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# Simulation of hypothetical small-break loss-of-coolant accident in modernized nuclear power plant

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**Abstract.** Nuclear power plant simulators are used for training and maintaining competence in order to ensure safe and reliable operation of nuclear power plants throughout the world. Simulators shall be specified to a reference unit and its performance validation testing shall be provided. The purpose of the study was to predict the response of the Krško modernized nuclear power plant (NPP) to a small-break loss-of-coolant accident (SB LOCA) and to use the reference calculation for validation of the Krško full-scope simulator (KFSS). For reference calculations the RELAP5/MOD2 best estimate system code was used and a verified plant specific standard input model of the Krško NPP, adapted for 2000 MWt power (cycle 17) and new (replacement) steam generators. The RELAP5/MOD2 calculated reference results suggest that the plant system response to an SB LOCAs with the break in the cold leg is the slower the smaller is the break area, and vice versa. The core heatup occurred in most of the calculated cases. A comparison of the results obtained with KFSS cycle 19 and calculated reference results showed a good agreement and it indicates that the simulator validation testing in 2000 for this kind of accident was successful.

**Key words:** accident analysis, full-scope simulator, RELAP5/MOD2, nuclear power plant

## Simulacija hipotetične male izlivne nezgode v posodobljeni jedrski elektrarni

**Povzetek**. Simulatorji jedrskih elektrarn so namenjeni usposabljanju in vzdrževanju zmožnosti osebja, da zagotavlja varno in zanesljivo obratovanje. Simulator mora, kolikor je mogoče, posnemati delovanje referenčne elektrarne in ga je treba pred prvo uporabo validirati. Glavni namen te raziskave je bil napovedati odziv posodobljene jedrske elektrarne Krško na malo izlivno nezgodo in referenčni izračun uporabiti tudi za validacijo popolnega simulatorja. Za referenčni izračun smo uporabili realistični sistemski program RELAP5/MOD2 in že preverjeni splošni vhodni model za jedrsko elektrarno Krško (JEK), vendar smo ga priredili za 2000 MW toplotne moči (17. gorivni cikel) in upoštevali nove zamenjane uparjalnike. Referenčni izračuni s programom RELAP5/MOD2 so pokazali, da je bil odziv elektrarne na malo izlivno nezgodo z zlomom v hladni veji tem počasnejši, čim manjši je bil presek zloma, in narobe. V večini izračunanih primerov se je sredica pregrela. Primerjava rezultatov, pridobljenih s popolnim simulatorjem Krško za 19. cikel, z rezultati referenčnih izračunov je pokazala dobro ujemanje in kaže, da je bila preveritev simulatorja v letu 2000 za to vrsto nezgode uspešna.

**Ključne besede:** analiza nezgode, popolni simulator, RELAP5/MOD2, jedrska elektrarna

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## 1 Introduction

Nuclear power plant simulators are used for training and maintaining competence in order to ensure safe and reliable operation of nuclear power plants throughout the world. Therefore the Krško nuclear power plant (NPP) decided to obtain a full-scope simulator upon the plant modernization made in 2000. The plan for verification and qualification of the Krško full-scope simulator (KFSS) is described in [1]. In the preparatory phase the impact of the specific full-scope simulator on the nuclear safety of Krško NPP was also assessed [2].

The full-scope simulator is a simulator incorporating detailed modelling of systems of the reference unit with which the operator interfaces in the control room environment. The control room operating consoles are included. Such a simulator demonstrates the expected unit response to normal and off-normal conditions.

The simulator shall be specified to a reference unit and its performance validation testing shall be provided. Functional requirements for the full-scope nuclear power plant control room simulator used for operator training or examination are established in ANSI/ANS-3.5 standard [3]. Among other the standard requires also simulator performance validation. The intent of validation is to "ensure that no noticeable difference exists between the simulator control room and simulated systems when evaluated against the control room and systems of the reference unit" [3]. The baseline data order of preference to ensure simulator fidelity shall be as follows: (1) data collected directly from the reference plant, (2) data generated through engineering analysis with a sound theoretical basis, (3) data collected from a plant which is similar in design and operation to the reference plant, (4) data that do not come from any of the above sources, as for example operator experience, expectations, engineering judgment and Safety Analysis Report type of analysis.

In the case of the Krško NPP only the two first baseline data sources were used. Validation against the first data source can be found in [4]. In the present study, the second source baseline data for small-break loss-of-coolant accident (SB LOCA) were generated by RELAP5/MOD2 for KFSS validation (for details the reader is referred to [5]). The calculated reference data are highly important as no real cold leg SB LOCA transient data exists for Krško NPP. Namely, this is the design basis accident, which is not expected to occur in the plant lifetime.

The RELAP5/MOD2 is the best estimate thermal-hydraulic computer code. In the world, the use of the best estimate codes is a common practice for simulator validation; a recent example is a comparison between the best estimate RELAP5/MOD3 code and a simulator that works on a fuzzy network model [6].

The purpose of this paper is to present the calculated results of a spectrum of the SB LOCA that were intended to be used for KFSS validation (see plan [1]). Additionally, a comparison between the KFSS data obtained in 2002 and RELAP5/MOD2 thermal-hydraulic computer code calculation is shown for scenarios with 5.08 cm and 20.32 cm breaks, which were used also for verification of the KFSS during the Krško NPP modernization.

# 2 Methods used

## 2.1 Computer codes description

A full two-fluid non-equilibrium, non-homogeneous model is used in RELAP5/MOD2 to simulate one-dimensional two-phase flow. The basic thermal-hydraulic model uses six equations: two mass conservation equations, two momentum conservation equations and two energy conservation equations. The system of basic equations is enclosed with empirical correlations. For a more detailed description refer to [7].

The Krško full scope simulator uses ANTHEM2000, which is a ROSE (Real-time Object-oriented Simulation Environment) based version of ANTHEM thermal-hydraulic code. ANTHEM is a non-equilibrium, nonhomogeneous drift-flux model of two-phase flow. The thermal-hydraulic model uses six equations: conservation of liquid mass, gas mass and non-condensable mass, conservation of mixture momentum and conservation of liquid energy and gas energy. A more detailed description is given in [8].

## 2.2 Description of input models

As the basis for the performed analyses, the Krško NPP has delivered the verified input model for RELAP5/MOD2. The input model is documented in the plant reports which shows that it was verified for steady state and transients like SB LOCA, steam generator tube rupture, etc. For more details the reader is referred to [1]. The model consists of 309 volumes, inter-connected with 339 junctions, and 299 heat structures with 1622 mesh points. For the containment analysis a standard verified input model for the Krško NPP was used to assure the quality of the results [9]. The thermal-hydraulic input model for ANTHEM consists of 79 volumes and 106 junctions [4]. It can be seen that thermal-hydraulic input model for ANTHEM is simpler than the RELAP5/MOD2 input model. In both input models all important components of the plant are included. The input models consist of a reactor vessel and two closed coolant loops connected in parallel (primary side) and a separate power system conversion system provided for electricity generation (secondary side). Each of the two loops contains a reactor coolant pump, steam generator, loop piping, and important control and safety systems.

In the present analysis a spectrum of five different break sizes was considered with breaks of 2.54 cm, 5.08 cm, 7.62 cm, 10.16 cm and 20.32 cm (1 to 8 inch) in diameter, located in the cold leg of the reactor coolant system. Such breaks imply evaporation of water in the reactor vessel and discharge of reactor coolant through an opening. This is what is called the loss-of-coolant accident. Based on experience of simulating SB LOCA in the Krško NPP [10,11], models for analysing different scenarios of SB LOCA transient were provided. These included break models for five selected breaks and additional triggering logic for various systems for the case with assumed loss of off-site power. Because of the oscillatory behaviour of certain primary system parameters during operation of the low pressure injection system (LPIS) the calculation was aborted around 3000 s for scenario with 20.32 cm of the break size. Therefore slight changes were introduced into the original input model. These changes included certain junction area reductions in the reactor vessel. The calculation was then normally completed.

## 2.3 Scenarios description

In order to investigate basic phenomenology during SB LOCA, the scenarios were simple. No operator actions were specified in the scenarios except reactor coolant pump trip per emergency operating procedures while the protection system logic was modelled. The protection system logic senses a condition requiring safety systems actuations. It has two independent and redundant protection trains. Each protection train actuates only safety systems associated with it. Safety systems are designed to mitigate consequences of hypothetical design bases accidents. For SB LOCA the most important safety system is the emergency core cooling system designed to cool the core. It consists of a high pressure safety injection (HPSI) pump, accumulator (tank with borated water under pressure) and low pressure safety injection (LPSI) pump. The pumps deliver water from a refuelling water storage tank to the reactor vessel. An important safety system is also an auxiliary feedwater (AFW). It provides water to the steam generator on the secondary side to maintain the heat sink. In the analysis, both protection trains with their associated safety systems were assumed available for all break sizes except for 20.32 cm (8 inch), where loss of the off-site power was assumed together with a successful emergency diesel generator start. After the emergency diesel generator start only one protection train was available.

The initiating event was opening the valve simulating the break in the cold leg. After the break opening a rapid primary pressure drop followed. It caused the reactor trip upon the low pressurizer pressure signal at 12.99 MPa. The reactor trip further caused the turbine trip. The safety injection (SI) signal was generated upon the low-low pressurizer pressure signal at 12.27 MPa. The SI signal actuated the HPSI and LPSI pumps and motor driven AFW pumps. Upon the SI signal the two main feedwater pumps were tripped, too. The reactor coolant pumps were tripped manually upon the subcooling signal according to the emergency operating procedures allowing additional 60 s for operator actions. The above described sequence of events was typical for all the analysed cases. For the description of typical physical phenomena and processes occurring during SB LOCA the reader is referred to [10,12].

## 3 Results and discussion

The SB LOCA reference calculations with RELAP5/MOD2 best-estimate computer code were performed for 200 s of steady-state condition and 10000 s of transient condition. In the next figures the calculated plant response dependent on the break size and comparison between KFSS results from the year 2002 and RELAP5/MOD2 reference calculations for 5.08 cm and 20.32 cm equivalent diameter break size are shown.

The parametric reference calculations with RELAP5/MOD2 using a modified input model are shown in Figs. 1 to 3. The sequence of events is determined by the primary pressure. It can be seen that the calculated plant response during SB LOCA largely depends on the break size. Larger is the break size faster is the pressure drop (Fig. 1), the core uncovers faster and earlier (Fig. 2) and the core is heated up earlier (Fig. 3). Fig. 3 also shows that the largest peak cladding temperature (PCT) of 831 K was calculated for 5.08 cm break. Besides, due to the break size the scenarios may change for different operator interventions and assumed safety systems available.

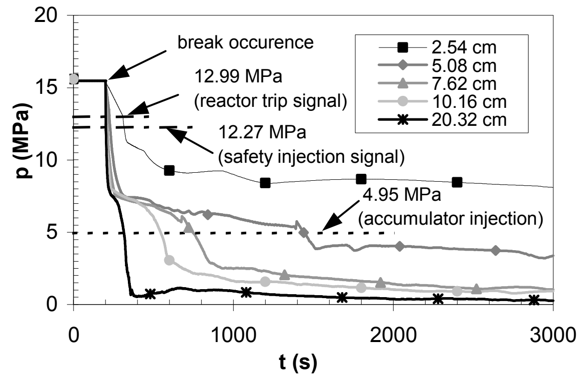


Fig. 1: RELAP5/MOD2 calculated primary system pressure for a spectrum of break sizes

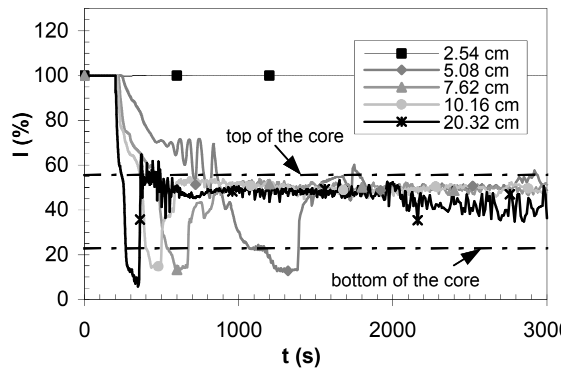


Fig. 2: RELAP5/MOD2 calculated reactor vessel level for a spectrum of break sizes

The results of best estimate calculations can be used also for operator training and independent evaluation of licensee SB LOCA calculations. The best practice in the world also shows that simplified best estimate codes are used for thermal-hydraulic models employed in simulations [13].

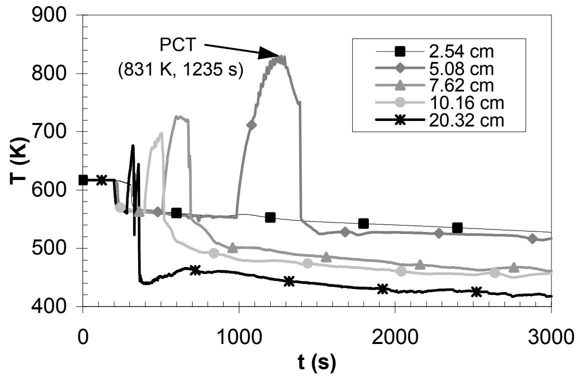


Fig. 3: RELAP5/MOD2 calculated clad temperature for a spectrum of break sizes

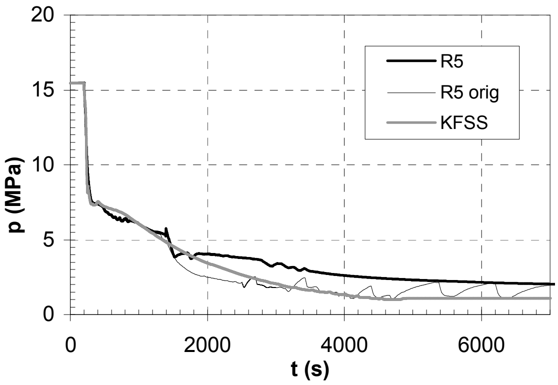


Fig. 4: Comparison of the primary pressure between RELAP5/MOD2 and KFSS for 5.08 cm break

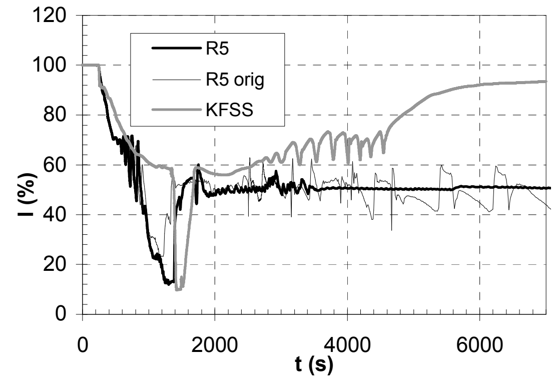


Fig. 5: Comparison of the reactor vessel level between RELAP5/MOD2 and KFSS for 5.08 cm break

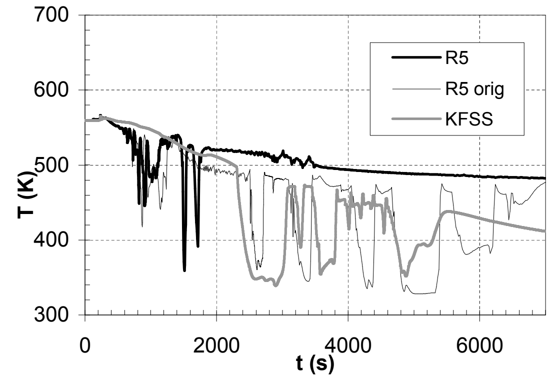


Fig. 6: Comparison of the cold leg temperature between RELAP5/MOD2 and KFSS for 5.08 cm break

In the KFSS validation process scenarios with 5.08 cm and 20.32 cm break were used. Two reference calculations were performed, with the original and the modified RELAP5/MOD2 input model (see Section 2.2). Important plant parameters like primary system pressure, reactor vessel level and cold leg temperature were compared. This set of parameters does not include typical parameters used in the RELAP5/MOD2 analysis like cladding temperatures, break flow, pressure drops across the loop etc. because the plant is not instrumented for measuring such parameters and so these parameters are not available for operator decision making. Besides, such parameters are not proposed in the guidelines for simulator operability testing included in ANSI/ANS-3.5 [3]. According to ANSI/ANS-3.5 standard, one of the most important criteria for validation is that any observable change in simulated parameters corresponds in the direction to those expected from an actual or best estimate response of the reference plant to the malfunction.

In Figs. 4 to 6 simulator data (marked KFSS) are compared to RELAP5/MOD2 calculations with original (marked R5 orig) and modified input model (marked R5). It can be seen that an agreement between the simulator and RELAP5/MOD2 prediction for scenario with 5.08 cm break is satisfactory. All important physical phenomena were simulated including core uncovery causing core heatup. Nevertheless, the pressure obtained by KFSS decreased faster causing start of the LPSI pump. Therefore, the KFSS level in Fig. 5 recovered around 5000 s. Figs. 4 to 6 also show that input model modifications made for scenario with 20.32 cm break have an adverse effect on the results in the case of 5.08 cm break size. The reason for this is that the primary system pressure was high enough not requiring operation of the LPSI pump for which modifications were made.

A comparison between the KFSS results and RELAP5/MOD2 results for scenario with a 20.32 cm break size is shown in Figs. 7 to 9. The simulator results were available for 1500 s only. During this period the influence of the modified input model on the results was negligible. The primary system pressure shown in Fig. 7 is in a good agreement except for a slight overprediction around 400 s. Fig. 8 shows

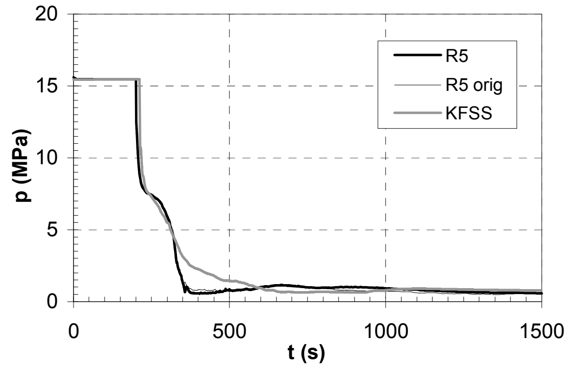


Fig. 7: Comparison of the primary pressure between RELAP5/MOD2 and KFSS for 20.32 cm break

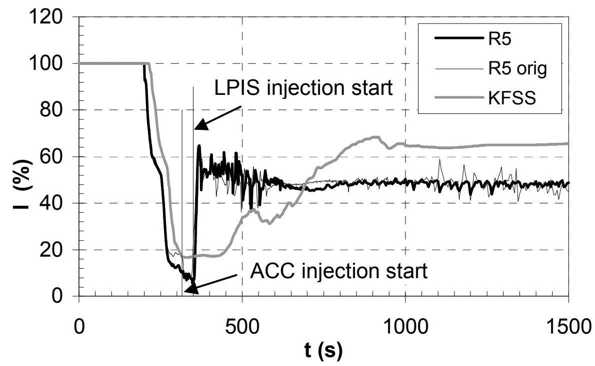


Fig. 8: Comparison of the reactor vessel level between RELAP5/MOD2 and KFSS for 20.32 cm break

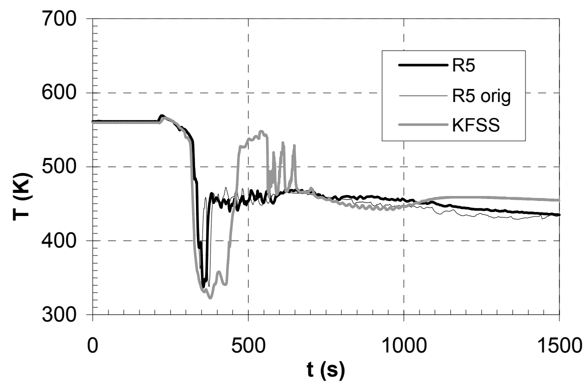


Fig. 9: Comparison of the cold leg temperature between RELAP5/MOD2 and KFSS for 20.32 cm break

the reactor vessel level. After the break occurrence the level decreases until accumulator injection (316 s) which causes with some delay level recovery. In the R5 calculation the accumulators empties at 365 s and the LPSI pump then keeps the level (actuated around 350 s). The disagreement in the level is because of differences in the primary system pressure. The accumulators (ACC) and LPSI pump start to inject cold water when the primary system pressure is below 49.5 MPa and 11.3 MPa, respectively. The injected water then cools the primary system (see Fig. 9). In the case of R5 calculation the initial temperature drop is caused by accumulator injection while LPIS keeps the temperature below 480 K.

A comparison of parameters in Figs. 4 to 9 shows that the direction of changes for the simulated parameters corresponds between the simulator and RELAP5/MOD2 predictions and is thus assumed to be correct. Nevertheless, there are some quantitative differences in the results. The first reason is that the RELAP5/MOD2 thermal-hydraulic input model is more detailed than the KFSS thermal-hydraulic model. Secondly, the previous study [11] showed that the best estimate RELAP5 code has uncertainty therefore some quantitative differences in the frame of uncertainties are expected (for example up to 25 K in the case of cold leg temperature). Nevertheless, quantification of uncertainties is far beyond the scope of this study since qualitative agreement is required for the simulator. On the other hand, as the ANTHEM code is integrated in the ROSE, the simulator offers better and easier modelling of the logic control, protection and safety systems, relief, safety and isolation valves, pumps of the emergency core cooling system and details in modelling of other components of pipelines. These systems and components also affect results. Other important sources of discrepancies are different times of the emergency core cooling system activation (20.32 cm break). They depend on the primary system pressure, which governs the transient progression. Finally, the simulator data were measured in 2002 on the simulator using current valid setpoints and existing core (cycle 19) for the purpose of this comparison. Namely, the data used for simulator verification were not electronically archived.

## 4 Conclusions

With the RELAP5/MOD2 a hypothetical SB LOCA in the modernized Krško NPP was simulated. The input model used for the analysis, assumptions used and modifications made to the input model were described. The predicted results of the SB LOCA analysis with RELAP5/MOD2 were used for KFSS performance validation testing. The comparison performed later showed that acceptance criteria for simulators were met and that simulator results are in good agreement with the best estimate calculation performed by RELAP5/MOD2. Besides, for the simulator validation the best estimate SB LOCA analysis can be applied for operator training on physical phenomena and processes. This analysis can be also used as an independent analysis to the license SB LOCA calculation performed by the original nuclear steam supply system designer - supplier. However, the conclusions on the results would be limited because uncertainty was not evaluated.

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