findings obtained in this work can be used to standardise the modelling of research reactors using system thermal-hydraulic codes such as RELAP5. Models B, C, and D, which allow for recirculation, can be considered the most realistic when compared to other pool-type reactor simulations such as [9] and [10]. One has to keep in mind the lack of experimental measurement data for the VR-1 reactor. For some applications, models B, C, and D might be inapplicable. Future research might discuss the possible implementation of directional loss coefficients in the FCV junctions to obtain corresponding flow patterns and stratification.

Results from SYS-TH code application on research reactors can be used for modeling passive systems based on natural convection. Similar characteristics and difficulties can help to transfer experiences. Simulation of natural circulation is proving to be crucial for the future SYS-TH code development and implementation.

List of symbols

LOCA Loss of coolant accident

LOFA Loss of flow accident

RIA Reactivity induced accident

SYS-TH System thermo-hydraulics

FCV Free coolant volume (refers to volume 40 and 166, see Figure 4)

SV Stratified volume (refers to volume 40, see Figure 4)

Acknowledgements

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References

[1] T. Hamidouche, A. Bousbia-Salah, E. K. Si-Ahmed, F. D'Auria. Overview of accident analysis in nuclear research reactors. *Progress in Nuclear Energy* **50**(1):7–14, 2008. https://doi.org/10.1016/j.pnucene.2007.11.089

[2] L. Frybortova, J. Rataj, L. Sklenka, et al. The training reactor VR-1 – 30 years of operation. In *2020 21st International Scientific Conference on Electric Power Engineering (EPE)*, pp. 1–7. 2020. https://doi.org/10.1109/EPE51172.2020.9269254

[3] T. Bily, J. Rataj, O. Huml, O. Chvala. Effect of kinetics parameters on transients calculations in external source driven subcritical VR-1 reactor. *Annals of Nuclear Energy* **123**:97–109, 2019. https://doi.org/10.1016/j.anucene.2018.09.007

[4] J. Rataj, et al. Bezpečnostní zpráva školního reaktoru VR-1 [In Czech; Safety analysis report of the training reactor VR-1]. Tech. rep., KJR FJFI ČVUT v Praze, 2017.

[5] The RELAP5 development team. RELAP5/MOD3 code manual. Models and correlations. Tech. Rep. NUREG/CR-5535-V4, Nuclear Regulatory Commission, 1995.

[6] R. Henry, I. Tiselj. Computational fluid dynamic model of TRIGA Mark II reactor. In *22nd International conference nuclear energy for New Europe (NENE)*, p. 209. 2013.

[7] J. Rataj, L. Sklenka. Limity a podmínky pro trvalý provoz školního reaktoru VR-1 [In Czech; Operational limits and conditions for the continuous operation of the VR-1 training reactor]. Research Report CTU-14117-S-004-14, ČVUT FJFI, Katedra jaderných reaktorů, Prague, 2014.

[8] Q. Wang, P. Gao, X. Chen, et al. An investigation on flashing-induced natural circulation instabilities based on RELAP5 code. *Annals of Nuclear Energy* **121**:210–222, 2018. https://doi.org/10.1016/j.anucene.2018.07.035

[9] P. A. Reis, A. L. Costa, C. Pereira, et al. Assessment of a RELAP5 model for the IPR-R1 TRIGA research reactor. *Annals of Nuclear Energy* **37**(10):1341–1350, 2010. https://doi.org/10.1016/j.anucene.2010.05.013

[10] C. Introini, D. Chiesa, M. Nastasi, et al. A complete CFD study on natural convection in the TRIGA Mark II reactor. *Nuclear Engineering and Design* **403**:112118, 2023. https://doi.org/10.1016/j.nucengdes.2022.112118